CR1500001

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

February 18, 2000

MEMORANDUM TO: Chairman Meserve

FROM: Dana A. Powers, Chairman
ACRS

SUBJECT: RESPONSE TO FOLLOW UP QUESTIONS POSED BY INDIVIDUAL COMMISSIONERS FOLLOWING THE NOVEMBER 4, 1999 MEETING WITH THE ACRS

Attached is the response to the follow up questions posed by you and individual Commissioners in your memorandum of December 22, 1999. Several of the questions relate to matters which are still under Committee deliberation and are works in progress, such as risk-informing appendices A and B to 10 CFR Part 50; use of core damage frequency; and large, early release frequency as permanent risk metrics in the regulatory process. The response reiterates the Committee's concern about the analytic capabilities that are available to support the move towards a risk-informed regulatory system.

I and/or individual members would be willing to meet and discuss with individual Commissioners any follow up to this response.

cc: Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
EDO
SECY

ACRS
Larkin/SP

CR150

February 17, 2000

Responses to Questions Posed by Individual Commissioners Following the November 4, 1999, Meeting with the ACRS

Question 1. License Renewal

During the briefing the ACRS mentioned a concern about moving the allowed license renewal application period too far ahead of the end of the license period because some one-time inspections associated with renewal should be undertaken after the longest possible service (i.e., as close to the end of the 40-year life as possible). In that light, please describe the Committee's position on:

- **the plant attributes and one-time inspections of most concern.**
- **whether there is a specific length of time prior to the end of the license period before which renewal should not be considered in light of the importance of results of one-time inspection to license renewal.**

Response:

The aging-related degradation inspections committed to in the Calvert Cliffs license renewal application are a small, eclectic collection of one-time inspections of components that are not included in any other aging management program. Some systems that will have components subjected to one-time inspections at Calvert Cliffs are listed below:

SYSTEM	INSPECTION FOR
Reactor pressure vessel internals	stress corrosion cracking; stress relaxation
Auxiliary feedwater system	erosion; general corrosion
Component cooling water system	wear; erosion; general corrosion
Compressed air system	general corrosion
Containment spray system	general corrosion
Emergency diesel generator	general corrosion; erosion; microbiologically induced corrosion; wear; cycle fatigue
Feedwater system	general corrosion; erosion

Heating, ventilation and air conditioning system	general corrosion; erosion; elastomer degradation
Service water system	general corrosion; selective leaching

From this list, it is apparent that components from significant safety systems are involved in the one-time inspection program.

The purpose of the one-time inspections is to confirm that possible, but unlikely, age-related degradation is not occurring or that if it is occurring, it will not affect the component's intended function over the extended period of the license. The one-time inspections attempt to detect degradation, and if found determine how far the degradation has progressed and whether it poses any significant consequences over the period of an extended license. The one-time inspections are, then, intended to support a negative hypothesis. Because of the time dependency of age-related degradation and the lack of adequate understanding of age related degradation mechanisms, the confirmation of assumptions and the reductions in uncertainties, are best achieved by doing inspections as late in plant life as possible. This maximizes the likelihood that degradation will have progressed to the point that it is detectable and that meaningful estimates of its progression during the period of the extended life can be made. Inspections, therefore, should be done in the last five years and certainly no earlier than the last ten years of the current license. Allowing the one-time inspections to be done at the start of the second twenty years of the current 40-year license is not going to bolster confidence that degradation processes will be detected or assessed properly.

Question 2. Low-power and Shutdown Operations Risk; and Multiple SSCs Out of Service During Maintenance

The slides concerning low-power and shutdown operations risk suggest that ACRS does not have the same level of comfort with the adequacy of the analysis as does the staff and the industry. Please discuss the deficiencies, if any, which the Committee believes exist in current low-power and shutdown risk analyses and activities, and what additional research, information and activities are needed to resolve those deficiencies.

Response:

It is our position that the staff does not have an adequate understanding of risk during low-power and shutdown operations **[Rreference 2]**. Certainly, the understanding is not comparable to the understanding we have of the risk during power operations even though scoping assessments of shutdown operations would have us believe that the annualized risk is similar to that of power operations. Because shutdown operations should occur only a small fraction of the time during a

year, the results of scoping studies performed for the staff suggest that the conditional risk during shutdown operations is quite high.

We have been told by representatives of the former Office for Analysis and Evaluation of Operational Data (AEOD) that 58% of the events to which NRC has assigned Augmented Inspection Teams involved low-power or shutdown events. We do not know whether such events should be considered potential precursors to serious events because credible analyses with proper attention to unproceduralized intervention have not been done. Still, one of the highest conditional core damage frequencies assigned by the NRC's Accident Sequence Precursor program was for a shutdown event involving the loss of coolant inventory (Wolf Creek draindown event). **[Reference 3]**

We are, however, not persuaded that existing capabilities for shutdown risk analysis are adequate. We have pointed to complications that affect risk analysis for shutdown operations including the dynamic nature of risk, the variability in plant configurations during shutdown operations, the difficulties in analyzing human performance, and the challenge of realistic treatment of unproceduralized intervention capabilities possible during plant shutdowns. ACRS members have called attention to the difficulty of defining success criteria for shutdown events, complexities of radionuclide releases from fuel during shutdown events, and the complications of radionuclide transport during shutdown events that are not included in existing radionuclide behavior models. There is not a good understanding of: the fuel failure mechanisms in low pressure water - air environments; how fuel burns; the nature of fission product species in oxygen-rich atmospheres; the nature of fission product releases under shutdown conditions; or the characteristics of aerosols formed. The source term characteristics for shutdown accidents could be significantly different from those anticipated for accidents during power operations.

We are aware that economic pressures have driven the nuclear industry to pay much closer attention to both efficiency and safety during shutdown operations. There can be no doubt that these industry efforts have reduced risk, though we have no way of knowing by how much. Events still occur during low-power and shutdown operations. Analyses done by the industry and presented at a workshop held by the Office of Nuclear Regulatory Research (RES) staff indicate that shutdown operations still make significant contributions to risk even with the improved processes for planning and limiting shutdown operations.

If low-power and shutdown operations simply increased the estimated risk of a plant by factors on the order of two, we would not be especially concerned. We are not convinced that core damage frequency estimates often quoted to one or even two significant figures are, in fact, accurate to a factor of two. Concern arises when results of risk analyses are used to assign safety significance to particular structures, systems, and components. Omission of important elements of a plant's risk profile can create enormous errors in the categorization of structures, systems, and components according to their risk significances. We have, therefore, recommended that the NRC staff develop methodologies and conduct analyses of representative plants to develop an

understanding of risk during low-power and shutdown operations comparable to the understanding we now have of risk during power operations.

Question 3. The Balance between PRA Results and Defense in Depth

Please elaborate on how "PRA results" and "defense-in-depth" can complement each other and how can methods be developed to establish when risk information should become more dominant for decisionmaking relative to deterministic considerations, while the defense-in-depth philosophy is appropriately maintained?

Response:

The defense-in-depth safety philosophy can be implemented into a risk-informed regulatory system in two ways, which should be distinguished. The first implementation is traditional. Defense in depth is used to compensate for the inadequacies in our abilities to assess risk and to understand all the sources of uncertainty. The second implementation involves balancing the various elements of safety. At its simplest, this implementation might be viewed as striking a balance among the measures directed at preventing accidents and measures taken to mitigate accidents. A more involved implementation might include striking a balance among the types of events that can lead to accidents, measures to intervene against accidents once they are started, as well as measures to mitigate accident consequences. The balance that is to be achieved in this implementation would be a matter of policy.

Probabilistic risk assessments can be used to confirm that regulatory objectives of acceptable risk and balance are achieved. This type of defense in depth would be preserved in a risk-informed regulatory system no matter how advanced the methods of risk assessment. Traditional forms of defense in depth would still be used to various extents in a risk-informed regulatory system where risk assessment methods were incomplete or the uncertainties associated with risk predictions were objectionably large. Traditional implementation of defense in depth would become less important as risk analysis methods improved.

Question 4. Low-power and shutdown operations risk

a. Comment on the impact on total risks during low power and shutdown conditions since the vulnerabilities occur at low decay heat with ample coolant inventory at low RCS pressure, and with sufficient time to take corrective and compensatory actions.

Response:

Risk studies by the staff, licensees, and foreign institutions have yielded values of the core damage frequency during low-power and shutdown operations comparable to those for full-

power operations. Scoping studies by the staff **[References 4, 5]** have focused on operational modes involving open containments and even open reactor coolant systems. Hazardous conditions identified in these studies typically involved low coolant inventories. Accidents can be initiated by a further loss of coolant or by loss of the heat sink. Under conditions of low decay power, accidents will develop slowly. There will be long periods for detecting hazardous conditions. It is a good thing there is this slow accident progression, since in some of the recent events days passed before hazardous plant conditions were detected. Once detected, hazardous plant conditions can be corrected easily since the containment and even the reactor coolant system are open. Unproceduralized actions for recovery are imaginable even when many safety systems are not available. Current PRA technology and the technology used for the studies of shutdown risk are not capable of accounting for these unproceduralized corrective actions. We are concerned that this limitation of the current PRA technology for low-power and shutdown risk assessment may give a distorted picture of critical structures, systems, and components.

Although the treatment of unproceduralized actions may lead to conservatism in the PRAs, other parts of the risk analysis may not be conservative. The scoping assessments of risk have screened out some operational modes as unimportant to risk. The screening done may not be consistent with experience that suggests startup and shutdown modes may involve substantial risks due to human error. Furthermore, reactivity excursions caused by human error can occur during startup and shutdown operations and can rapidly lead to core damage.

This leads to a significant point that we want to make. If the concerns of shutdown risk were simply about doubling or tripling of the core damage frequency for a plant because of non-operational risks, the issue would be almost academic. We believe that current estimates of risk are not more accurate than factors of two or three for operating reactors. The real issue is the categorization of systems, structures, and components in accordance with their risk significance. We are concerned that this categorization could be in error. Also, if the knowledge base on risk still relies on the scoping studies done by the staff for two plants, then other risk-significant aspects of low-power and shutdown operations may be neglected. In addition, the risk outliers among the current fleet of plants may not be detected.

b. Discuss your opinion on how the new language in 50.65(a)(4) should take care of concerns of configuration control at power as well as at low power and shutdown?

Response:

It was our position that the language of 10 CFR 50.65(a)(4) should have been revised to be, "Scope of the assessment may be limited to structures, systems, or components that a risk informed evaluation process has shown to be significant to public health and safety for the proposed configuration" **[Reference 6]**. We recommended this language for two distinct reasons. The most obvious of reasons is the simple, theoretical result that the risk importances of structures, systems, and components change as the plant configuration changes. It is entirely

possible that a structure, system, or component found to have low-risk importance for a PRA analysis with annualized, average availabilities will assume a much greater conditional risk importance in the specific configurations that arise while doing maintenance on a particular plant at a particular time when some redundant and diverse capabilities may not be available. The significant swings in risk that occur during plant operations are well established and can even be quantified to some extent. Episodic excursions in risk by factors of up to about 100 do occur.

The second reason for recommending specific language in 10 CFR 50.65(a)(4) is the truism of operations that one ought never deliberately present to operators an unanalyzed plant configuration. We have difficulty imagining engineering analyses that could be the basis for contra-indication of this time-tested principle of safety management of hazardous operations.

The staff is incorporating the thought behind the language we had recommended for 50.65(a)(4) into the proposed Revision 3 to Regulatory Guide 1.160 **[Reference 7]**. We are still in the process of reviewing this Regulatory Guide, but we are confident that adequate guidance will be provided in this Guide.

We are not persuaded that the requirements of 10 CFR 50.65(a)(4) obviate the need to develop further the capabilities to analyze risk during low-power and shutdown operations. An important element of the rule is the risk analysis of shutdown plant configurations. An important element of the requirements of the rule is the ability to categorize the risk significance of structures, systems, and components. We believe that the staff should have tools for such analyses and characterizations comparable to those available for analysis of power operations. We do not believe reliable judgments can be made on the relative risks associated with online maintenance versus maintenance during outages without tools to assess risks during outages adequately.

Question 5. Maintenance rule.

In the recent draft of Section 11 of NUMARC 93-01 the scope of the 50.65(a)(4) assessment includes those SSCs modeled in the PRA and other high safety significant SSCs identified through the risk significance determination process described in Section 9.3 (i.e., importance measures and the expert panel considerations). I understand that the NRC staff is ready to endorse this position. Discuss how the quality of the PRA takes into consideration multiple components taken out of service at the same time. Discuss the importance of state-of-the-art PRAs (different levels) for risk-informing Part 50.

Response:

The proposed approach to the determination of the scope of structures, systems, and components for 10 CFR 50.65(a)(4) assessments is to use the PRA logic structure to determine the plant systems and their functions (i.e., system trains) that fall within the scope, rather than to identify individual components using importance measures. This requires less rigor in many of the PRA

elements. For example, the PRA quantification approach, the parameter estimates for failure rates, initiating event frequencies, and the detailed success criteria, which are often sources of concern and uncertainty, do not impact the decisions for scope determination. The parts of the PRA that need to meet the "quality test" are: (a) the choice of the initiating events and the plant functions required to mitigate their impact, (b) the identifications of systems that can perform those functions, and (c) the support systems that are required for those systems. Additionally, subtle interactions or dependencies between systems should be considered. The scope of the structures, systems and components would be limited to those systems and their functions considered to be important in determining the appropriate level of safety. Thus, the risk impact of multiple components taken out of service at the same time would be evaluated appropriately when the PRA is requantified by considering the affected system trains being out of service.

Over several NRC/NEI meetings, the staff articulated its expectations on the quality of PRAs needed for determining the structures, systems, and components scope in 50.65(a)(4) assessments. These expectations are included in Section 11.3.3, "Scope of Assessment for Power Operating Conditions" of the revised NUMARC 93-01 **(Reference 8)**. The important attributes of the PRA used to define the 50.65(a)(4) assessment scope are:

(a) The PRA should reflect the as-built plant and the plant operating practices, and
(b) The PRA should include both frontline/support system dependencies and support system/support system dependencies to the extent that these inter-system dependencies would have a significant effect on the key plant safety functions. If the modeling of inter-system dependencies is determined to be inadequate, the licensee should either revise the PRA to address the inter-system dependencies, or add the structures, systems, and components (SSCs) to the 50.65(a)(4) assessment scope.

In addition, Appendix E of Section 11.0 of NUMARC 93-01 provides information on methods to evaluate the PRA for use in defining the 50.65(a)(4) scope. There is also guidance on the role of the plant Expert Panel to compensate for known limitations of the PRA for determining the SSC scope. For example, the Expert Panel should use qualitative judgment to determine the low safety-significant SSCs and SSCs important to containment performance that should be in the scope. In the case of low safety-significant SSCs, as identified by the PRA, and those outside the scope of the PRA, the Expert Panel judgment for excluding SSCs from the 50.65(a)(4) scope should be based on engineering analyses and insights, operational experience, and information from licensing basis documents and design basis accident analyses.

It has been our position that with the qualitative understanding of risk derived from the in-depth study of five representative plants (NUREG-1150) **[Reference 9]** and, to a lesser extent, the insights gained from the Individual Plant Examination (IPE) submittals, it should be possible to make great strides in developing a risk-informed version of 10 CFR Part 50 including Appendix A on the General Design Criteria and Appendix B on Quality Control and Quality Assurance.

The eventual availability of insights from the Individual Plant Examination of External Events (IPEEEs) submittals will add to our capability to make these changes.

On the other hand, once a risk-informed version of 10 CFR Part 50 is available and is used by licensees, high-quality PRA capabilities must be available to both the staff and to licensees that choose to be regulated by the risk-informed version. Regulatory Guide 1.174 **[Reference 10]** establishes a correlation between the quality and scope of a PRA and the level of application that such an assessment can support. This correlation is based on the technical requirements and the level of detail that the risk-assessment model needs to include to effectively support the application. As 10 CFR Part 50 becomes more risk informed, thus allowing more complex and technically challenging applications, the detail, scope, and capability of the risk-assessment models supporting such applications will have to increase. It is apparent that the scope and depth of PRAs will continue to be driven by the applications that they are meant to support.

An issue of some interest is whether risk analysis capabilities need to be taken to Level I or to Levels II and III. At present, core damage frequency and large, early release frequency are used as risk metrics on a voluntary basis. With these metrics, only Level I PRAs and some limited analysis of early containment failures under accident conditions will be needed. Such capabilities are within the current state of the art for operational events. We have suggested that the Commission formally acknowledge core damage frequency as a risk metric and establish a safety goal using this metric that can be applied to individual plants. **[Reference 11]**

We have not arrived at a position on whether use of core damage frequency and large, early release frequency as risk metrics should be a permanent situation in risk-informed regulation or the agency should move toward using actual risk metrics eventually. Clearly, using risk metrics would create a more coherent regulatory structure headed by the Commission's Safety Goal Policy. At this time, we believe it is more important that the Level I risk assessment capabilities be expanded in technically defensible ways to include all modes of plant operation and to include external events such as fire.

Question 6. Low-Power and Shutdown Operations Risk

Please provide the basis for the statement, "Current estimates of high risk during plant shutdown may be conservative" and in what context is the term "conservative" used?

Response:

The term "conservative" was used to indicate that the scoping studies of shutdown risk may have yielded unrealistically high estimates of risk posed by the mode of operation examined for each of the two plants. A major concern to us is that the scoping studies of risk were unable to account for unproceduralized recovery steps available to plant personnel. Since the containments and even the reactor coolant systems were taken to be open for the modes of operation studied in

the scoping effort, detection of loss of inventory or loss of heat sink would be expected to be prompt. Even with few safety systems available, plant personnel could improvise some method to deal with the situation.

We also recognize that arguments can be advanced to suggest that the risk estimates were not conservative. In particular, definitions of success paths for recovery may be overly optimistic and source terms associated with shutdown accidents may be unrealistically optimistic. These counter arguments simply lead to the point we have made in the past and continue to repeat. There is not now an adequately developed understanding of risk during low-power and shutdown operations.

Question 7. NRC Safety Research Program

Dr. Powers and I have discussed this matter in the past. I have expressed my view that the ACRS review would be of greater benefit to the Commission in the planning and budgeting processes if it identified not only areas where the ACRS believes more research is warranted, but also identified areas where ongoing or planned research is likely to be of limited value to the agency. Of course, such a review would have to be framed in the context of the agency's strategic and performance goals, and should recognize the importance of anticipatory research especially as it pertains to emerging technologies. Please describe the scope and focus of this year's examination of the NRC research program.

Response:

In our 1998 and 1999 reports [**References 12, 13**], we reviewed activities in several areas, covering most of the research programs carried out by the Office of Nuclear Regulatory Research (RES).

In our 2000 report, an advance copy of which was provided to the Commission on February 8, 2000, we present more of an overview. We examined the internal and external contexts that together determine the needs for research and the corresponding responses of the agency. We discussed how the role of NRC research has evolved and may develop in the future. Along the way, we described some major issues that the Commission may face and which we believe will require the development of a better knowledge base through appropriate research.

In our 1998 report, we made several recommendations about how the NRC should conduct its research, one of which was the need to achieve a closer tie between research activities and agency needs. In our 2000 report, we make some suggestions about how this could be done. Also, we presented specific evaluations of research requirements in response to what we view as

the more significant of the future issues. We anticipate discussing our 2000 report with the Commission in the future.

Question 8. Low-Power and Shutdown Operations Risk

I appreciate the importance of sound reactivity management. Please provide ACRS' perspective of the actual risk significance of the low power reactivity events referred to in this bullet.

Response:

Our position is that the staff does not have an adequate understanding of risk during low-power and shutdown operations. Some analyses of shutdown risks exist. In this country, the results have focused on events such as loss of inventory or loss of heat sink. Reactivity events have not been found to make important contributions to the predicted risk in these studies. This is largely because the modes of operation where reactivity events are likely have been "screened out" as making small contributions to risk. (This screening may itself deserve some further examination in light of the Wolf Creek draindown event, which took place during an operational mode deemed by NRC research to be a small contributor to risk). Reactivity events, especially boron dilution events, have received much more attention in Europe. The shutdown risk assessment for Great Britain's Sizewell plant, which is regarded as a modern pressurized water reactor (PWR), gave significant attention to reactivity events that have root causes of human error. Sizewell attributed about 20% of the risk of low-power and shutdown operations to events involving reactivity excursions - notably boron dilution events. In light of the reactivity control events that have occurred in recent years at U.S. nuclear power plants, it is important that these events be included in any comprehensive study of risk during low-power and shutdown operations. On the other hand, the shutdown risk analysis done for AP600, an advanced PWR, found little risk importance for reactivity events. Clearly, reactivity events constitute just one more area where understanding of low-power and shutdown risks is developed inadequately.

Question 9. Low-power and Shutdown Operations Risk

In Dr. Powers' opening remarks (slide 7), he indicates that the industry is paying close attention to shutdown operations. He also indicates that the NRC may be 'outgunned' by licensees in the risk analysis of shutdown operations. That leads me to believe that in ACRS' view, licensees have a good understanding of the risk associated with shutdown operations. In contrast, this bullet implies that the NRC and/or licensees must do more in this area to ensure safety. Since we did not have an opportunity to discuss this topic, I am uncertain as to ACRS' position. Please provide me ACRS' bottom line on shutdown operations risk and the basis for this bottom line.

February 17, 2000

Response:

We have been addressed by licensees concerning low-power and shutdown risk. We have visited plants and have seen the analytic tools that are being used by licensees to better plan shutdown operations. We have asked line organizations of NRC repeatedly what analytic tools they have to independently assess risk during low-power and shutdown operations. Line organizations have no such tools. We have asked line organizations if they have an understanding of licensee tools sufficient to independently interpret the results obtained with these tools. Repeatedly, individuals have responded that they are not sufficiently familiar with the analytic tools used by the licensees. This is certainly what one could call being "outgunned" technically. One would be more confident in the ability of the agency to ensure adequate protection of the public health and safety during shutdown operations, if the confidence of the NRC management were bolstered with actual examples of analytic tools that are not a decade out of date.

Other signs of improving technologies being made available to licensees have been detected, especially during our visits to the plants. Indeed, one would expect that if licensees do see sufficient economic benefits from risk-informed regulations, they will be ever more willing to invest in the development of superior analytic tools and willing to acquire superior analysis talent. We see at the NRC little evidence that there is a well-developed understanding of the analytic support that a risk-informed regulatory system will require. To the contrary, we see a willingness to maintain the current analytic capabilities and to rely on "an integrated decisionmaking process" to address regulatory issues. We are concerned that this process will have the appearance, at least, of a return to more subjective and less predictable regulation, especially if licensees are presenting results of state-of-the-art analytic tools unfamiliar to line organizations at NRC. At least one member of the ACRS feels that it is time for a thorough redevelopment of the agency's PRA Implementation Plan.

References

1. Memorandum dated December 22, 1999, from Chairman Meserve, NRC, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Follow up Questions to the Advisory Committee on Reactor Safeguards Meeting with the Commission, November 4, 1999.
2. ACRS report dated April 18, 1997, from R. L. Seale, Chairman ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Establishing a Benchmark on Risk During Low-Power and Shutdown Operations.
3. U. S. Nuclear Regulatory Commission, NUREG/CR-4674, Vol. 21, "Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report," December 1995.
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6143, Vol. 1, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," July 1995.

5. U. S. Nuclear Regulatory Commission, NUREG/CR-6144, Vol. 1, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," October 1995.
6. ACRS report dated May 11, 1999, from D. A. Powers, Chairman, NRC, to Shirley Ann Jackson, Chairman, NRC, Subject: Modified Proposed Final Revision to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
7. U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, proposed Revision 3 to Regulatory Guide 1.160 (DG 1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," November 17, 1999.
8. Final Draft of Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated October 8, 1999.
9. U. S. Nuclear Regulatory Commission, NUREG-1150, Vol. 1, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990.
10. U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
11. ACRS letter dated May 11, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Elevation of CDF to a Fundamental Safety Goal and Possible Revision of the Commission's Safety Goal Policy Statement.
12. U. S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, NUREG-1635, Vol. 1, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission," June 1998.
13. U. S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, NUREG-1635, Vol. 2, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission," June 1999.