NUREG-1251
Vol. II

Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States

Final Report

Appendix - Public Comments and Their Disposition

U.S. Nuclear Regulatory Commission

UNITED STATES NUCLEAR REGULATORY COMMISSION

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CONTENTS

I. INTRODUCTION

A draft of NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," was issued for public comment in September 1987. A notice of the issuance of the report was published in the Federal Register (52 FR 33304-5) on September 2, 1987. The comments received, together with further work by the NRC staff, were taken into account in preparing the final version of the report. This appendix contains the comments received and the NRC staff's response to significant issues raised in the comments, including the nature and basis of the resulting changes to the draft report.

Federal Emergency Management Agency staff reviewed a draft of Chapter 4, "Emergency Planning," and contributed comments and suggestions based on that review. These were taken into account by the NRC staff.

II. LIST OF COMMENTERS

The commenters and the abbreviations by which they are referred to in this appendix are:

ANBEX	ANBEX
James Asselstine*	JA
Battelle Pacific Northwest Laboratories	PNL
Combustion Engineering, Inc.	CE
Illinois Department of Nuclear Safety	IDNS
Lawrence Livermore National Laboratory	LLNL
New York Power Authority	NYPA
North Carolina Department of Crime Control and Public Safety, Division of Emergency Management	NC
Nuclear Information and Resource Service	NIRS
Nuclear Management and Resources Council	NUMARC
Ohio Citizens for Responsible Energy, Inc.	OCRE
Sandia National Laboratories (summary comments--October 23, 1987; detailed comments--November 16, 1987)	SNL
Westinghouse Electric Corporation	W
Yankee Atomic Electric Company	YA

* Mr. Asselstine, a former NRC Commissioner, commented on an earlier draft while he was still a member of the Commission.

III. COMMENT HIGHLIGHTS AND NRC STAFF RESPONSE

This section recapitulates the highlights of the comments received, often in a very condensed form, for the convenience of the reader; it also provides the NRC staff's responses to the comments. Some comments with a similar thrust or related subject matter have been combined. For the comments in their entirety, see Section IV.

The comments and staff responses are organized according to subject matter and are presented in the order in which the subjects are addressed in the main report (NUREG-1251, Vol. I). The chapters involved are identified, as are specific sections where appropriate. The abbreviated name of the commenter is provided in parentheses.

SUMMARY

Comment: The report should state more explicitly that U.S. high standards of safe design, operational controls, management, and staff motivation and dedication contrast sharply with serious weaknesses in Chernobyl design and operation. (NUMARC)

Violation of regulations is much less likely in the United States than in the Soviet Union; the Summary should echo this conclusion. (CE)

Response: The staff recognizes the strengths of U.S. design and operational requirements and practices that provide protection against major accidents and the aspects of U.S. practices that would prevent or mitigate features of an accident similar to the one at Chernobyl. It also recognizes that, even in the absence of specific analogies, the events at Chernobyl suggest that particular issues be examined further, as part of a required persistent vigilance. The Summary is intended to convey this general perspective.

Comment: The report does not substantiate the conclusion that some aspects of emergency planning need to be reexamined. (NC)

Severe-accident studies should not be expanded. (NYPA)

Response: The Chernobyl experience concerning the release and dispersion of radioactive material, the effects related to ingestion pathways, decontamination methods and their effectiveness, and the relocation of nearby populations should provide--notwithstanding technical, socioeconomic, and cultural differences--a valuable extension of the data base for U.S. emergency preparedness. These considerations justify, in the staff's judgment, the indicated followup efforts in these areas.

Comment: The conclusion that no immediate changes are needed is wrong, in view of serious inadequacies in NRC's Severe Accident Policy. Emphasis on cost-risk tradeoffs to eliminate or forestall requirements of marginal importance to risk is particularly contrary to the care and vigilance sought. (OCRE)

Response: The staff has carefully examined particular issues with a nexus to Chernobyl and has assessed the nature, extent, and limits of their bearing on U.S. commercial reactor safety regulations. The general conclusions reflect these assessments. The staff has not considered it necessary or appropriate to expand the scope of this assessment to include a more general reexamination or rejustification of U.S. reactor safety regulatory practices.

Comment: Though the design and operation of the RBMK reactor are vastly different from U.S. practices, there are many areas beyond those identified in the report where improvements could be made. (IDNS)

The Chernobyl accident provided important information on radionuclide transport, ingestion pathways, etc. Some approximations in accident analyses merit reexamination. (SNL)

Response: Specific suggestions by commenters concerning the need for further study of particular areas have been taken into account. The followup studies indicated for radionuclide-release source terms and for ingestion pathway, decontamination, etc. (noted in the Summary and discussed in Chapters 5 and 4, respectively) are relevant to the specific issues noted in the above comments.

Comment: The report takes an overly narrow view of the lessons of the accident. The report should address such questions as the likelihood of major releases occurring at a U.S. plant over the next 50 years; whether such likelihood is acceptable; and, if not, what measures can be backfitted to reduce those risks. Also, we should learn from measures being adopted in European countries. (JA)

Response: General risk assessments and safety policies are outside the scope of this report. They are addressed elsewhere (draft NUREG-1150, the Safety Goal Policy, the Severe Accident Policy, the Backfit Rule, etc.) This report is devoted and limited to the assessment of issues with a reasonably apparent nexus to factors that played a role in causing the Chernobyl accident and the course and consequences of the accident.

Exchange of information on filtered venting of containments on an international level is discussed in the Summary and in Section 3.2. The NRC has an active program in regard to the international exchange of information on reactor safety measures. Under this program, the NRC hopes to obtain information that will be beneficial in areas that have some nexus to Chernobyl as well as in other areas.

CHAPTER 1--ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES

Because most of the commenters provided some comments on Chapter 1, the staff has consolidated and summarized the comments that pertain to a specific issue. However, if the comment is from one commenter, the abbreviated name of that commenter is provided in parentheses.

Comments relating to this chapter, listed by the primarily pertinent section, were received from the following commenters. (See the list in Section II for the complete names of the commenters.)

Section	Commenters
1.1	IDNS, LLNL, NIRS, OCRE, PNL, SNL, W
1.2	OCRE, NUMARC, NYPA, PNL
1.3	NUMARC, PNL, SNL, W
1.4	NUMARC, W, YA
1.5	IDNS, NUMARC, NYPA, OCRE, SNL
1.6	CE, IDNS, LLNL, NUMARC, NYPA, OCRE, PNL
1.7	SNL

1.1 Administrative Controls To Ensure That Procedures Are Followed and That Procedures Are Adequate

Comment: Chernobyl's message is that administrative controls can be violated and that administrative controls cannot compensate for basic design weakness. The solution is to make reactor designs inherently safe and foolproof through automated controls that cannot be easily defeated and that minimize the consequences of human error and to reduce the likelihood that administrative controls will be violated.

Response: The concerns expressed in this comment are largely addressed in this chapter. The staff is supporting current research programs leading to improved controls and control rooms (e.g., improved control room instrumentation, control room design standard), which have as their objective the minimization of human error. The staff also recommends increased emphasis on inspections to ensure that administrative controls are in place and are being followed. (See revised Section 1.1.4.)

In regard to the possible violation of procedures and administrative controls, the assessment and conclusions (Sections 1.1.3 and 1.1.4) have been changed to recognize the value of a study of the frequency, nature, and severity of violations and to recommend that such a study be undertaken. Such a study could provide a firmer basis for either a reassuring conclusion or for consideration of additional means of reducing inadvertent violations or deterring willful ones.

Comment: The report should stress the need for ensuring that all safety regulations are adequately implemented and adhered to. The draft report does not recognize that this is not always the case. It also fails to note that hazardous plant configurations can be avoided by hardware modifications as well as by administrative guidelines.

Response: The report does stress the need for ensuring that all safety regulations are adequately implemented and adhered to. These regulations are implemented by administrative controls at each nuclear power plant. In Section 1.1 of the report, the staff reviews these administrative controls for adequacy. Similarly, although the staff agrees that hazardous plant configurations can be avoided by hardware modifications, the focus of this section is on the issue of administrative controls. This certainly does not deny the appropriateness of solutions in regard to hardware control when they are the best choice for correcting a safety problem.

Comment: The draft report does not address the fact that regulations and procedures are not self-enforcing, and thus that additional measures are required. The NRC should propose these additional measures in the final report.

Response: Section 1.1 addresses administrative controls to ensure that procedures are followed and that they are adequate. Thus, it is to these administrative controls that the NRC looks to ensure that procedures are followed. The NRC also maintains an effective enforcement capability, for the most part centered in the regional offices and onsite resident inspectors. In addition, the headquarters Office of Enforcement is responsible for enforcement actions. (See item (5) in Section 1.1.1.) The need for additional measures (to ensure that regulations and procedures are followed) is not currently evident. However, this judgment may be reexamined after the study on violations of procedures recommended in revised Section 1.1.4 is completed.

1.1.1 Current Regulatory Practice

(1) NRC Requirements and Guidance for Procedure Development and Use

Comment: Current regulatory practice in regard to the development and use of procedures needs to be strengthened. Current procedures do not take into consideration human factors. Better control of procedures is needed, both of updated procedures and temporary procedures.

Response: The staff has addressed emergency operating procedures and reviewed and approved guidelines on utility procedures. The staff's review of plant-specific emergency operating procedures prepared in accordance with these guidelines indicates that some improvements are needed. Under its ongoing program, the NRC's Office of Nuclear Regulatory Research staff reviews other procedures to determine if safety-significant problems exist. If problems are evident, the staff recommends necessary improvements to the procedures.

(4) Training on Procedures

Comment: Even advanced simulators have limited capability to simulate severe accidents. An effort should be made to incorporate current knowledge of severe accidents into the capabilities of simulators.

Response: The NRC staff is continuing its research on responses to severe accidents. Research programs addressing the issue of simulator studies of severe accidents have been proposed and may be implemented in the future.

1.1.2 Work in Progress

(1) Technical Specifications Improvements

Comment: Priority should be given to the review of Technical Specifications to ensure that they are clearly written, their content is limited to safety-related requirements, and they are closely related to the reactor operator's job. The

NRC should take whatever steps are necessary to expedite completion of the Technical Specification Improvement Program.

Response: The staff's effort to improve Technical Specifications through the Technical Specification Improvement Program is a priority program. (See item (1) in Section 1.1.2.) The attributes described in the comment will be addressed in the program.

(2) Symptom/Function-Based Emergency Operating Procedures

Comment: Reviews of new symptom-based emergency operating procedures (EOPs) have revealed a wide variation in the level of human factors principles applied. Additional NRC guidance may be needed to improve these EOPs. Some EOPs do not require operator knowledge of whether or not the plant is initially operating within the plant's safe operating envelope.

Response: The staff is aware of deficiencies in the new symptom-based EOPs. This is noted in Section 1.1.3. This section also reiterates the staff belief that the concept of maintaining plant conditions within the safe operating envelope should be emphasized in operator training. This is important regardless of the nature of the symptom-based EOPs, and will enhance operator ability to implement EOPs, regardless of how they are written.

1.1.4 Conclusions and Recommendations

Comment: Implementation of symptom-based EOPs needs to be linked to better diagnostic tools to ensure that the correct EOP is selected. Better operator tools are needed (e.g., computer operator aids) as well as the new symptom-based EOPs.

Response: There is no question that additional operator aids could assist in the rapid diagnosis of abnormal operating conditions, and the staff supports continuing research in this area. For the purposes of this section (i.e., adequate procedures that will be followed by operators), the staff stresses the importance of symptom-based EOPs and of operator training in these procedures.

Comment: There is no guarantee that procedures will always be followed, even with administrative controls.

Response: It is possible that the conclusion drawn in this section of the draft report implied a relationship between symptom-based procedures and administrative controls that was inappropriately characterized. Section 1.1.4 has been revised to ensure that this implication no longer exists.

1.2 Approval of Tests and Other Unusual Operations

Comment: The NRC should undertake prior review and approval of all 10 CFR 50.59 changes, tests, and experiments.

Response: 10 CFR 50.59 establishes which changes, tests, and experiments may be done solely under a licensee's administrative procedures and which must get prior NRC approval. At this time, new criteria and guidelines for 10 CFR 50.59

reviews are being developed, as noted in revised Section 1.2.2. These guidelines should reduce even further the small number of changes, tests, and experiments that are incorrectly conducted without prior NRC review and approval. It is impractical for the NRC to undertake prior review and approval of all 10 CFR 50.59 actions.

1.2.2 Work in Progress

Comment: The Atomic Industrial Forum is now part of the U.S. Council for Energy Awareness. Criteria and guidelines for licensees conducting 10 CFR 50.59 reviews are now being developed in a joint effort by the Nuclear Management and Resources Council and Electric Power Research Institute's Nuclear Safety Analysis Center. In addition, the final industry guidelines for the development of adequate and consistent 10 CFR 50.59 implementation programs should be available by April 1988.

Response: Section 1.2.2 has been revised to reflect this information.

Comment: The industry effort to develop 10 CFR 50.59 review criteria and guidelines is strongly endorsed. There is no need to revise 10 CFR 50.59.

Response: The NRC has no plans to revise 10 CFR 50.59 at this time. See revised Section 1.2.2.

1.2.4 Conclusions and Recommendations

Comment: If the joint Nuclear Safety Analysis Center/Atomic Industrial Forum efforts to prepare guidelines for 10 CFR 50.59 reviews still result in instances of inconsistent documentation or too narrow a determination of the unreviewed safety question, the NRC should provide additional guidance to industry in this area.

Response: Additional guidance will be prepared, if necessary, on the basis of the NRC review of the industry guidelines as revised in the light of NRC's comments. (See revised Section 1.2.2.)

Comment: The last sentence of this section, "...whether current NRC testing requirements...appropriately balance risks and benefits." is important and should be emphasized.

Response: The staff agrees with this comment. The staff is actively studying the risks and benefits of testing requirements with a view to considering changes to enhance net safety.

1.3 Bypassing Safety Systems

Comment: The definitions of three terms, "bypass," "defeat," and "out-of-service," are suggested in lieu of the terminology used in the report. (W)

Response: The terms "operating bypass" and "maintenance bypass" are acceptably defined in Institute of Electrical and Electronics Engineers Standard 603-1980 and are endorsed by Regulatory Guide 1.153. The suggested definition of "defeat" refers to an extraordinary "bypass" under accident recovery conditions;

the term in this sense is not generally accepted or familiar. Retention of the terms used in the draft report is warranted because their definitions are well established as well as useful.

1.3.1 Current Regulatory Practice

(1) Technical Specification Restrictions on the Use of Bypasses

Comment: Design-basis events should not be referred to as "bounding" or "worst case" unless these terms are more clearly defined. Design-basis analyses make assumptions concerning operability (e.g., availability of electrical power) that may not be "bounding" or "worst case" from a risk point of view. (SNL)

Response: The terminology was changed (by inserting "design-basis" between "most severe" and "transient or accident" in the fourth sentence of item (1) of Section 1.3.1) to limit the context clearly to design-basis events.

1.3.4 Conclusions and Recommendations

Comment: Maintenance is important in ensuring plant safety; more resources should be devoted to improving NRC's ability to evaluate maintenance performance. (PNL)

Response: The commenter does not recommend any changes to the report. Further, the comment addresses the adequacy of NRC's research in the field of maintenance and does not bear directly on the implications of the Chernobyl accident. No changes were made.

1.4 Availability of Engineered Safety Features

Comment: The issues described in this section appear to be a subset of those described in Section 1.3.

Response: Section 1.3 addresses administrative controls intended to ensure that systems needed are not going to be bypassed. Section 1.4 addresses requirements concerning the availability of engineered safety features, notably as reflected in the Technical Specifications. For example, Section 1.4 examines the question of whether the Technical Specifications adequately cover the availability of engineered safety features during plant operation in low-power and shutdown modes. The issues covered by Sections 1.3 and 1.4 are related.

1.4.3 Assessment

Comment: The report fails to mention that under the plant conditions identified for which the Technical Specification requirements may not be consistent with the safety analysis, a significantly longer time period exists for operator action to restore the safety function before any serious consequences to the plant occur. Thus, these inconsistencies do not represent a serious threat to safety because the operators generally are able to identify such events and take appropriate manual actions to maintain the plant in a safe condition.

Response: The staff agrees. A paragraph to reflect this consideration has been added to Section 1.4.3.

Comment: The wording of question (1) could imply that future Technical Specifications would not allow loops, legs, or channels of redundant safety subsystems to be disabled or removed from service or to undergo maintenance while the plant is at power.

Response: This is not the intent of question (1). The question has been modified so that it refers to an entire engineered safety feature.

1.4.4 Conclusions and Recommendations

Comment: The recommendation that licensees perform comprehensive reviews of their specific design and Technical Specifications for each mode of operation is unnecessary.

Response: The staff believes that this review is both necessary and important, since it is needed to provide an adequate basis for the Technical Specifications' "Bases" section, which industry is committed to revise. Section 1.4.4 has been changed to reflect this view.

1.5 Operating Staff Attitudes Toward Safety

1.5.1 Current Regulatory Practice

Comment: The report states that maximum working hours have been specified to ensure that rested, qualified operators are available. Yet in reality, operators have indicated that in some plants working double shifts is common, specifically to maintain the staffing levels required by NRC regulation.

Response: Most utility Technical Specifications specify maximum working hours. If this is the case, the licensee cannot modify these limits on maximum working hours. The staff is implementing a Technical Specification Improvement Program. (See item (1) of Section 1.1.2.) The possibility of including maximum working hours in all Technical Specifications will be addressed. Section 1.5.1 has been revised so as not to imply that maximum working hours are currently specified in NRC regulations.

Comment: Plant personnel need to have a realization that nuclear power is an inherently dangerous technology and that safety is of the utmost importance. This realization (attitude) should be vigorously enforced by the NRC. Enhanced NRC monitoring of licensee operations is necessary (to ensure that plant personnel and management act together with the utmost concern for safety). The NRC should establish specific criteria for emergency operating procedures (EOPs) and operator training and set an effective date (soon) by which acceptable EOPs and effective training are to be in place.

Response: As stated in Section 1.5.1 of the draft report, regulations do not directly address operator attitudes or sense of vigilance, nor does it appear practical to set up a system designed to enforce an attitude. The staff believes that operators are aware that safety is of utmost importance. Furthermore, this is emphasized in operator training programs. Additional NRC monitoring to ensure management and operator actions/concerns regarding safety does not appear to represent an effective means of achieving this objective. At this time, symptom-based EOPs are in place at all utilities, and operators have been trained in these procedures. The staff has a program in place that addresses increased emphasis on symptom-based procedures. Under this program, the staff may consider additional criteria for these procedures and any related operator training.

1.5.2 Work in Progress

Comment: The impact of NRC's program for the requalification of licensed operators as implemented by the staff is more negative than positive. If the requalification examinations do not measure the ability to operate the reactor safely, the purpose of the licensing process has been subverted.

Response: A modified requalification evaluation process for licensed operators and senior operators has been prepared in response to comments on the initial process and is being pilot-tested. Implementation of the modified process is expected to begin in late summer of 1988. Section 1.5.2 has been revised to provide this information. The staff fully intends that the requalification process will continue to measure the ability of operators to operate the reactor safely.

Comment: The discussion of the requalification evaluation process in the second paragraph of Section 1.5.2 should include a discussion of how the proposed rule change would affect the process.

Response: This paragraph has been modified and is now in accordance with recently revised 10 CFR 55.

Comment: The requirement that senior operators hold a degree would not increase operating and management expertise on shift. The degree requirement may impede the career advancement of highly qualified and experienced individuals and thus negatively affect safety.

Response: The issue of degree requirements for senior operators is being considered elsewhere within the NRC and need not be addressed in this report. However, the staff does agree that consideration of a rule requiring senior operator candidates to hold a degree in engineering or a related science does not in itself promote an increased operating staff awareness of safety. The reference has been deleted from Section 1.5.2.

1.5.3 Assessment

Comment: Some human factors experts have taken exception to frequent shift rotations and other utility working practices.

Response: There will always be some disagreement on matters such as shift rotations or shift length among the human factors community. The staff believes that most human factors experts would agree with the staff's position as given in this section. A research program that is in the planning stage will provide additional information on the effect of shift length on operator performance.

1.5.4 Conclusions and Recommendations

Comment: An important conclusion is that operators need thorough training in the bases for safety features/limits and in basic reactor safety. This needs more emphasis in the report.

Response: Such a conclusion has been added to Section 1.5.4.

1.6 Management Systems

1.6.2 Work in Progress

Comment: The NRC should reestablish work to provide a technical basis for assessing management and operator performance.

Response: Work in this area has not been terminated. Current efforts (as noted in Section 1.6.2) include the development of licensee performance indicators and improvements in the systematic assessment of licensee performance program.

Comment: It appears that the NRC will attempt to treat the symptoms of poor management rather than the direct cause without first defining unsatisfactory management. A measurable definition of management safety goals and specific fundamental management requirements and tasks would allow for a more precise assessment of management systems.

Response: The effect of supervisory and management practices on operator performance in regard to safety is the subject of a current NRC research program.

1.6.3 Assessment

Comment: The designation of an onsite nuclear-safety manager to be placed in charge of safety is unnecessary and would not be expected to improve plant safety. All line organizations and support organizations (site safety committee, the shift technical advisor, the technical support center, etc.) are responsible for plant safety. Furthermore, the NRC resident inspector provides an independent onsite safety review function. What is important is the need to instill professionalism and safety consciousness in shift supervisors and all classes of operators.

Response: The staff agrees with this comment. The assessment has been changed, and the recommendation that the establishment of a position for an onsite nuclear-safety manager be considered has been removed.

Comment: The assignment of a high-level, onsite nuclear-safety manager having no other responsibilities has merit. Such an individual should be licensed as a senior reactor operator and should concentrate only on matters of substantive safety significance.

Response: Most commenters disagreed with the recommendation that a position for a dedicated, high-level, onsite nuclear-safety manager be established, arguing that such a position would not be expected to result in increased plant safety and could be expected to result in a decreased emphasis on individual responsibility for safety. The staff has accepted these comments and has revised Sections 1.6.3 and 1.6.4 accordingly.

1.7 Accident Management

Comment: The NRC and industry should work together to develop explicit procedures for dealing with accident scenarios that develop slowly, for example, loss of containment heat removal capability in boiling-water reactors (BWRs). (SNL)

Response: The staff agrees. At this time, for example, the response of BWR Mark I and II containments to severe accidents is receiving special attention through a special BWR task force effort.

1.7.2 Work in Progress

Comment: This section indicates the need for reliable instrumentation and control equipment that will be used in responding to severe accidents. Although the section implies a need to ensure the reliability of equipment, no mention was made of a need for appropriate qualification procedures for the severe-accident case. (SNL)

Response: The staff did not intend to prejudge that there was a need for additional (i.e., severe-accident-related) equipment reliability. It may not be possible to qualify instruments for all accident conditions or to engineer around a loss of instrumentation. Accommodation for this potential condition may need to be made in accident-management planning.

Comment: This section implies that the NRC/Industry Degraded Core Rulemaking program examination of four reference plants covered essentially all accident sequences and recommended accident-management actions. This is not the case. (SNL)

Response: This portion of Section 1.7.2 has been rewritten so as not to make this implication and to indicate that significant additional effort is needed to implement the Commission's Severe Accident Policy.

1.7.4 Conclusions and Recommendations

Comment: The statement that "there is no need to increase or alter the scope of the ongoing programs" appears to contradict other statements regarding the need to improve degraded core accident responses. (SNL)

Response: The staff did not intend to make this implication. This section of the report has been rewritten and clarifies the staff position on this point.

CHAPTER 2--DESIGN

2.1 Reactivity Accidents

Some of the comments addressed here pertain to Section 2.2, "Accidents at Low Power and at Zero Power," as well as to Section 2.1.

Comment: The Nuclear Management and Resources Council believes that the focus of NRC's review of this subject should be on accidents initiating at low power. Thus, Sections 2.1 and 2.2 could be consolidated and focused on low-power tests or transients that could cause reactivity excursions. (NUMARC)

Response: Reactivity transients initiating at low power will be addressed in the staff's study; however, there is no need to rewrite Sections 2.1 and 2.2.

Comment: Sections 2.1 and 2.2 of the draft report discuss reactivity accidents and accidents at low power and at zero power. The Nuclear Management and

Resources Council (NUMARC) will provide the overall coordination of the generic aspects of this issue, and work with the four vendors, the vendor owners groups, and the Electric Power Research Institute. Results from this work will be integrated into the Technical Specification Improvement Program (TSIP) as appropriate. NUMARC is working very closely with the four owners groups on Technical Specification improvements. Each group has committed to implement major programs to develop topical reports that revise the present Standard Technical Specifications. In addition, application of the improvements in four plant-specific Technical Specifications to operating plants is going forward in parallel. These efforts include developing improved bases and utilizing human factor considerations to make them more operator friendly. (NUMARC)

Response: The staff welcomes the NUMARC efforts on the Technical Specification Improvement Program. In Sections 2.1.4 and 2.2.4, the staff has added sentences stating that these efforts will be coordinated with the severe accident program and results will be made available to the industry to help develop improvements in the Technical Specifications if they are needed.

Comment: The description of several of the postulated accident scenarios should be refined or eliminated because they are ill defined and difficult to understand and, in many cases, the connection to the Chernobyl accident is vague at best. (CE)

The Industry Degraded Core Rulemaking program and related NRC programs extensively evaluated dominant severe-accident initiators. If the concerns raised in the draft report have already been adequately addressed through these programs or have been found to be of sufficiently low probability, they should be dropped from further consideration. (CE)

Response: The staff believes that all important relationships between postulated events and the Chernobyl accident have been adequately covered in Section 2.1.1. A major rewrite of Sections 2.1 and 2.2 is not necessary. In Section 2.1.3, the staff changed the boiling-water-reactor transient "multiple safety-relief valve failure to operate" to read "overpressurization with limited relief" in order to describe the event better. It also added sentences in Sections 2.1.4 and 2.2.4 to emphasize that the work in regard to reactivity accidents will be coordinated with the severe accident program and results will be available to the industry so that improvements in Technical Specifications can be made if needed.

Comment: Reactivity changes do occur with temperature changes; however, the central issue is the injection of cold water. (PNL)

Response: This is addressed under the list of transients to be analyzed.

Comment: In the third paragraph in Section 2.1, the sentence, "A critical reactor generally...." makes little sense; it may be assumed to imply that overmoderation is required to maintain criticality in a light-water reactor at low power and high burnup. The next sentence, "In a PWR, boron...." states that a decrease in boron density results in a decrease in the possibility of a positive moderator coefficient. Clarification of this sentence would help.

The stated conditions where a positive moderator void coefficient may be present, for both pressurized-water reactors and boiling-water reactors, includes minimally inserted control rods. This condition was considered to be a contributing factor to the Chernobyl accident, as this is the least effective location for response. If this superior response time and control rod worth of the light-water reactors discussed in Section 2.1 are sufficient to override this concern, it could be explicitly stated at this point.

The phrase "normal...conditions" is used several times in Section 2.1 (pages 2-2 and 2-3 of the draft report) in assertions about the potential for large reactivity insertion events in light-water reactors. Unless the word "normal" is defined to include not only conditions that are expected in everday operation, but also those that may occur with some reasonable probability, these assertions are not relevant when assessing the possibility of a Chernobyl-type event in a light-water reactor.

There is no discussion in this section (although there is mention of it in Chapter 5) on the impact of the spatially skewed core power distribution on the reactivity state of the Chernobyl reactor. As that has been assessed as having significant impact on the initiation and development of the accident, there should be some mention of U.S. reactor experience with and analysis and implications of such highly skewed power distributions. (LLNL)

Response: The sentence, "A critical reactor generally...." is correct. The preceding sentence describes conditions under which overmoderation can be approached in boiling-water reactors. The next sentence, "In a PWR, boron...." has been corrected. The staff has added a sentence to the last paragraph of Section 2.1 to define "normal conditions." In regard to reactor safety, "normal conditions" refer to those conditions that exist while the reactor is operating within Technical Specification limits. The staff also has added a paragraph in Section 2.1 addressing the "positive scram" case. The power distributions used in the proposed reactivity studies will be those appropriate to U.S. light-water reactors.

2.1.1 Current Regulatory Practice

Comment: In the United States, the general design criteria (GDC) for light-water reactors in 10 CFR 50, Appendix A, provide the overall regulatory requirements for reactivity control. In particular, GDC 11 requires that light-water reactors have self-limiting power feedback behavior, without any protection system or operator action. This self-limiting power feedback is one major difference between the RBMK reactor and the light-water reactors in the United States. The RBMK reactor is in its most reactive configuration and therefore has the highest potential for power generation with a complete lack of coolant. This is in contrast to a light-water reactor, which is least reactive in this condition. Therefore, with regard to coolant-induced reactivity excursions, U.S. light-water reactors bear no comparison to the RBMK reactor. The documentation of the consequences of reactivity insertion accidents for Westinghouse pressurized-water reactors already exists in various plant Final Safety Analysis Reports; these consequences are within the acceptable limits specified in NUREG-0800. (W)

Response: The staff believes that the discussion in Section 2.1.1 is adequate. The staff agrees that the consequences of reactivity transients with single-failure assumption meet the limits specified in NUREG-0800. What the staff intends to do as an outgrowth of the assessments in the report is to extend the analyses to cover multiple failures at different operational modes and estimate the probability of occurrence. No change is necessary.

Comment: The NRC staff concluded that pressurized-water reactors generally have a negative temperature and void coefficient in the normal operation range. In light of the severe consequences of the Chernobyl accident, perhaps a 3-D [three-dimensional] analysis of the total fuel cycle with special attention to local perturbations is appropriate.

The RBMK reactors were to be altered after the accident to limit control rod withdrawal and to increase the number of control rods. Although these are RBMK-specific modifications, the U.S. pressurized-water reactors have two independent safety shutdown systems (control rods and boron); however, it is understood that the control rods cannot control the reactor power level in the event all the boron is lost. It seems that increasing the number of control rods in pressurized-water reactors warrants additional consideration by the NRC. The impact on plant design, efficiency, and cost should be evaluated.

Pump cavitation and loss of flow, particularly at lower power, was a known concern of the Soviets and could have contributed to the accident. The NRC may want to evaluate more fully the potential effects pump cavitation may have on commercial U.S. reactors, particularly boiling-water reactors during low power or accidents/transients. (PNL)

Response: The staff has added a sentence in Section 2.1.1 to the effect that three-dimensional analyses of the core including moderator coefficients over the whole range of fuel cycles are commonly made and conservative moderator temperature coefficients are selected. The conservatism is intended to cover local perturbations. In regard to the concerns about U.S. pressurized-water-reactor shutdown systems, transients with cooldown and deboration will be studied. This should indicate if any problem areas exist. As for the concern about pump cavitation or loss of flow at a lower power in U.S. boiling-water reactors, the staff points out that no significant positive void coefficients exist at low temperatures.

Comment: In Section 2.1.1, under item (2), "Moderator Boron Dilution in a Pressurized-Water Reactor," the statement that "this event, where it may be pertinent, is being reviewed on a case-by-case basis" appears to contradict the statement in Section 2.1.2 that "no work is currently being done on any events considered for this issue." (SNL)

Response: Section 2.1.2 has been revised so that it does not contradict the statement in Section 2.1.1.

2.1.3 Assessment

Comment: One may always postulate more severe consequences for reactivity insertion events by assuming combinations of failures of lower and lower probability. By ensuring that reactivity insertion events with consequences significantly more severe than those within the design basis have a probability of occurrence that is significantly less than of those that define the design basis, we establish an adequate safety margin for operation of light-water reactors. (W)

Response: The third paragraph of Section 2.1.1 addresses the issue of the low probability of occurrence of reactivity events with multiple failures. Since the staff cannot determine a priori if consequences will be well beyond the current design-basis envelope or not, it is planning to examine the events listed in Section 2.1.3. In Section 2.1.4, the staff has added a sentence to the effect that the judgments used to identify accident sequences will be reviewed in order to confirm the appropriateness of previous decisions by using probabilistic methods.

Comment: In assessing the differences in reactivity feedback behavior between the RBMK reactor and the light-water reactor, the draft report does not point out a very fundamental characteristic of the light-water-reactor design. Although local reactivity coefficients may be positive within a limited range of conditions, the overall integral power feedback is always strongly negative, including that under postulated accident conditions. The reactivity coefficient approach, although adequate for conservative analyses, does not give a complete picture of the reactivity state of the core. (W)

Response: In Section 2.1.3, under item (3), "Positive Moderator Reactivity Coefficient in a Pressurized-Water Reactor," the staff has added an explanatory statement that in most cases positive moderator void effects will be compensated by fuel heatup effects.

Comment: The staff recommends that the selection of reactivity events for analysis and the actual analyses done in the past be reevaluated as a result of the Chernobyl accident.

Although recognizing that it is always prudent to reconfirm the validity of past analyses, the commenter considers the issues raised in Section 2.1 to be of very low priority. Basic design differences between Chernobyl and U.S. plants, including the absence of positive void coefficients and the presence of fast-acting control rods at U.S. plants, ensure that the superprompt critical reactivity excursion that occurred at Chernobyl will not occur at U.S. plants. Moreover, a large number of reactivity insertion events are already considered for U.S. plants. (YA)

Response: In Section 2.1.3, in the paragraph before the last, the staff has added a sentence stating that one of the purposes of the analyses in Sections 2.1 and 2.2 is to determine the probability of these events so that priorities within NRC can be arranged.

Comment: In Section 2.1.3, a number of accidents are identified as candidates for further study. The commenter believes that sufficient safeguards have been incorporated in the plant design, Technical Specification limits, etc., to preclude the occurrence of serious consequences from these events. The commenter illustrates this view by discussing some events listed for pressurized-water reactors, that is, multiple rod ejection, unlimited boron dilution, opening of loop stop valves in a loop containing unborated water, and loss-of-coolant accident or other injection with unborated water. (W)

Response: Technical Specifications may be violated. The probability of violations is low but violations are possible. The recommended study will examine the increased risk from such violations to determine whether the conclusions of the commenter can be confirmed.

Comment: The Electric Power Research Institute (EPRI) and other industry bodies have performed work in the area of reactivity accidents, low-power operation, and positive moderator reactivity coefficients. There also have been interactive meetings between EPRI and vendors concerning this area. To date the results reported support the statement in the second paragraph of Chapter 2 that "the nuclear design of U.S. reactors,...provides assurance against a Chernobyl-type superprompt critical reactivity excursion." Most of the accidents are listed in Final Safety Analysis Reports and/or emergency operating procedures.

For those sequences beyond the design basis that have not undergone extensive reactivity analysis, probabilistic analysis should validate the credibility of the sequence before detailed reactivity analysis. (NUMARC)

Response: Probabilistic screening will be done.

Comment: The boiling-water-reactor transient "multiple safety-relief valve failure to operate" is not truly a reactivity accident. (NUMARC)

Response: The boiling-water-reactor transient "multiple safety relief valve failure to operate" has been changed to "overpressurization with limited relief."

Comment: In Section 2.1.3 (page 2-8 of the draft report), the events listed for pressurized-water reactors should include power excursions during refueling operations for completeness. (SNL)

Response: The staff does not believe that a Chernobyl type of event would occur during refueling operations; therefore, no change is necessary.

Comment: In connection with the positive moderator reactivity coefficient in a pressurized-water reactor, the potential reactivity addition at operating conditions is overstated. Furthermore, in the event of no external action, the pressurized-water reactor under these conditions will eventually come to a stable, zero-power condition at elevated temperature. (W)

Also overstated is the maximum potential reactivity insertion from cold conditions. Moderator heating cannot be accomplished at any significant rate without nuclear heating and any nuclear heat will bring with it prompt negative Doppler feedback. (W)

Response: In Section 2.1.3, under item (3), "Positive Moderator Reactivity Coefficient in a Pressurized-Water Reactor," the staff has added the explanation that the numbers are taken from Final Safety Analysis Reports in which extremes of allowable operating conditions limited by Technical Specifications are used.

Comment: Probability screening may be an effective way to reduce the numbers of accidents appearing in the list in Section 2.1.3 (page 2-8 of the draft report). It is strongly suggested that reviews of these accidents be performed by the NRC or its contractor rather than the licensee. (PNL)

Response: The staff agrees. It intends to use a probability screening process to prioritize transients and has added a sentence in the last paragraph of Section 2.1.3 to this effect. It also has made a change in Section 2.1.4 to indicate that the transients listed in Section 2.1.3 will be analyzed by the NRC or NRC contractors.

Comment: The Chernobyl accident should prompt the NRC to reopen the Anticipated Transients Without Scram (ATWS) rulemaking. The final ATWS rule fell far short of the staff's 1980 recommendations in NUREG-0460, Volume 4, particularly for boiling-water reactors. Unmitigated ATWS in boiling-water reactors poses a threat not unlike the reactivity excursion that occurred at Chernobyl. A pressurization transient (e.g., main steam isolation valve closure) without recirculation pump trip would result in an autocatalytic pressure-power spiral, with the negative void coefficient. The pressure pulse would collapse the voids, increasing power, which increases pressure, which collapses voids, which increase power, etc. Such an accident is, in the staff's judgment, apparently of such low probability that it might be ignored (NUREG-1251, p. 2-8). "Probability" (and our assessment of it) has been demonstrated by the Chernobyl accident not to be a useful concept in addressing reactor risk and safety. Had a risk analyst been asked prior to April 1986 to estimate the probability of this sequence of events leading to a severe accident at an RMBK reactor, the analyst would no doubt have replied that such a combination of events, errors, and deliberate procedure violations is so unlikely that it need not be considered. But it happened. The NRC should examine the events listed in Section 2.1 of NUREG-1251 and implement appropriate design changes or operating limits regardless of the perception of the probability of these events. (OCRE)

Response: The staff has revised the last paragraph in Section 2.1.3 to address uncertainties in probabilities. The last sentence of the paragraph was also revised to indicate that this study may be a basis for changing the ATWS rule.

2.1.4 Conclusions and Recommendations

Comment: Section 2.1.4 suggests that a reexamination of accident sequences and design approvals of the older plants may be warranted if only to "reconfirm their validity." The basis for this conclusion is that more "sophisticated tools" are now available. It should be noted that reload analyses utilize approved state-of-the-art methods to support core reloads whenever there have been plant modifications or changes in the methodology to correct an analysis deficiency. Therefore, the more sophisticated tools are utilized when required to ensure the safety of the plant. These tools include both the NRC-approved methodologies and the awareness of current events that could change the conclusions of an existing accident sequence. Further attention to this area of concern is not warranted. (NUMARC)

Response: The reference to "sophisticated tools" has been deleted, and Section 2.1.4 has been reworded.

Comment: The commenter believes that the ongoing efforts to evaluate severe accidents are adequate and that it is unnecessary to continue to expand the events to be evaluated until the current work is complete, has been subjected to a thorough technical review, and has been accepted by the technical community. Rather than begin new studies, both the NRC and the nuclear industry should concentrate their efforts on completing the review of NUREG-1150 and related work. (NYPA)

Response: The staff believes that there is a need to confirm the validity of previous conclusions using probabilistic methods. Section 2.1.4 was revised to clarify it in this regard. Also, a sentence was added to the effect that the results of the study on reactivity transients will be coordinated with severe accident programs and will be available to the industry.

2.2 Accidents at Low Power and at Zero Power

Several of the comments addressed under Section 2.1, "Reactivity Accidents," pertain to this section as well.

2.2.1 Current Regulatory Practice

Comment: On page 2-9 of the draft report, under item (1), "Steamline Break," the staff states that Combustion Engineering has only documented the full-power steamline break analysis in the Combustion Engineering Standard Safety Analysis Report (CESSAR). It should be noted that Appendix C to Chapter 15 of CESSAR documents the zero-power steamline break analysis. (CE)

Response: The error has been corrected.

Comment: The design-basis analyses discussed in Section 2.2.1 (page 2-10 of the draft report) depend on assumptions concerning the operability of plant systems (e.g., electrical power). They may not always be bounding from a risk standpoint. (SNL)

Response: The staff agrees with the comment that design-basis analyses may not be bounding from a risk standpoint; therefore, the analyses in Sections 2.1 and 2.2 will be coordinated with the implementation of the Severe Accident Policy. The staff has added sentences in Sections 2.1.4 and 2.2.4 to address this point.

2.2.2 Work in Progress

Comment: The draft report appears to define zero-power accidents as accidents that initiate in modes 4, 5, or 6. As such, the zero-power concern appears to be directed at decay heat removal, not reactivity transients. These zero-power (decay heat removal) light-water-reactor accident sequences bear no relation to the Chernobyl accident sequence. Although studies of such sequences may be appropriate, their linkage to Chernobyl appears tenuous.

In Section 2.2, the staff states that "the entire subject of decay heat removal is being addressed in Unresolved Safety Issue A-45." This appears to imply that A-45 is addressing explicitly these low-power and zero-power sequences. This is not the case. However, these sequences have been studied by both NRC and industry and are the subject of a number of reports by the Electric Power Research Institute (EPRI), Institute of Nuclear Power Operations (INPO), and the NRC Office for Analysis and Evaluation of Operational Data (AEOD). Under

Generic Issue 99, the NRC is considering some of the zero-power sequences already analyzed by EPRI, INPO, and AEOD. NRC Generic Letter 87-12 also addresses zero-power sequences for pressurized-water reactors. Taken together these industry and NRC programs adequately address the zero-power accident. (NUMARC)

Response: The reference to the subject of decay heat removal, Unresolved Safety Issue A-45, has been deleted because the Chernobyl accident does not directly suggest a problem regarding decay heat removal.

2.3 Multiple-Unit Protection

Comment: The general tone of this section is that no additional NRC or utility action is required. The document's summary, however, seems to recommend something more. (PNL)

Response: Section 2.3 does state that shutdown-system sharing in new plants should be restricted. It also states that control room habitability will be addressed in light of new source terms and that contamination outside the control room should be considered for new plants. The Summary is consistent with these statements. (Compare Section 2.3.4 and item (4), "Multiple-Unit Protection," in the Summary.)

Comment: The principal question with respect to multiple-unit sites appears to be control room habitability in severe accidents. Extensive work has already been done to address this area. (NUMARC)

Response: The thrust of this section includes but is larger than control room habitability during severe accidents. No change is warranted.

2.3.1 Current Regulatory Practice

Comment: Current best-estimate source terms for severe accidents are quite different from those used to design control room filters. (SNL)

Response: The report clearly indicates that severe-accident research may result in additional control room habitability criteria.

2.3.4 Conclusions and Recommendations

Comment: Sharing of systems should not be arbitrarily precluded for new plants. The goal should be to design systems so that a single common cause cannot compromise both units. (SNL)

The last sentence of Section 2.3.4 appears to contradict itself. (SNL)

Response: The intent of the concluding statement is generally consistent with the commenter's view. The last sentence has been rewritten for better clarity. (It is the last two sentences now.)

2.4 Fire Protection

Comment: Since, unlike RBMK cores, light-water-reactor (LWR) cores cannot burn, LWR operators need not be prepared to fight fires that could simultaneously release extremely dangerous amounts of radioactivity. LWR fires can threaten reactor safety, as evidenced by the 1975 Browns Ferry fire, but the firefighting aspects of such an event are separated in both time and distance from the reactor core. Firefighters at U.S. LWRs should be trained and equipped to fight the fire scenarios that are credible for our reactor design. The commenter does not believe that this should include the extremely high radiation fields associated with uncontained burning reactor cores. (NUMARC)

Response: A paragraph has been added, near the end of Section 2.4.3, which points out this significant difference between the Soviet RBMK and U.S. light-water reactors. However, as noted in that paragraph, a defense-in-depth safety philosophy nevertheless requires the ability to fight fires with radiation present, even in the absence of extreme, Chernobyl-like conditions.

Comment: Appendix R has led to major reviews of plant fire protection beyond the original licensing studies of fire protection. Many fire protection initiatives and plant modifications resulted. These and related efforts have brought about a very high degree of fire protection and mitigation capability at U.S. reactors. New programs are not warranted. (NUMARC)

Response: Although the effects of the efforts referred to by the commenter are recognized, the NRC under its research program is investigating the risk significance and uncertainty sources related to such issues as smoke control, control system interactions, interactions between seismic events and fire, and adequacy of fire barriers. Firefighting with radiation present is considered when evaluating fire-related issues. Statements have been added at the end of Sections 2.4.3 and 2.4.4 to clarify this perspective.

Comment: NRC fire protection regulations leave many loopholes. The development of requirements and their enforcement have been slow. (NIRS)

Response: The staff recognizes that there may be a need to further enhance fire protection. A sentence added in Section 2.3.2 notes that the NRC is currently evaluating the need for research on smoke control and manual firefighting. Two paragraphs added at the end of Section 2.4.3 note that the NRC is also evaluating the risk attributed to firefighting with radiation present and the risk and uncertainties of various other issues (control system interactions, interactions between seismic events and fire, fire barriers, etc.) A sentence added to Section 2.4.4 notes that any additional improvements in fire protection for which the results of this research indicate a need will be determined at the conclusion of the research.

CHAPTER 3--CONTAINMENT

3.1 Containment Performance During Severe Accidents

Comment: The substantial containments of U.S. reactors contrast sharply with the absence of such containments at Chernobyl. (YA, CE)

These differences should be recognized when considering additional areas of study. (CE)

No new programs or initiatives are required as a result of Chernobyl. (YA)

Response: The differences are recognized; no new programs have been recommended. However, the Chernobyl lessons should be taken into account in ongoing programs as discussed in the report.

Comment: Chapter 3 is written with a perspective that containments, as currently designed, are unable to withstand the challenges of severe accidents, and that modifications such as venting are required to reduce the uncontrolled release of radioactive material. Nuclear Management and Resources Council (NUMARC) containment integrity evaluations are portrayed as initiatives to identify such modifications for boiling-water-reactor containments. In fact, it is not a foregone conclusion that modifications to containments are necessary. The NUMARC effort on containment integrity started by assessing boiling-water-reactor Mark I containment integrity rather than assuming an answer. Results to date show no overriding generic vulnerabilities. Rather, they confirm the wisdom of performing plant-specific severe-accident evaluations of entire plant systems, including containment, as called for in the NRC's Severe Accident Policy Statement. To assume a conclusion and implement generic modifications creates the risk of unnecessary, and perhaps even improper, design changes. (NUMARC)

Response: The text was modified to better characterize NUMARC's efforts.

Comment: NUREG-1150 should be referred to as draft NUREG-1150. (NUMARC)

Response: The change has been made.

3.1.2 Work in Progress

Comment: Filtered venting is only one aspect of overall accident management to minimize release of fission products from containment. A more integrated approach is needed for severe-accident-mitigation concepts. (SNL)

Response: The report clearly states that containment venting is being considered by the NRC as one of many options being considered in its overall approach to accident mitigation. No change is required.

3.1.4 Conclusions and Recommendations

See also last comment and response in Section 3.2.

Comment: The conclusion that current programs in the United States are adequate and that new programs or initiatives are not needed appears to deserve more justification than it has been given in the document. (SNL)

Response: As noted in the report, the Chernobyl accident called attention to certain issues associated with containment, such as severe-accident considerations (Section 3.1), the venting issue (Section 3.2), and the combustible gas issue (Section 5.3). However, the Chernobyl accident graphically demonstrated the value of substantial containments, such as those required for U.S. light-water reactors, for control of the overall risk of nuclear power plant operation. The Chernobyl experience does not establish a need for a general re-justification of those requirements, although it did reinforce the appropriateness of the work now in progress.

Comment: A review of containments should be conducted in conjunction with severe-accident reviews. (PNL)

Response: The NRC staff's containment performance improvement program has been identified in the text. This program together with the Individual Plant Examination Program and detailed risk studies should provide a sufficient U.S. review of containments. These analyses consider a large spectrum of accidents including containment reponses.

3.2 Filtered Venting

Comment: Studies to date indicate that filtered vents have advantages in some sequences while presenting competing risks for other sequences. The net benefit of adding filtered vents may not be as positive as perceived by its proponents.

Also, this section tends to use the terms "filtered venting" and "venting" for boiling-water reactors (BWRs) interchangeably. Filtered vents are specific backfit structures, whereas "venting" as currently defined as an emergency procedure guideline strategy uses the BWR suppression pool for scrubbing. (NUMARC)

Response: Filtered venting has been suggested as a containment improvement and is appropriate for discussion in the report. The staff tended to use the terms "filtered venting" and "venting" interchangeably in the report with regard to BWRs. Since venting is likely to be through the suppression pool, this would also make it a filtered vent, given the presence of substantial radioactivity in the drywell.

Comment: Many U.S. containments are of questionable strength, in particular the General Electric plants. The report does not discuss the dangers during de-inertion periods. Various degradations of containment reliability are caused by utilities ignoring the regulations. (NIRS)

Response: The report was not modified to reflect this comment. Containment de-inerting (for MARK I and MARK II boiling-water-reactor containments) is permissible for limited periods. NRC's inspection and enforcement staff, which includes NRC resident inspectors at each operating plant, monitors reactor operations carefully for actions not in compliance with NRC regulations. This issue, however, is being evaluated under the staff's containment performance improvement program.

Comment: The commenter presents arguments for a general view that the NRC actions indicate insufficient respect for the importance of containments and urges the NRC to formulate containment performance requirements for severe accidents without delay. (OCRE)

Response: The report has not been modified because it clearly states that the NRC has existing programs under way to examine and assess improvements in containment and plant performance under severe-accident conditions. The scope and timing of these efforts are considered appropriate.

Comment: Containment integrity and containment venting should be of primary importance in reactor safety research. (IDNS)

Response: The report was not modified because it clearly states that detailed studies are being carried out to investigate venting and also describes other ongoing work in regard to containments.

CHAPTER 4--EMERGENCY PLANNING

Comment: The report fails to examine the benefits of the Soviets' evacuation and sheltering methods and to determine U.S. regulatory areas in need of upgrading, such as evacuation beyond the 10-mile emergency planning zone, ingestion zone size, preparedness, and relocation issues. (NIRS)

The possibility of protective actions being required beyond the 10-mile emergency planning zone should be emphasized more. (IDNS)

Emergency planning should be based on the worst case. We should try to learn more from Soviet mitigation measures. (OCRE)

Response: The report has been modified to provide clearer explanations relative to these concerns. Additional detail has been provided regarding the Chernobyl evacuation and some actions taken by the Soviets to lessen the impact.

The report has been modified to provide a clearer rationale for the size of the planning zones and to explain why these are not based on worst-case assumptions.

The concluding paragraphs of each section of this chapter address plans for further studies and possible reexamination of various stated issues. Much of the further work planned or recommended by the staff addresses issues raised in these comments.

Comment: The report does not substantiate the conclusion that some aspects of emergency planning need to be reexamined. (NC)

Response: The report was not modified in response to this comment. The staff plans to reexamine some aspects of emergency planning, consistent with insights on accidents applicable to U.S. reactors. The reasons for the conclusion are given in the report.

4.1 Size of the Emergency Planning Zones

Comment: Emergency plans must be sufficiently flexible so that adjustments for emergency circumstances, such as those that may affect evacuation routes, can be made. (SNL)

Response: U.S. emergency planning is significantly flexible with regard to evacuation routes, etc. There is no need to modify the report.

4.1.3 Assessment

Comment: The staff refers to the United States as having experienced its "Chernobyl" accident at Three Mile Island Unit 2 in 1979. Regardless of the emergency planning context in which such a statement was used, it is inappropriate and should be removed. (YA)

Response: The statement has been removed.

4.2 Medical Services

Comment: Inclusion of the Polish experience with potassium iodide, including any problems, would provide a more comprehensive view of the use of this substance as a thyroid blocking agent. (NUMARC)

Response: The report has been modified to include a statement regarding the Polish experience with potassium iodide.

Comment: Chernobyl clearly demonstrates the need to stockpile potassium iodide. (ANBEX)

There should be predistribution of potassium iodide. (OCRE)

The Soviet and Polish experience with potassium iodide should be considered more fully. (IDNS)

Response: The report was revised to provide a clearer explanation of Federal policy regarding the use of potassium iodide and of the rationale underlying that policy.

4.2.4 Conclusions and Recommendations

Comment: The magnitude of the required medical response to a severe accident at a highly populated site may exceed local resources. (SNL)

Response: There is no need to modify the report as it is believed that sufficient medical resources can be quickly marshalled in the United States in the event of an accident.

4.3 Ingestion Pathway Measures

Comment: Aspects pertaining to ingestion and decontamination, particularly as they are affected by the differences between Soviet and U.S. conditions, should be more clearly explained. (NUMARC)

Response: Sections 4.3 and 4.4 have been modified to provide additional discussion of ingestion and decontamination.

4.4 Decontamination and Relocation

Comment: The report should be updated to reflect the results of the June 1987 Federal field exercise in the area of the Zion plant in Illinois. (NUMARC)

Response: The report has been modified to reflect experience gained from that exercise.

CHAPTER 5--SEVERE-ACCIDENT PHENOMENA

Comment: Although the statement in the first paragraph of Chapter 5 that "the specific accident mechanisms involved at Chernobyl have no exact parallel in

U.S. reactors" is true, we should concentrate on commonality of the potentially destructive phenomena, not on the differences in accident initiation. (SNL)

Response: The staff's intent is consistent with the commenter's view, is reflected in the balance of this chapter, and will be borne in mind in followup activities. No change in the report is necessary.

5.1 Source Term

Comment: The report states that the Chernobyl release reached a height of 1000 to 2000 meters. However, a large amount of material also rose to about 7000 meters, most likely as a result of meteorological convective activity. (LLNL)

Response: The 1000 to 2000 meters is identified in the report as an initial plume height. The statement in the report is not inconsistent with the second part of the comment. No change is needed.

Comment: A recent study sponsored by the U.S. Department of Energy concluded that the first day's releases were approximately 10%, not 25%, of the total release. (NUMARC)

Response: The figure of approximately 25% is based on the Soviet report (see USSR, 1986, in the reference section of the main report). A statement about the study sponsored by the U.S. Department of Energy referred to by the commenter has been added. That study indicated larger total releases and an initial energetic release that was a smaller fraction of the total release.

Comment: Some hypotheses are given about the increased release rate on the sixth day and the decreased rate on the tenth day following the accident. It is not clear from the report whether there are any plans or possibilities for further studies to confirm one or more of these proposed scenarios. A reliable assessment of the release mechanisms which occurred at Chernobyl will be necessary in order to address the two major issues given at the beginning of the section. (LLNL)

Response: The staff is not aware of any studies to further establish the exact mechanisms of release. It seems reasonable to study all three hypothesized mechanisms in the context of light-water-reactor accident scenarios, and this is stated in the report. This is more important than trying to pin down the responsible mechanism for the largely irrelevant RBMK accident sequence. No change is warranted.

5.1.1 Current Regulatory Practice

Comment: The source terms in Atomic Energy Commission Report TID-14844 are very conservative bounding limits because containment buildings exist at U.S. light-water reactors. (NUMARC)

Response: The report accurately characterizes the TID-14844 release as the release into the containment. The NRC does not assume that TID-14844 quantities are released to the atmosphere. No change is needed.

Comment: The staff notes that the most severe release categories from WASH-1400 entail releases of volatile fission products of comparable or greater magnitudes than were released at Chernobyl, although the releases of low-volatility species were higher for Chernobyl. The staff should make clear that such larger releases need not be considered for U.S. light-water reactors because there is no counterpart to such an uncontrolled explosion and subsequent carbon fire as occurred at Chernobyl. (YA)

Response: Because larger releases can occur in light-water reactors as a result of other events (e.g., core-concrete interactions, high-pressure melt ejection, and oxidation by air after vessel failure), the statement suggested by the commenter cannot, in the staff's judgment, be made. No revision is needed.

5.1.2 Work in Progress

Comment: Although NRC severe-accident research is discussed in length, it should be noted that there also have been industry efforts in this area. It is appropriate to consider the Industry Degraded Core Rulemaking (IDCOR) research program and Electric Power Research Institute's (EPRI's) source term research program. These industry-supported initiatives serve to demonstrate the conservatism of the NRC position on severe accidents. (NUMARC)

Response: Section 5.1.2 has been modified to include the research by EPRI and under the IDCOR program. There is no basis, however, for concluding that the industry-supported initiatives serve to demonstrate the conservatism of the NRC position.

5.1.3 Assessment

Comment: The detailed discussion in Section 5.1.3 is interesting background information and interesting Chernobyl source term analysis, but has little relevance to light-water-reactor source terms. It could be shortened or eliminated. (NUMARC)

Response: The discussion is essential to this report in that it provides a basis for the assessment of implications by analogy for radionuclide releases in U.S. reactors. No change is needed.

Comment: Resuspension of fission products due to energetic hydrogen combustion is an issue that has been raised and briefly investigated. However, the commenter is not aware that this resuspension mechanism has been considered in any "source term evaluations," and certainly there has been no prior experimental confirmation. (SNL)

Response: Resuspension due to energetic hydrogen combustion may be relevant to NRC's light-water-reactor research effort. However, at Chernobyl hydrogen combustion occurred at the beginning of the accident, whereas the report discusses late enhanced releases. This has no counterpart in the Chernobyl scenario. A change in the text would not be appropriate.

Comment: The possibility of large containment openings occurring in the event of an accident, especially openings that might make possible aerial dumping that could reach core debris, is very remote. The fundamental strategy is to keep the core contained in the reactor vessel. (NUMARC)

Steel model containments have failed catastrophically, suggesting that a large containment opening may conceivably occur in some U.S. plants during a severe accident. (SNL)

The 1/8-scale steel containment model tested by Sandia National Laboratories in 1984 failed catastrophically. Studying the accident-management strategies used by the Soviets and establishing plans for their use are necessary. (OCRE)

Response: The report has been revised in light of these comments. The revised text retains the view that an extension of the statement that "large openings of the containment appear unlikely" to suggest that they will not occur at all is not justified. The revised section does emphasize that prevention is a vital element in accident management.

Comment: The statement (in item (3)(b), "Effects of Materials Deposited on the Core") that large openings of the containment in a severe accident are unlikely has no factual basis. The commenter refers to the 1984 scale-model tests at Sandia National Laboratories. (OCRE)

Response: The paragraph has been rewritten; the statement in question has been removed.

Comment: In Section 5.1.3 and several other sections in the report, the staff notes that the initial releases during the Chernobyl accident occurred with no warning and that such sudden releases would not be expected for accidents at western plants. However, there is a lack of coordination between U.S. evacuation plans and the analyses of severe-accident progression. We should use what we know about severe accidents to optimize accident management and emergency plans. (SNL)

The source term code package does not include models that address important radionuclide release phenomena such as revaporization, direct containment heating, and residual fuel oxidation. These phenomena need to be quantified and included in our predictive analyses. (SNL)

Response: Under its research program, the NRC continues to develop an experimental database to develop, assess, and improve models for revaporization, direct containment heating, and residual fuel oxidation. The results of source term research will be reflected as warranted in accident management, emergency planning, and other pertinent regulatory areas. The comment is not directly germane to the text; no changes were made.

Comment: A release of 10 days is not impossible, regardless of what is stated in WASH-1400. Industry Degraded Core Rulemaking Program predictions of revaporization showed releases of iodine from some plants beginning after almost 2 days. Releases of iodine from water pools could be very protracted according to analyses done at Oak Ridge National Laboratory. Mechanical aerosol generation by core debris/concrete interactions could cause releases of 10^3 to 10^5 curies/day some 10 days after accident initiation. (SNL)

Response: The implication noted by the commenter was not intended. The report has been revised so as not to make this implication. The revised text notes that the release rate differed from those usually predicted in analyses for U.S. plants.

Comment: Air oxidation of UO_2 might occur in a light-water-reactor accident sequence leading to a particulate or aerosol release. (SNL)

Response: This comment perhaps suggests that more attention should be paid to this subject in the light-water-reactor research program, and such a comment is entirely in line with the conclusions in the report (Section 5.1.4). However, to better address the thrust of this comment, the report was revised so as not to imply that there would be no large source of gas flow.

Comment: Another possibility is that the graphite slumped (which is known to have occurred) about this time and the slumping enhanced cooling or brought cool materials onto the hot debris. (SNL)

Response: A statement recognizing this possibility has been added.

Comment: What is the basis for the speculation (under item (3)(d), "Hydrogen Generation From Dispersal of Fragmented Debris") that the hydrogen-air mixing could not have occurred "rapidly enough"? (SNL)

Response: This is an apparent misreading of the text. The text said only that it was "not clear that the hydrogen could have become mixed...rapidly enough..."; it did not assert that it could not. Hydrogen generation and combustion are receiving major attention in NRC's research program. (The hydrogen issue is further discussed in Section 5.3.) No change is required.

Comment: The argument that the "source term is specific to the RBMK design and U.S. source terms are expected to be lower" (at the end of Section 4.1.3) is difficult to reconcile with statements that the "total quantity of fission products released from Chernobyl was large and is considered to be comparable with the quantities predicted to be released for the worst cases...studied for U.S. LWRs using...the most recent source term methodology" (Section 5.1.3, second paragraph). (IDNS)

Response: Section 4.1.3 has been revised to resolve this inconsistency.

Comment: Although the draft report indicates that the Chernobyl release can be matched by only the worst sequences from U.S. reactors, it fails to state where the Chernobyl sequence falls within the range of plausible accidents for Soviet reactor designs. The final report should clearly state that the accident sequence at Chernobyl also fell at the extreme edge of the Soviet spectrum. (IDNS)

Response: This report addresses implications for U.S. reactors. The staff has no specific basis for asserting the conclusion indicated by the commenter for Soviet reactors. However, it is hard to imagine a much worse accident. No changes are appropriate.

Comment: The most important conclusion of the source term analysis is that methodologies must be improved to derive source term estimates from environmental measurements. (IDNS)

Response: Valuable as environmental measurements can be, their scope of application to the derivation of source terms is necessarily limited because a great variety of potential accident sequences must be considered. The study of accident phenomena in controlled experiments along with forward calculations seems more appropriate than back calculations in this field.

5.2 Steam Explosions

Comment: This section should be shortened or eliminated because the "steam explosion" discussed in Vienna is unrelated to the specific steam-generated shock-wave definition associated with the term "steam explosion" as used by U.S. reactor safety experts. In Vienna, the term "steam explosion" was a general expression for mechanical failure of the reactor vault due to steam overpressure, such as in a fossil boiler explosion. This definition is quite different from the "alpha" containment failure mode discussed in WASH-1400. The discussion is interesting but of little relevance to light-water-reactor severe accidents because of this important semantic difference. (NUMARC)

Response: The staff agrees that the Chernobyl prompt-burst steam explosion is different from the steam explosions that can occur in the slow core meltdown that is relevant to the safety of U.S. light-water reactors. The text makes the distinction. Section 5.2 has been shortened and simplified.

5.2.2 Work in Progress

Comment: The reference given does not exist. The Severe Accident Risk Reduction Program (SARRP) Sequoyah containment event analysis report became NUREG/CR-4700, SAND86-1135. This report does not examine the sensitivity of risk to alpha-mode failure. Neither statement (1) nor (2) is contained in the Sandia National Laboratory/SARRP analysis. (SNL)

Response: The draft report referenced and quoted in the first paragraph of Section 5.2.2 was not published, and the material in it was changed before it was published in other reports. Therefore, this paragraph has been rewritten. The referenced report, SAND-1013, 1986, has been changed to NUREG/CR-4551, 1987, in the list of references.

Comment: The commenter does not know of additional technical assessments being made of "alternative contact modes, multiple steam explosions, and the potential effects of steam explosions on safety systems and/or functions." There is no ongoing research in the United States that will reduce uncertainties with respect to alpha-mode failure. (SNL)

Response: Much of the work discussed in the paragraph has been terminated. The last paragraph of Section 5.2.2 has been rewritten.

5.3 Combustible Gas

Comment: The NRC is failing to properly regulate the hydrogen-explosion-prevention area (among others). (NIRS)

Response: The current requirements and work in progress with respect to hydrogen control are discussed in Section 5.3.1/5.3.2. The issue is assessed in relation to the Chernobyl implications in Section 5.3.3. A general rejustification of current requirements and the nature, scope, and pace of ongoing programs is not within the scope of this report.

5.3.1/5.3.2 Current Regulatory Practice/Work in Progress

Comment: It would be more truthful for the NRC to admit that current hydrogen control requirements are probably final, rather than interim, pending longer term efforts regarding severe accidents. (OCRE)

Response: As discussed in Section 5.3.1/5.3.2, an additonal database for hydrogen control during postulated core-melt accidents is being developed under the severe-accident research program. Accordingly, inclusion of the assertion suggested by the commenter would not be correct.

5.3.3 Assessment

Comment: The commenter questions the analysis in Section 5.3.3. What experimental data or calculations support the belief that mixing could not take place in 7 seconds or less? The possibility of a hydrogen explosion or detonation contributing to the destruction of the Chernobyl containment, altering the chemistry of the fission products, and affecting their dispersion in the atmosphere by thermal and mechanical augmentation is very relevant to light-water-reactor accidents and should not be dismissed. (SNL)

Response: The relevant statements have been revised to address the detonation issue.

5.3.4 Conclusions and Recommendations

Comment: The Chernobyl accident suggests the importance of understanding many phenomena and the need for additional research on the rates and magnitudes of hydrogen generated during explosive and nonexplosive fuel-coolant interactions, the mixing of high-velocity steam-hydrogen jets with surrounding air, the potential for steam explosions to directly initiate detonations in hydrogen-air mixtures, the influence of high-velocity-flow-induced turbulence on deflagration-to-detonation transition (DDT), and the possibility that hot hydrogen-air mixtures are more prone to DDT than cold mixtures. (SNL)

Response: The assessment and conclusions (Sections 5.3.3 and 5.3.4) have been revised to take into account the commenter's point.

Comment: The NRC should require all containments to be inerted. Controlled ignition is too uncertain. (OCRE)

Response: Currently only the Mark I and Mark II type containments have to be inerted. Mark III boiling-water-reactor and ice condenser pressurized-water-reactor containments are required to have installed a hydrogen control system capable of accommodating an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water without loss of containment integrity. The particular type of hydrogen control system to be selected is left to the discretion of the licensee and must be found acceptable by the NRC on the basis of a suitable experimental and analytical database. The failure of igniters to operate during a station blackout degraded core accident could possibly be resolved by the use of a passive backup system. The feasibility of using a given passive catalytic igniter system has been investigated by the NRC, and the results seem promising. However, it is the NRC position that any additional work in this area would be developmental in nature and would therefore not be performed by the NRC. No change is needed.

CHAPTER 6--GRAPHITE-MODERATED REACTORS

6.2.2 Design

Comment: With regard to item (4), "Fires and Explosions," flooding of the reactor pressure vessel as a backup defense against a chimney fire at Fort St. Vrain seems extreme; the modular high-temperature gas-cooled reactor (MHTGR) should be designed so that chimney fires can be prevented without resorting to flooding. (LLNL)

Response: Work is currently under way to address the subject of chimney fires, and progress may show that, even with chimney-type geometry, graphite fires could not occur in either the MHTGR or Fort St. Vrain. This information was added to the report.

Comment: The NRC should study severe accidents beyond the design basis of both the high-temperature gas-cooled reactor (HTGR) and the modular high-temperature gas-cooled reactor (MHTGR), regardless of their perceived probability. To neglect these accidents would be missing the prime lesson of Chernobyl. The NRC should take a proactive approach with the MHTGR (and all proposed advanced reactor designs) so that unresolved safety issues are identified and resolved before the plants are built and operating. (OCRE)

Response: The report has been revised in partial response to this comment. The revised text cites a 1984 study of HTGR severe accidents, in addition to noting that a limited probabilistic risk assessment for Fort St. Vrain is in progress. Furthermore, a statement has been added concerning the consideration of severe accidents for the MHTGR. (Section 6.2.2, item 7, "Severe Accident Phenomena")

With regard to the commenter's recommendation that the NRC should take a proactive approach, it is noted that the MHTGR is being proposed in conformance with NRC's Advanced Reactor Policy Statement. The proactive approach was considered in the development of this policy statement and need not be reconsidered here.

IV. PUBLIC COMMENTS RECEIVED

This section provides the comment letters that were received. The page on which each letter begins is as follows:

The chapters or sections to which the comments primarily pertain are noted in the margins of the comment letters unless they are prominently identified in the letters.

ANBEX
BOX 861 COOPER STATION N.Y. N.Y. 10276
(212) 505-6212

November 13, 1987

Division of Rules and Records
U.S. Nuclear Regulatory Commission
Washington, DC 20555

As per the notice in the Federal Register, this is to present my comments on the draft document issued by your Agency entitled "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States", NUREG-1251, published August 1987.

My interest in this document comes from the fact that my company, ANBEX, is one of the two US pharmaceutical firms approved by the FDA to produce potassium iodide (KI) tablets for use in a radiation emergency. Accordingly, I closely followed the use of KI at Chernobyl, and read with interest your document: "Report on the Accident at the Chernobyl Nuclear Power Station" NUREG-1250.

To summarize my comments, I wish to express my objections to your conclusions reached in NUREG-1251 concerning KI. In light of Chernobyl, the policy you advocate is insensitive to the needs of Americans, and incompatible with the events that occurred there.

Specifically, you report in NUREG-1250 that: 4.2

> The Russians were apparently well prepared for large-scale distribution of KI tablets to the general public....Thousands of measurements of I-131 (radioiodine) activity in the thyroids of the exposed population suggest that the observed levels were lower than those that would have been expected had this prophylactic measure not been taken. The use of KI by the Pripyat population in particular was credited with permissible iodine content (less than 30 rad) found in 97% of the 206 evacuees tested at one relocation center. It is also important to note that no serious side effects of KI use have been reported.

But in spite of this acknowledgement of the value of the Russian stockpiles, and the known lack of any similar substantive stockpiles in the Unites States, you conclude in NUREG-1251 that:

> The apparently successful use of KI by the Soviets does not alter the validity of U.S. Government policy that predistributing or stockpiling KI for use by the general public should not be required. Rather, this decision should be made by individual States and by local authorities.

This policy is inconsistent with your responsibilities and is clearly not in the best interests of citizens of the United States. It guarantees that KI will not be available if it is needed, and consequently one of the most effective public health measures taken by the Soviets will be denied to Americans.

It is my contention, in fact, that the the events at Chernobyl, once again, clearly demonstrate the need to stockpile KI, and the necessity for responsible health officials to insure that sufficient supplies are available in the event of a similar accident here. Not to do so is to introduce a significant risk of preventable injuries with no corresponding benefit.

It is insufficient and an abdication of your responsibility to put the burden of securing KI supplies on State and local governments. Stated simply, unless Washington acts, smaller governments won't, since most lack an understanding of the benefits of KI, and, quite appropriately, look to Federal agencies for guidance on nuclear policy issues. Thus, your failure to require predistribution or stockpiling assures that the benefits of the agent that is credited with protecting hundreds of thousands of Russian citizens will be lost to Americans. Assuming you accept the credible possibility of an accident, one has to wonder why you appear not see the potential danger of this policy. 4.2

The willingness to reject KI in spite of the safety benefits it offers can not be explained by what took place at Chernobyl. Instead, it can only be understood by realizing that the conclusions in NUREG-1251 come from ignoring what happened there.

NUREG-1251, itself, essentially says this. It notes:

> While valid arguments may be made for the use of KI, the preponderance of information indicates that a nationwide requirement for the predistribution or stockpiling for use by the general public would not be worthwhile. This is based on the ability to evacuate the general population and the cost effectiveness of a nationwide program which has been analyzed by the NRC and DOE National Laboratories (NUREG/CR-1433).

But NUREG/CR-1433 was written 6 years prior to Chernobyl, and is seriously flawed. The basis for its cost effectiveness argument is the unrealistic assessment that reactor accidents that release significant quantities of radioiodine into the atmosphere would not be expected to happen more than about once in a thousand years. And under this infrequent occurrence, so the argument goes, there is no need for KI.

But the "millennium" estimate was generated before the accidents at Chernobyl, Three Mile Island, and a host of other near misses. Consequently, no one, including the NRC, accepts this probabilistic estimate today, and more recent figures based on more years of actual operating experience suggest that serious core-melt accidents are likely to occur far more frequently. In fact, I am advised that a serious core-melt accident about every 20 years would not be improbable.

Obviously, as others have pointed out, under this expected frequency, the "cost-effectiveness" argument presented in NUREG/CR-1433 is invalid. Yet for the NRC to quote it in NUREG-1251, in spite of your official rejection of the probabilities it is based on, reflects poorly on the conclusions you draw.

In addition, your conclusion not to predistribute or stockpile KI also rests on a presumed "ability to evacuate the general population" in case of an accident. I suspect, though, that few independent experts would agree that this could easily be done. As you know, an area of 1000 square miles could be endangered in a major iodine release, and this could require the evacuation of millions of people. To presume the ability to do this, and, consequently, not to acquire KI as a result, strikes me as dangerous policy. Further, as your document points out, although the Soviets were able to move hundreds of thousands of people in a relatively short time, there are significant differences between their society and ours. And in any event, those being evacuated would be far safer if they also took KI. 4.2

Finally, I must take exception to your statement that "the preponderance of information indicates that a nationwide requirement for the predistribution or stockpiling for use by the general public would not be worthwhile." This is simply incorrect. There is no "preponderance" of information against stockpiling or predistributing KI. In fact, the opposite is true. For example, the National Council on Radiation Protection has strongly endorsed KI, and noted, "every available appropriate outlet should be considered as a stockpiling and distribution point". And the Presidential Commission that investigated Three Mile Island (the Kemeny Commission) made the recommendation that: "An adequate supply of the radiation protective (thyroid blocking) agent, potassium iodide for human use, should be available regionally for distribution to the general population...affected by a radiological emergency."

Further, the FDA's highly supportive position regarding KI is well known, the American Thyroid Association has concluded that "Potassium iodide in an appropriate dosage form (130 mg scored KI tablet) be manufactured in sufficient quantities should its usage be required", and acceptance of stockpiling is an acknowledged fact in numerous European countries (which is why it was available at the time of Chernobyl). Also, the testimony by recognized experts before the House Committee on Interior and Insular

-3-

Affairs, and most of the published literature on this question is overwhelmingly in favor of the use, and/or predistribution, and/or stockpiling of the drug.

In fact, with the exception of the commercial nuclear power industry and a small group who support Dr. Rosalyn Yalow, it is difficult to find anyone who (or anything written that) is "against" potassium iodide.

Finally, one should not ignore the fact that early KI distribution has already been requested. As we know, at Three Mile Island it was called for immediately (although it was not available for nearly 6 days, a delay which planners were criticized for), and at Chernobyl it was distributed from stockpiles within hours of the accident. Surely, one must expect that if another accident occurs, the people charged with the responsibility of dealing with medical issues will also demand early supplies.

But where these supplies are to come from has apparently never been explored or considered. I can assure you that neither my company or the only other supplier keep substantial quantities in inventory, and it seems almost certain that if it is ever required, then once again there will be a frantic search for the drug, followed by the hurried preparation of inadequate supplies of an inferior substitute, and no delivery until it is too late. 4.2

Stated simply, unless one starts with the assumption that core-melt accidents are impossible (in which case there should be no need for any emergency preparedness), it is difficult to make a coherent argument against keeping KI on hand. In addition, because the product is inexpensive, easy to store, and comes (in the case of KI sold by ANBEX) with a guaranteed shelf life, there are no serious logistical obstacles to stockpiling.

Although the question can never be fully answered, one has to wonder what the impact of Chernobyl would have been on the health of affected populations if the Soviets had not had KI in stockpile. Certainly, this document makes it plain that matters would have been much worse. For FEMA and the NRC not to act in recognition of this is a poor, and extremely dangerous, oversight.

Sincerely,

Alan Morris

Alan Morris
President
ANBEX, Inc.

If you wish clarification of any of my comments, I can be reached at (201) 586-9282

For Comments, please note
my new address:

Amber
c/o Alan Morris
113 Morris Ave
Denville, NJ 07834

201-586-9282

COMMENTS OF FORMER COMMISSIONER ASSELSTINE (2/87)

I cannot approve this document. Apart from the rather obvious grammatical, stylistic, and inconsistency problems which others have already highlighted, I do not agree with the substance of the report. The report, like previous industry and government statements on the Chernobyl accident, takes an overly narrow view of the lessons of the accident. Not surprisingly, the report concludes that there are no immediate changes needed in this country to address the lessons of Chernobyl. I continue to believe that this misses the point. Specifically, the report should address such questions as: how does the Chernobyl release category compare with the release categories which are possible for U.S. reactors; what is the likelihood that such releases will occur at a U.S. plant over the next fifty years, and can we say with high assurance that we have reduced that likelihood to an acceptably low level; are such releases and their likelihood acceptable from a public health and safety and offsite property contamination standpoint; if they are unacceptable, what practical measures can be backfitted on U.S. plants to reduce these risks; and finally, what lessons can we learn from the measures being adopted after Chernobyl in the European countries, such as Sweden, Switzerland, Finland, West Germany, and France, and what can we learn from the way that these countries make backfitting decisions? If the report fails to address these more fundamental issues, I will have additional views to be included in the Federal Register notice seeking public comment on the report. Summary

Battelle
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Telephone (509) 375-2878
Telex 15-2874

October 29, 1987

Dr. Eric S. Beckjord, Director
Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Beckjord:

Reference: Letter Eric S. Beckjord to W. R. Wiley, dated September 8, 1987

I have had appropriate members of my staff review the draft NRC report, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plant in the United States," NUREG-1251. I am pleased to forward our comments.

We appreciated the opportunity to review this document, particularly in light of the extensive efforts that PNL staff have expended in the past 18 months in understanding the Chernobyl accident and assessing its implications.

We would welcome the opportunity to provide any further assistance in this area that you might require.

Sincerely,

Thomas T. Claudson

Dr. Thomas T. Claudson
Director, Engineering Technology
Pacific Northwest Laboratory

TTC:gw

cc: WR Wiley

Battelle

ATTACHMENT

TECHNICAL COMMENTS ON NUREG-1251

"Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States"

General Comments

The report is quite thorough in its analysis, evaluation and comparison of the implications of the accident at Chernobyl and the design features, regulations, and administrative controls at US nuclear plants. NRC has done a good job of considering all agency programs in their evaluation of the accident implications. PNL reviewers agree with the overall conclusions drawn in the report that:

1. No immediate changes are needed in the NRC's regulations regarding the design and operation of commercial power plants in the US

2. Some reexamination of existing requirements and those under development may be warranted in light of the accident

3. Some accident types may require further analysis or consideration by NRC (e.g., reactivity accidents, accidents at low or zero power)

4. The experience gained from the Chernobyl accident should remain as part of the background to be considered in future reactor safety issues.

Specific Comments

Section 1.1.1 Procedure Development and Use. In the discussion about bypassing safety systems, the report discusses administrative controls on procedures, permanent procedure changes, and temporary procedure changes. PNL reviewers suggest that controls on procedure changes be strengthened. We have observed situations where a given procedure has multiple change forms attached to the front of the procedure that modify the body of the procedure. These changes pile up on the procedure, increase the risk of human error, and are not always incorporated into the body of the procedure through the revision (rewrite) process in a timely way.

The control of temporary procedure changes should also be evaluated carefully. Situations, particularly on backshifts and weekend shifts, could arise where inadequate consideration is given to the safety effects of temporarily revising a procedure. Currently, the criterion is that the change not alter the intent of the procedure. We observe that this criterion could be met while a significant alteration outside the intent of the procedure may occur to reduce the safety of plant operations. We suggest that the review criterion for temporary procedure changes be expanded in scope to consider more than the specific intent of the procedure being changed.

The report states "procedures are violated in licensed plants, but only rarely with the knowledge that a violation is being committed." PNL reviewers object to the implication of this statement that procedures may be routinely violated and that accountability is poor. We believe that this is not the intent of the report's authors. A more appropriate statement is "procedural violations do occur infrequently, however, seldom are the procedures knowingly disregarded."

Section 1.1.2 Symptom/Function-Based Emergency Operating Procedures. This section indicates that licensees are required to implement symptom/function-based EOPs incorporating good human factors practices. PNL's reviews of new EOPs for NRC indicate that licensees are trying to incorporate human factors practices into EOPs, however, we find a wide variation in the level of human factors principles actually applied. This suggests that some guidance may be needed to improve the overall level of human factors considerations in the procedures.

Section 1.2.4 Approval of Tests and other Unusual Operations. PNL reviewers agree with the conclusions of this section that the results of the joint NSAC/AIF effort should be reviewed. We suggest, however, that in spite of these results, if NRC has found instances of inconsistent documentation or too narrow a determination of the unreviewed safety question considerations, NRC should provide additional guidance to industry in this area. The guidance may need to consider appropriate scope and levels of detail to be contained in 10 CFR 50.59 reviews and additional guidance on unreviewed safety question determinations.

Section 1.3.4 Bypassing Safety Systems. This section describes work in progress to assure the availability of redundant safety systems and equipment. The NRC's Maintenance and Surveillance Program indeed has objectives that focus on improving the availability of all plant equipment, however, since the conclusion of the Phase I effort several years ago, the level of effort devoted to this program is very small. Given the importance of maintenance in assuring the availability of plant equipment, the reviewers doubt that the level of effort is commensurate with the importance of maintenance. We recommend more resources be devoted to improving NRC's ability to evaluate maintenance performance.

Section 1.6.4 Management Systems. This section recommends the consideration of a dedicated high-level, onsite, nuclear-safety manager with no other responsibilities or duties. PNL reviewers discourage this notion on the grounds that the plant manager should be ultimately responsible for the safety of the plant. The creation of a nuclear safety manager, independent from the plant manager, as is the onsite QA manager, opens the door for additional conflicts, finger-pointing and transferring of responsibilities. The plant manager should be responsible for all aspects of plant operations and clearly understand that safety is of overriding importance. He should be held responsible and accountable for all aspects of plant operation; he has the power and the responsibility by virtue of his position. Furthermore, there are enough checks and balances on safety required by each plant's technical specifications. They require an onsite safety review group, independent of plant operations, to review all aspects of plant operation. Additionally, the tech specs charter a corporate safety review group to check all aspects of plant operation as well as to oversee the work of the onsite review group. These groups have the power to halt unsafe operations. The reviewers suggest that the guidance and training for these independent review groups is much more effective than an onsite nuclear-safety manager.

Section 2.1 Reactivity Accidents. A positive void coefficient was identified as a primary factor in the Chernobyl accident. NRC concludes PWR's "generally have a negative temperature and void coefficient in the normal operation range." PNL reviewers suggest that in light of the severe consequences of the Chernobyl accident, perhaps a 3-D analysis of the total fuel cycle with special attention to local perturbations is appropriate.

The RBMK reactors were to be altered after the accident to limit control rod withdrawal and to increase the number of control rods. While these are RBMK-specific fixes, PNL reviewers observe that the US PWRs have two independent safety shutdown systems (control rods and boron), however, it is understood that the control rods cannot control the reactor power level in the event all the boron is lost. It seems that increasing the number of control rods in PWRs warrants additional consideration by NRC. The impact on plant design, efficiency and cost should be elevated.

Pump cavitations and loss of flow, particularly at low power, was a known concern of the Soviets and could have been a contributor to the accident. PNL reviewers suggest that NRC may want to evaluate more fully the potential effects pump cavitation may have on commercial US reactors, particularly BWRs during low power or during accidents/transients.

This section (page 2-2) discusses the moderator temperature coefficient in some detail. Some parts in the section do not specify moderator temperature coefficient. If the discussion intends to discuss the overall temperature coefficient, we observe that often the overall temperature coefficient is controlled by the fuel temperature coefficient.

Section 2.1.1 Current Regulatory Practice in Reactivity Accidents. Reactivity changes do occur with temperature changes, however, the central issue is the injection of cold water.

Section 2.1.2 Assessment of Reactivity Accidents. Probability screening may be an effective way to reduce the numbers of accidents appearing in the list on page 2-8. PNL reviewers strongly suggest that reviews of these accidents be performed by the NRC or its contractor rather than the licensee. We anticipate that the results would have more wide-ranging applicability and cost less.

Section 2.3 Multiple-Unit Protection. The general tone of this section is that no additional NRC or utility action is required. The document's summary, however, seems to recommend something more.

Chapter 3 Containment. PNL reviewers agree with the general assessment of the NUREG, however, the central lesson has been overlooked. The Chernobyl reactor had a containment designed to withstand a specified design basis accident. Clearly, a credible accident beyond the design basis occurred. A concerted, formal review of containment DBAs should be conducted in conjunction with reactivity, low power, and severe accident reviews suggested in this report.

COMBUSTION ENGINEERING

November 2, 1987
LD-87-061

Mr. Samuel J. Chilk
Office of the Secretary
c/o Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 02555

Subject: Comments on NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States"

Dear Mr. Chilk:

On September 2, 1987, a notice in the Federal Register requested comments on the subject report by November 2, 1987. Combustion Engineering has reviewed NUREG-1251 and offers the following general comments for your consideration. Specific comments are included in the Enclosure to this letter.

The report's opening summary appropriately concludes that no immediate changes are required in the NRC's regulations regarding the design and operation of U.S. Light Water Reactors. This is in obvious recognition of the uniqueness of the Soviet design and its inherent weaknesses. The balance of the report, however, digresses into a detailed discussion of ongoing and recommended safety research and development in an apparent attempt to give additional impetus to these studies.

Many of the accident scenarios, which are identified in the report as safety concerns requiring additional evaluation, involve such a diversity of system and component failures of extremely low probability that their occurrence is highly improbable. In discussing these scenarios, the Staff has repeatedly admitted that they are not likely to lead to a Chernobyl-type event, and yet further analysis is recommended. Combustion Engineering recommends that the NRC reassess the safety concerns identified in NUREG-1251 and delete those which are only loosely tied to the Chernobyl accident in order to provide a more balanced and accurate assessment of its implications.

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Post Office Box 500
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(203) 688-1911
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Should you have any questions on these comments, please do not hesitate to call me or Mr. L. E. Philpot of my staff at (203) 285-5210.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Director
Nuclear Licensing

AES:ss
Enclosure

SPECIFIC COMMENTS ON NUREG 1251

1. Although the Introduction and Summary consistently indicate that "...regulatory provisions at nuclear plants in the United States are adequate...", there is insufficient reference in these sections to findings related to the operator's role in maintaining plant safety. Since the problem at Chernobyl involved a clear violation of regulations, Chapter 1 of the report examined the attitude toward regulatory compliance at U. S. reactors and concluded that the U. S. utility attitudes, the high level of NRC surveillance and the potential onerous penalties create an environment that makes a violation of regulations in the U. S. much less likely. The Introduction and Summary should also echo this conclusion. Summary

2. The suggestion that a dedicated high-level, onsite, nuclear-safety manager be designated and placed "in charge of safety" is unnecessary. The utility operating staff is responsible for plant operation and must, therefore, be in charge of plant safety as well. Adequate provisions already exist for 1) safety support via the site safety committee and 2) technical support via the shift technical advisor and the technical support center (during unusual situations). 1.6

3. Many adjectives such as conceivable, credible, extreme, low-probability, possible, etc. are used in describing the large number of postulated initiating events in Chapters 2 and 5. These relative terms tend to confuse the issue of what may really happen to a U. S. reactor or what must be considered in its design. In addition, the description of several of the postulated accident scenarios should be refined or eliminated because they are ill defined, difficult to understand and, in many cases, the connection to the Chernobyl accident is vague at best. 2.1, 2.2, 5.1

   The Industry Degraded Core Rulemaking program and related NRC programs extensively evaluated dominant severe accident initiators. If the concerns raised in NUREG-1251 have already been adequately addressed through these programs or have been found to be of sufficiently low probability, they should be dropped from further consideration.

4. One error has also been identified in Section 2.2 where it is stated on page 2-9 that Combustion Engineering has only documented the full power steamline break analysis in CESSAR. It should be noted that Appendix C to Chapter 15 of CESSAR documents the zero power steamline break analysis. 2.2

5. As the final barrier against the release of substantial amounts of radioactivity in the event of a severe accident, the containment system of U. S. Light Water Reactors is an extremely important safety system. This is also one of the prevailing design difficiencies that resulted in the large release during the Chernobyl accident. Chapter 3 should at least recognize this important design difference and address the inherent advantages of the U. S. containment concept when considering additional areas of study. 3

STATE OF ILLINOIS
DEPARTMENT OF NUCLEAR SAFETY
1035 OUTER PARK DRIVE
SPRINGFIELD 62704
(217) 785-9900

TERRY R. LASH
DIRECTOR

December 5, 1987

The Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Rules and Procedures Branch

Re: Draft for Comment, Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States, NUREG-1251, August, 1987.

The Illinois Department of Nuclear Safety hereby submits its comments on the above-referenced draft report. The Department requests that the appended comments be considered in preparing the final report on the Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States.

Thank you.

Sincerely,

Terry R. Lash
Director

TRL:bdb

Attachment

STATE OF ILLINOIS
DEPARTMENT OF NUCLEAR SAFETY
1035 OUTER PARK DRIVE
SPRINGFIELD 62704
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TERRY R. LASH
DIRECTOR

COMMENTS OF THE

ILLINOIS DEPARTMENT OF NUCLEAR SAFETY

ON

IMPLICATIONS OF THE ACCIDENT AT CHERNOBYL FOR SAFETY REGULATION OF COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES

DRAFT FOR COMMENT
NUREG-1251, AUGUST, 1987

December 5, 1987

COMMENTS OF THE

ILLINOIS DEPARTMENT OF NUCLEAR SAFETY

The Illinois Department of Nuclear Safety (IDNS) hereby submits its comments on the Draft for Comment, Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States, NUREG-1251. IDNS is the lead agency in Illinois for preparing emergency plans for and, in cooperation with the Illinois Emergency Services and Disaster Agency, coordinating emergency responses to accidents at nuclear power plants.

The accident involving Unit 4 at the Chernobyl Nuclear Power Station in the Soviet Union in April, 1986, was by far the worst in the history of nuclear power production. Never before has an industrial accident had the tremendous global impact or stirred such a widespread response. Nor has any previous nuclear reactor accident resulted in so many prompt fatalities or set the stage for so many latent detrimental health effects. Like the Three Mile Island accident, the Chernobyl accident jolted the public's acute awareness of the possibility of severe nuclear power plant accidents. The fiery issues of risk and necessity associated with the nuclear industry were rekindled on a global scale in the Chernobyl aftermath. People across the world were wondering, "Can it happen here?"

The Chernobyl accident has shown that a system thought to be safe can catastrophically fail when the mechanisms to maintain safety are bypassed. In light of the fact that humans are responsible for running

Summary

nuclear power plants and the risk of human error always exists, a disastrous reactor accident is also possible in the United States. This possibility must be acknowledged and addressed in the context of acquiring the knowledge that can be gained by studying the Chernobyl accident.

The Illinois Department of Nuclear Safety believes that all aspects of the Chernobyl accident must be thoroughly analyzed for the purpose of learning how to improve the level of safety at existing and future nuclear power plants. The lessons to be learned in preventive and mitigative actions for nuclear reactor accidents must be heeded by all regulators, utilities, and governing bodies in order to reduce further the chances of such horrific events. The Chernobyl accident is and will continue to provide data by which the temperamental issues in the nuclear industry can be addressed and actions in the nuclear arena can be measured.

Summary

Although the draft report contains a thorough description of the many ongoing activities in severe accident analysis and safety research, it presents implications in a way that, in effect, states that there is nothing new to learn from the Chernobyl accident. IDNS believes that NUREG-1251 does not,as it should, vigorously explore the Chernobyl accident to discover parallels between this accident and what might happen at U.S. nuclear power plants, and to translate these parallels into recommendations for action. Rather, the draft report explains all possible implications as though the purpose is simply to reinforce the NRC's previous conclusion in the Severe Reactor Accident Policy statement that ". . . existing plants pose no undue risk to the public." (50 Federal Register 32138; August 8, 1985.)

The draft report repeatedly emphasizes the differences between the Soviet and the U.S. reactor designs, concluding that such differences make drawing valid implications about U.S. reactors impossible. IDNS agrees that the design and operation of the RMBK reactor are vastly different from U.S. practices. Nonetheless a thorough, thoughtful analysis of the Chernobyl accident would, in our opinion, identify many areas where important implications can be drawn and significant improvements in the U.S. program can be made. Summary

ADMINISTRATIVE CONTROLS - The draft report states that ". . . administrative controls in place at Chernobyl were not effective in maintaining conditions within the safe operating envelope." (p. 1-1.) It further states that "After full implementation of symptom-based EOPs [Emergency Operating Procedures], administrative controls will be adequate to ensure that operations and other safety-related activities will be performed in accordance with approved written procedures." (p. 1-7, emphasis added.) The draft report seems to suggest, therefore, that an accident similar to the one at Chernobyl will not occur in the U.S., because of the better administrative controls. Another 1.1 implication, however, could also be that the current administrative controls are inadequate to assure safe operation with approved written procedures. In any event, there is no guarantee that procedures will always be followed, even with administrative controls.

The operating histories of commercial reactors in the United States contain numerous incidents where written operating procedures have been violated. For example, operators have been asleep at their posts, non-licensed personnel have performed tasks specifically reserved for licensed staff, and maintenance personnel have unknowingly removed safety systems from service. The leaking valve at Three Mile Island (TMI), for example, was not identified rapidly as the cause of the loss from the primary cooling system, because the temperature signal downstream of the valve had been reading high for some time. The NRC assessment emphasizes that administrative controls will be adequate once the symptom based EOPs are implemented. These EOPs were required soon after the TMI accident, over eight years ago, yet they still are incomplete. This fact is not emphasized, nor does the report indicate when such EOPs will be available. 1.1

IDNS agrees with the NRC ". . . that the benefits of a high-level, onsite nuclear-safety manager, who has no other responsibilities or duties, should be examined" (p. 1-2.) and that such an addition has merit. IDNS further believes that such an individual should be licensed as a Senior Reactor Operator (SRO) on the plant, should report directly to a corporate executive officer (e.g. Executive Vice President - Nuclear), rather than site management, and should concentrate only on matters of substantive safety significance. 1.6

The draft report emphasizes that the NRC examines and licenses operators on the technical details of their procedures and their responsibility for following procedural provisions. The draft report, however, fails to comment on the results of such examinations in a manner which would allow readers independently to draw their own conclusions. For example, the Reactor Operator/Senior Reactor Operator requalification process is normally administered by the facility and rarely results in failure by the applicant. On the other hand, requalifications administered by the NRC have, in some instances, resulted in failure rates exceeding 50% of the applicants tested. If the testing process 1, 1.1, 1.5

truly measures the applicants' abilities to operate the reactor safely, such a failure rate implies that the public has not been adequately protected since so many licensed operators did not pass the NRC requalification exams. Such a failure rate among experienced operators could indicate also that the testing process did not measure the operators' 1
abilities adequately. If the requalification examinations do not measure the ability to operate the reactor safely, the purpose of the licensing process has been subverted; that purpose being to determine each applicant's ability to operate a nuclear power reactor safely.

The draft report indicates that safety systems in U.S. plants are designed to "fail safe." It does not discuss the facts that certain systems have failed in other than a "safe" condition nor that poor 1.1
administrative procedures can defeat or bypass the capabilities of such systems.

The draft report also states that "Regulations under 10 CFR 50.36 require TS [Technical Specifications] which . . . mandate shift manning levels. Furthermore, maximum working hours have been specified to ensure that rested, qualified operators are available." (p. 1-18.) Yet in reality, operators have indicated that in some plants working double 1.5
shifts is common, specifically to maintain the required manning levels. The draft report does not discuss the danger that may exist if limits on the maximum working hours are waived to maintain the required manning levels.

In our view, the chapter on Administrative Controls and Operational Practices incorrectly concludes that little, if any, implications from 1
the Chernobyl accident require action because regulations and procedures are in place to prevent problems. The draft report mistakenly does not

address the fact that regulations and procedures are not self-enforcing, and thus that additional measures are required. IDNS urges the NRC to propose these additional measures in the final report.

DESIGN - The draft report states that ". . . the NRC review takes into consideration the radiological effect of an accident at one unit [of a multiple-unit site] on the other." (p. 2-12.) The draft report further states that, "In the event of a severe accident in one unit of a multiple-unit site, the control room operators are adequately protected by design features that will ensure a habitable environment." (p. 2-14.) 2.3
Although this is the intent, U.S. reactors are not, in fact, always constructed such that one reactor at a site cannot adversely affect another. The Zion facility in Illinois, for example, was recently fined for having ventilation systems improperly installed such that adequate ventilation was not assured.

Although the NRC has established and has begun implementation of regulations dealing with the problems of fire protection, the draft report fails to point out that the implementation of these regulations is very plant-specific. Contrary to the uniform implementation policies for other design requirements, the regulations dealing with fire protection are established by negotiating the plant-specific fire protection system 2.4
independently for each site and utility involved. This approach does not assure a consistent and uniform level of safety across the entire industry, and may result in substantial differences in risk between similar reactor designs at different utilities.

The draft report emphasizes the requirements for a fire brigade and the ability to ensure adequate manual firefighting capability for all areas of the plant containing structures, systems, or components important to safety. (p. 2-15.) The draft report does not indicate that such fire brigades are typically comprised of operating staff at the plant. During an off-shift, for example, the operating staff may be limited to the following:

* One Shift Engineer, one Shift Technical Advisor, and three Reactor Operators, all of whom are prohibited from leaving the control room.
* Two Shift Foremen
* One Radwaste Foreman
* Six Equipment Attendants
* One Equipment Operator

Such a staffing level provides only ten individuals outside the control room staff. If five of these individuals are diverted from their normal duties to fight fires on-site, the operating staff will be reduced by fifty percent. The final report should consider whether such a major reduction in operating staff during an emergency would be acceptable. 2.4

The draft report also indicates that ". . . no specific guidelines are provided on fighting fires in a highly radioactive area." (p. 2-17.) But the injuries of the Soviet firefighters clearly illustrates the need for training to cope with such conditions.

In spite of these deficiencies, the draft report surprisingly concludes that ". . . the programs provide an adequate level of defense in depth for all anticipated events." (p. 2-19.) IDNS, however, believes that NRC should require improved firefighting capabilities as a result of the experience at Chernobyl. It is interesting to note in this regard that TVA has recently announced plans for installing a trained fire department at each of their plants. The final report should contain a list of recommendations for assuring that U.S. nuclear plants have improved firefighting capabilities. 2.4

CONTAINMENT - IDNS agrees with the NRC that containment venting is a primary issue of concern and that no requirement for venting capability should be adopted until the subject has been thoroughly analyzed. Containment integrity and containment venting designs and procedures, however, should be of primary importance in reactor safety research. The final report must emphasize this significant reactor safety issue. On-going and future research on the effectiveness of filtered vents may provide the basis upon which the NRC can include a containment performance criterion in a revised safety goal policy statement. The actions necessary to coordinate this research with NRC policy development should be a matter of high priority within the NRC. 3.2

The draft report indicates that for BWR's ". . . it is up to individual utilities to provide for filtered venting as a last resort to prevent gross containment failure on overpressure, and to prevent an excessive buildup of hydrogen in the containment, for accident conditions more severe than were considered in the original containment design." (p. 3-4.) The NRC should elaborate on this recommendation in the final report. For instance, it is questionable whether the Standby Gas Treatment System used in containment venting procedures could remain operable under the extreme pressures threatening a breach of containment. In this respect, containment venting may be preferable for overpressures less than the containment design pressure in order to ensure operability of the systems included in the venting procedure.

EMERGENCY PLANNING - Among the many important lessons to be learned 4
from the Chernobyl accident, IDNS believes those in the area of emergency

response are likely to be the most fruitful. Yet this portion of the draft report is probably the weakest. Chapter 4 is replete with references to the differences between the Soviet response and emergency planning in the U.S., but only where such differences were seen by the NRC as adversely impacting on the Soviet emergency response capabilities.

The chapter begins, for example, by stating that it is difficult to draw a link between the Chernobyl accident and emergency planning implications for U.S. reactors for four reasons.

(a) "First, there is a substantial difference in the emergency planning base.

(b) Second, the specifics of the Chernobyl release are unique to the RMBK design.

(c) Third, some aspects of the Chernobyl evacuation defy 4
comparison with similar aspects at U.S. plants because of economic and societal differences.

(d) Finally, it should be recognized that issues such as off-site decontamination and long-term relocation raise matters whose scope and timing extend beyond the 'emergency' actions for which detailed advance planning is beneficial and appropriate. As such, these matters may fall outside the traditional scope of emergency planning for U.S. plants." (p. 4-1.)

There are serious problems with each of these statements that could lead to incorrect conclusions about the status of U.S. nuclear emergency response capabilities.

First, differences in the emergency planning "base" have little to do with what we could learn from the Chernobyl accident. It is apparent that the Soviets developed and then acted on some emergency guides as the accident was occurring. IDNS, therefore, believed that it would be extremely worthwhile to examine the protective actions taken by the Soviets to see what improvements could be gleaned for our own system. The fact that the Soviets may have had no comparable site-specific emergency plans is not critical.

Second, a comparison of the Soviet source term with the PWR-2 accident in WASH-1400 indicates a remarkable similarity. Even though the reactor designs are not comparable, and the probability of a PWR-2 accident occurring may be significantly lower at U.S. reactors than at Chernobyl, it is possible that a source term similar to that at Chernobyl could occur. Indeed, in preparing for emergency response, we must assume worst case conditions to be able to respond adequately. The statement in
the draft report concerning the Chernobyl release in relation to emer- 4
gency planning is extremely misleading.

Third, although significant societal differences exist, the Chernobyl accident points out that temporary or permanent relocation of many people may be necessary. This problem should be re-addressed in emergency response planning in light of the Soviet experience.

Fourth, it is correct that many matters related to emergency response at Chernobyl fall outside the "traditional scope" of emergency planning. That is exactly the important lesson to be learned. Thus, should we not re-examine our concept of emergency response based on this experience? The traditional scope may need to be expanded considerably.

The draft report also states that ". . . the Soviets had to assemble 4000 buses and trucks for the Chernobyl evacuation, whereas, in the United States most people have access to private transportation . . .."

(p. 4-1.) Seemingly, No credit is given the Soviets for their demonstrated capability of assembling such a fleet and efficiently evacuating the people of Chernobyl and Pripyat with no indications of panic. Although the reference is made that a U.S. evacuation would rely primarily on private transportation, no evidence is given that such reliance is preferable or could even approach the Soviet efficiency. Are there no lessons to be learned regarding psychological trauma? What are the effects of mass evacuation, possibly for many months or years? The draft report unfortunately is silent on these important questions. 4

The draft report, moreover, goes to great length to reject any thought concerning the need to expand the ten-mile plume Emergency Planning Zone. IDNS questions the validity, however, of the supporting rationale. Indeed, the draft report seems to be internally inconsistent on an important point pertinent to this issue. 4.1 In particular, the argument that the ". . . source term is specific to the RMBK design and U.S. source terms are expected to be lower. . ." (p. 4-4.) is difficult to reconcile with statements that "The total quantity of fission products released from Chernobyl was large and is considered to be comparable with the quantities predicted to be released for the worst cases . . . studied for U.S. LWRs using . . . the most recent source term methodology." (p. 5-4.) 4.1, 5.1

Although IDNS is not now advocating expansion of the Emergency Planning Zone, the possibility of protective actions being required beyond a ten mile radius should be emphasized more, both in emergency planning documents and exercises, in light of experience at Chernobyl and recently completed studies such as NUREG-1150. Especially in terms of planning for nuclear emergencies, worse case conditions should usually be 4.1

assumed.

The draft report explains that thousands of physicians and health care personnel were summoned to treat the Chernobyl casualties. (p. 4-4.) But, although the draft report acknowledges the American Medical Association's activities addressing the implications of the Chernobyl accident and expounds the Commission's policy on "Emergency Planning - Medical Services," it fails to address the feasibility and reality of summoning thousands of health care officials to a potential "disaster zone" within a matter of hours. Nor does the draft report address the stress and anxiety that will propagate throughout the general public in the event of a severe or even minor nuclear accident.

Informed physician participation and preplanning of medical management of radiological accidents is long overdue. The accident at Three Mile Island fortunately resulted in no acute radiation exposures, but it did unveil an uninformed and ill-prepared medical community. Gordon K. MacLeod, M.D., Pennsylvania Secretary of Health during TMI, stated that ". . . the Department of Health had little or no capability to deal with this extraordinary serious emergency." Dr. MacLeod also noted that ". . . only one of the seven Harrisburg area hospitals had been prepared to handle some four to six nuclear accident victims." This is a far cry from the 203 plant and response personnel suffering from acute radiation sickness and requiring immediate medical attention at Chernobyl. 4.2

George Jackson of the American College of Physicians noted the inconsistent and often inaccurate reporting of the TMI accident resulted in ". . . so many conflicting reports in the press, [that] patients trusted neither industry or government. Thus, practicing physicians were deluged with telephone calls from their patients requesting interpretation of the accident's significance in terms of their personal health." Jackson further stated, "Many of these physicians simply did not have the knowledge or expertise to offer advice on nuclear matters."

Although TMI had very minute radiological impact, a 1981 study by the New York Academy of Sciences for the President's Commission of the Accident at Three Mile Island concluded that the episode " . . . had substantial immediate psychological effects on the people living in the area" and ". . . a lasting impact on the population of the area in terms of distrust of authorities with respect to nuclear power." The lack of physician involvement and public education in nuclear issues such as radiological emergency planning and response only serves to encourage such anxieties and will feed the psychological trauma of an uninformed 4.2
public that may endure a radiological accident.

The NRC's post-TMI efforts to involve the medical community in emergency planning and response have been a step in the right direction. The implication of the Soviet medical response to Chernobyl is that the NRC's efforts have been meager though. The NRC's "Emergency Planning - Medical Services" policy outlines the tasks of "emergency response personnel," but it does not detail the medical community's involvement in radiological emergency preparedness. The Chernobyl accident demonstrated the scope of response needed to manage a radiological accident. Literally thousands of persons from the medical community were summoned to treat casualties. Professor Marvin Goldman of the University of California at Davis commented on the Chernobyl accident, "The cloud of radioactive material that was ejected from the reactor blanketed much of the Northern Hemisphere and precipitated a fallout of small radiation doses and large societal apprehensions." The final report must address this reality and discuss how to involve the medical community more extensively in emergency response planning in the United States.

The draft report also describes the distribution of Potassium Iodide (KI) to school children within six hours of the accident, to the entire population of Pripyat the next morning, and ultimately to the population within 30-kilometers of the site. The Polish government also distributed vast quantities of KI to the general public. IDNS understands that the Soviets dispensed approximately five million doses and the Polish approximately six million doses. No credit is given the Soviets for the ability to dispense KI with such timeliness, nor is there any objective assessment of the effects of this action. The only statement is the 4.2
following: "The Soviets report no serious adverse reactions to KI." (p. 4-4.) What about the minor effects? With what frequency did they occur? Were there problems with overdosage? How was distribution handled? What planning and stockpiling of KI was in place before the accident? Where were these stockpiles maintained? Was there any indication that the public perceived KI as a universal blocking agent rather than limited to one organ (thyroid) and one radioactive species (iodine)? Are there data quantifying the effectiveness of KI in the USSR and Poland? Will the effectiveness of KI be considered in future epidemiological studies carried out in the USSR?

Although the draft report indicates that the decision to provide KI should be made by individual states rather than the Federal Government, the actions of the Soviet and Polish Governments are the only known large scale use of this drug by the general public during a radiation emergency. Thus, this experience has great implications to the U.S. Although the U.S. has a policy on the use of KI, it seems worthwhile to review the experience at Chernobyl to see whether it either supports or suggests modifications of that policy. 4.2

The draft report states that "the adequacy of the Federal guidance (Protective Action Guides for ingestion pathways) cannot be tested in the light of the Chernobyl accident because the specific Chernobyl releases are unique to the RMBK design." (p. 4-8.) IDNS does not accept this contention. First, there is no Federal guidance regarding the ingestion of contaminated drinking water, only food. The Federal guidance, in other words, is severely lacking. Second, the implication is that Protective Action Guides (PAGs) are for each specific type of radioactive material (radionuclide). Primary PAGs are based on dose. Only the secondary PAGs identify the specific nuclides, indicating the quantities of each nuclide that equals the primary PAG. Therefore, irrespective of any differences in source term, there is no reason the primary PAGs cannot be compared directly with those used by the Soviets and European countries. 4.3

Even a cursory review of the protective measures taken during Chernobyl would conclude that there are several important lessons to be learned. One key recommendation is that protective action guides should be re-examined for their consistency on an interagency and international basis. Inconsistencies in protective action guides between and within countries caused considerable confusion during the Chernobyl accident. A detailed analysis should be made of U.S. protective measures. At present, these guidelines are derived and applied by several different agencies within the U.S. How can we ensure that primary and secondary protective action guides are determined on a uniform basis with 4.1, 4.3

consistent methodologies being applied throughout? Also, how do we make certain that protective action guides are compatible within States, between States, and between the U.S. and other countries (particularly neighboring Mexico and Canada)? The final report should explain how the NRC is going to address these questions. 4.1, 4.3

The draft report also states that, "To date, decontamination and relocation planning has generally assumed that relocation would be short term and re-entry would be feasible." (p. 4-9.) As evidenced by Chernobyl, this policy is not prudent and should be re-examined. In addition, the effort to establish decontamination criteria should be given higher priority since it appears the work has been ongoing for many years and no formal recommendations have been published. 4.4

Another aspect of emergency planning totally ignored by the draft report is the refusal of Soviet farmers to evacuate without their livestock. IDNS is unaware of any references in nuclear accident planning documents that address large scale evacuation of farmstock in the U.S. If the Soviet farmers would not evacuate without their animals, what does the NRC expect from American farmers? The implications of this additional complexity are enormous, especially for sites in major farming areas. 4.1

The draft report also fails to address the issue of public acceptance of official protective action recommendations. What would have been the result if Pripyat had evacuated spontaneously rather than accepting the recommendation to shelter-in-place? Should we expect the same behavior from the American public? Some emergency response experts believe that spontaneous evacuation will occur, based upon their experience in a wide range of other disasters. The latest NRC guidelines

(NUREG-1210) also stress evacuation, almost to the exclusion of any other protective measure. The final report should address the issue of to what extent protective action recommendations would be followed by the U.S. public.

The conclusion that, "The Chernobyl accident and the Soviet response do not reveal any apparent deficiency in U.S. plans and preparedness . . ." (p. 4-4.) is extremely hasty based on the superficial assessment in the draft report. The draft report has not adequately examined Chernobyl in light of what the U.S. could learn. 4.1

<u>SEVERE ACCIDENT PHENOMENA</u> - The draft report states that "the total quantity of fission products released from Chernobyl was large and is considered to be comparable with the quantities predicted to be released for the worst cases . . . studied for U.S. Light Water Reactors using WASH-1400 as well as the most recent source term methodology . . . the Chernobyl release represents a near worst case in terms of the risks of nuclear energy." (p. 5-4.) Although the draft report indicates that the Chernobyl release can be matched by only the worst sequences from U.S. reactors, it fails to state where the Chernobyl sequence falls within the range of plausible accidents for Soviet reactor designs. The final report should clearly state that the accident sequence at Chernobyl also fell at the extreme edge of the Soviet spectrum. 5.1

The most important conclusion of the source term analysis is that methodologies must be improved to derive source term estimates from environmental measurements. Regardless of where an accident may happen, data characterizing emissions from the nuclear reactor may not be available because of inaccessibility of the area or a lack of information communicated by authorities at the scene. Transformation of environ-

mental measurements into source terms may be the only mechanism available to quantify releases. Based on a review of source term estimates for Chernobyl from an international spectrum of agencies, it is apparent that we currently lack a clear and consistent rationale for what environmental measurements should be taken, how environmental measurements should be used to derive source terms, and which format should be used in reporting 5.1
source term information. Considering the uncertainties in the source term for the TMI accident and the difficulties that arose during Chernobyl, the NRC must consider methods for quantifying source terms following a nuclear accident. Initiation of such an effort should be an important recommendation in the final report.

Lawrence Livermore National Laboratory

October 27, 1987

Mr. Eric Beckjord, Director
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Beckjord:

With regard to your letter dated September 8, 1987 to Roger Batzel, Director, Lawrence Livermore National Laboratory, I have been asked to respond on Dr. Batzel's behalf with our comments on NUREG-1251, **"Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States"**.

We have compiled our comments from within various areas of the Laboratory: Nuclear Systems Safety Program, Electronics Engineering Department, Mechanical Engineering Department, Hazards Control Department, Environmental Sciences Division, G Division, M Division and Z Division whose work relate to nuclear power safety.

As you will see from our comments, we generally agree with the conclusions reached by NRC in NUREG-1251. We feel, however, that clarification or amplification is needed in some sections of the report and have submitted our comments accordingly.

If we can be of any further assistance to the NRC, please let me know.

Very truly yours,

R O Godwin

R. O. Godwin
Associate Director for
Plant & Technical Services

ROG:ka
Enclosure

cc: R. Batzel

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COMMENTS ON NUREG-1251

The U.S. Nuclear Regulatory Commission report, **"Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States"**, NUREG-1251, discusses implications of lessons learned from the Chernobyl accident on regulation of U.S. nuclear power plants. The general conclusions of the document are that there are generic lessons to be learned but that no changes in regulations are needed due to the substantial differences in the design, safety features and operation of U.S. plants as compared to those in the U.S.S.R.

Given these general conclusions, further consideration of certain specific areas is recommended by the report. These include: 1) administrative controls over reactor regulation, 2) reactivity accidents, 3) accidents at low or zero power, 4) multi-unit protection, 5) fires, 6) containment, 7) emergency planning, 8) severe accident phenomena, and 9) graphite-moderated reactors.

After reviewing the report, we generally concur with the conclusions reached but would like to comment on certain portions of the material presented where clarification or amplification appears warrented. Our comments will mainly be confined to Chapter 1 (ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICE), Chapter 2 (DESIGN), Chapter 5 (SEVERE-ACCIDENT PHENOMENA), and Chapter 6 (GRAPHITE-MODERATED REACTORS). We have no comments on Chapter 3 (CONTAINMENT) and generally support the conclusions reached in Chapter 4 (EMERGENCY PLANNING).

Comments on Chapter 1: Administrative Controls over Reactor Regulation

Our comments on Chapter 1 are in two areas: administrative controls and management systems.

The conclusions reached concerning current regulations relating to administrative control (Section 1.1.4) suggests placing increased emphasis on the use of symptom-based emergency operating procedures (EOP) in U.S. plants. Although this is a worthwhile suggestion, implementation of enhanced EOPs needs to be coupled with efforts to provide the operators with better tools to diagnose the accident so that the right EOP is used. It is interesting to note that in both the Chernobyl and TMI-2 events, the accident progression was strongly influenced by operator error and that these errors, in both cases, were founded in an erroneous "cognitive process". At TMI, the operators shut down the high pressure safety injection system because they did not realize that the primary loop was operating under two-phase thermal-hydraulic conditions and therefore was not overfilled with coolant as the pressurizer level indicator had led them to believe. At Chernobyl, the operators did not understand that the physical behavior of their plant at 200 MWth, with higher than normal circulation flow, would be very different from that expected in the 700-1000 MWth range which had been defined for the test being run. The true lesson from this is that administrative controls cannot be counted on to prevent operator errors of commission when the operators fail to understand the physical reality upon which these controls and the associated operational procedures are founded.

The question, therefore, needes to be asked as to how to provide the operators with better tools to understand and interpret correctly the various conditions which may occur in a plant so that symptom or based EOPs can be used effectively. One solution would be to move toward the gradual development of computer operator aids in the control rooms based on new software technologies, coupled with the vast body of knowledge on operational transient and potential accident sequences available today. These aids should not be seen as alternatives to traditional operator training and practice but as on-line trainers and cognitive process helpers to help the operators absorb and interpret information for proper use of their EOPs. Simply stating that symptom-based EOPs should be emphasized, as is done in Section 1.1.4, may not be enough. To make such procedures effective, stronger action is needed, such as implementation of computer based aids.

In Section 1.6.2, a discussion is given of work in progress concerning regulatory monitoring of management systems used at U.S. nuclear power plants. It is stated in this section that NRC has terminated work intended to provide the technical basis for formulating new requirements in the field of licensee management and organization and instead will be relying on the use of performance indicators, appraisal results and individual attention to problem plants, to regulate management performance. In essence, it appears that NRC will attempt to treat the perceived symptoms of "management found wanting" rather than the direct cause, without first defining unsatisfactory management performance. A measurable definition of management safety goals and specific fundamental management requirements and tasks would allow for a more precise assessment of management systems. One would not have to specify how or in what fashion these requirements and tasks will be executed – only that they be executed.

Comments on Chapter 2: Design

Our comments on Chapter 2 apply to Section 2.1 where reactivity accidents in U.S. light-water reactors (LWR) and Soviet RBMK reactors (like the Chernobyl plant) are compared. The third paragraph in this section, starting in the middle of Page 2-2, is difficult to follow and needs clarification. In particular, the sentence beginning:

"A critical reactor generally ...."

makes little sense, although we assume it to imply that overmoderation is required to maintain criticality in a LWR at low-power and high burnup. The next sentence, beginning:

"In a PWR, boron ...."

states that a decrease in boron density results in a decrease in possibility of a positive moderator coefficient. We assume this refers to the derivative of the void reactivity coefficient with respect to moderator boron concentration. Clarification of this sentence would help.

The stated conditions where a positive moderator void coefficient may be present, for both PWRs and BWRs, includes minimally inserted control rods. This condition was considered to be a contributing factor to the Chernobyl accident, as this is the least effective location for response. If the superior response time and control rod worth of the LWR's, discussed on page 2-3, are sufficient to override this concern, it could be explicitly stated at this point.

The phrase *"normal ... conditions"* is used several times on pages 2-2 and 2-3 in assertions about the potential for large reactivity insertion events in LWR's. Unless the word *"normal"* is defined to include not only conditions that are expected in everyday operation, but also those which may occur with some reasonable probability, these assertions are not relevant when assessing the possibility of a Chernobyl-type event in a LWR.

In the first paragraph on 2-3, the comparison of void coefficients is referred to the prompt criticality conditions. The values required to achieve a prompt critical condition for both the RBMK and US-PWR reactors should be quoted.

The second paragraph of 2-3 discusses the operation of the control rods in the RBMK, and comments on the timing and implications. We see no mention of the so-called *"positive scram"* scenario. Although this scenario may be considered controversial by some, there are technical implications of such design conditions which appear to merit review of the type purportedly dealt with in this document.

There is no discussion in this section (although there is mention of it in Chapter 5) on the impact of the spatially skewed core power distribution on the reactivity state of the Chernobyl reactor. As that has been assessed as having significant impact on the initiation and development of the accident, there should be some mention of U.S. reactor experience with, analysis and implications of such highly skewed power distributions.

Comments on Chapter 5: Severe-Accident Phenomena

Our comments on Chapter 5 relate to the source term discussion given in Section 5.1. On Page 5-1 the report states that the source term from the Chernobyl accident reached a height of 1000-2000 meters. We agree, if the material over Eastern and Western Europe is the only consideration. We believe, however, that a large amount of material also rose to about 7000 meters, most likely as a result of meteorological convective activity. This is the material that was transported over Siberia, Japan and into the U.S. Most reports neglect this part of the cloud.

Also in Section 5.1 some hypoitheses are given about the increased release rate on the sixth day and the decreased release rate on the tenth day following the accident. It is not clear from the report whether there are any plans or possibilities for further studies to confirm one or more of these proposed scenarios. A reliable assessment of the release mechanisms which occurred at Chernobyl will be necessary in order to address the two major issues given at the beginning of the section.

Comments on Chapter 6: Graphite-Moderated Reactors

Our comments on Chapter 6 relate to Section 6.2.2 concerning the design of graphite-moderated reactors in the U.S. We generally agree with the conclusion of that section, that the use of helium coolant, the overall negative reactivity coefficient, completely diverse alternative shutdown and cooling systems and protection offered by the prestressed concrete reactor vessel against fires, explosions and fission-product release remove commercial power reactors of this type, such as Fort St. Vrain, from any vulnerabilities characteristic of the RBMK design. We, however, hope that new commercial reactor design of this type, such as the modular high-temperature gas-cooled reactor (MHTGR), will be designed to have the Fort St. Vrain characteristics and in particular, be designed to prevent chimney fires. Flooding of the reactor pressure vessel as a backup defense against a chimeny fire at Fort St. Vrain seems extreme and we would hope that the MHTGR is designed to prevent chimney fires without resorting to flooding.

NewYorkPower Authority

John C. Brons

November 19, 1987
JPN-87-058
IPN-87-054

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555
Attn: Chief, Rules and Procedures Branch
Division of Rules and Records
Office of Administration

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Comments on Draft NUREG-1251 - Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States

Reference: Notice of Availability of Draft NUREG-1251, Federal Register, September 2, 1987 (52 FR 33304).

Dear Sir:

The Power Authority has reviewed Draft NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," which was published for comment in August 1987. The Authority's detailed comments are provided in the attachment to this letter.

The Authority agrees with the NRC Staff's conclusion that the design of domestic light water reactors preclude an accident like the one at Chernobyl. The Authority also agrees with the Staff's reconfirmation of the Commission's finding on severe accidents; namely, that existing plants provide no undue risk to the health and safety of the public and that no immediate regulatory action is required to address severe accident risk. These Staff findings are the basis for the Authority's comments in the attachment that the scope of severe accident studies should not be expanded at this time. The technical evaluation of NUREG-1150, "Reactor Risk Reference Document," should be completed before undertaking the review of even less probable events accidents.

Summary

2.1, 2.2

Although the Authority substantially agrees with the conclusions on NUREG-1251, we strongly disagree with its recommendations concerning personnel qualifications issues. The NUREG recommends the consideration of requirements for a high-level onsite nuclear safety

Summary

1.6

manager and degrees for senior reactor operators. The Authority is opposed to both of these because of the possibility that they will reduce safety rather than improve it. The safety manager concept could result in a decreased emphasis on individual responsibility for safety. The requirement for a degree for senior operators would not increase operating and management expertise on shift since these cannot be acquired in a degree program. Instead, the degree requirement may impede the career advancement of highly qualified and experienced individuals and thus negatively affect safety. 1.6

The NUREG also concluded that emergency planning in the areas of emergency planning zone size, medical services and ingestion pathway measures are adequate. Based on the Authority's extensive involvement in emergency planning, we strongly agree with this conclusion. 4

Should you or your staff have any questions regarding the Authority's comments, please contact Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

for John C. Brons
Executive Vice President
Nuclear Generation

cc: Office of the Resident Inspector
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The Power Authority's detailed comments on NUREG-1251 are provided below by, arranged by subject. The chapter and section numbers referred to in the comments correspond to those used in the NUREG.

High-level Onsite Nuclear Safety Manager

Section 1.6.3 states that there is no single individual onsite who is responsible for nuclear safety and nothing else; and, that safety is everyone's responsibility, concurrent with their other duties. The NRC Staff recommends the study of the benefits of having a high-level, onsite nuclear-safety manager, who has no other responsibilities or duties. The NUREG does not describe the duties of this individual or how the position would fit into the onsite organization.

The Authority considers the fact that safety is considered everyone's responsibility to be a strong point in the commercial nuclear power industry. The Authority believes that there is no substitute for an attitude that requires each individual to be cognizant of, and responsible for, the safety of all of his activities. The establishment of a new position responsible for nuclear safety, and only nuclear safety, presents a very real risk of diluting the existing attitude toward safety. This possibility is recognized in the NUREG itself. 1.6

The Authority cannot foresee how the safety manager's responsibilities can be separated from those of the utility's onsite senior manager who is responsible for operating the plant. The Authority also cannot see the wisdom of trying to divide this authority. Separating the responsibility for safety from the responsibility for generating power suggests that there is a conflict between the two. The Authority does not believe that this is the case or should be the case. The Authority believes that safety and power generation go hand-in-hand.

Nuclear Safety Evaluations Performed in Accordance with 10 CFR 50.59

Sections 1.1.1, 1.2, 1.2.1 and 1.2.2 discuss the adequacy of administrative controls to assure that modifications to the plant, changes to procedures, and tests and experiments are carried out safely. These sections make frequent reference to 10 CFR 50.59 which requires that a safety review of changes be conducted prior to their implementation. 10 CFR 50.59 provides criteria for the following: evaluating whether the change may be implemented without affecting the safety of the plant; documenting the evaluation; and, determining whether NRC approval is required prior to implementing the change. The NUREG states that in general, utility activities governed by 10 CFR 50.59 are carried out satisfactorily but that some recent 10 CFR 50.59 evaluations have been inconsistent in depth and the quality of documentation. In addition, there have been recent violations of the regulation which have resulted in enforcement penalties.

The Authority agrees that the safety evaluation process is basically sound but that it can be improved. The NUREG cites work under way by the Atomic Industrial Forum and the Nuclear Safety Analysis Center to develop criteria and guidelines for utilities to use in performing 10 CFR 50.59 reviews. The Authority strongly endorses this effort which is now being conducted under the auspices of the Nuclear Utility Management and Resources Council (NUMARC). This effort will help to standardize 10 CFR 50.59 reviews by doing the following: clarifying the criteria for reviews; clarifying the requirements for NRC review and approval prior to implementation; and, establishing guidelines for the documentation of the reviews. The Authority sees no need to revise 10 CFR 50.59. Instead, the guidelines should be reviewed by the NRC Staff and, if the guidelines are acceptable, they should be incorporated into an I & E Manual Chapter.

Technical Specifications Improvement Program

Sections 1.1.2, 1.4.2 and 1.4.4 refer to the Technical Specification Improvement Program (TSIP). Priority should be given to the effort to review technical specifications to assure that they have the following attributes: they are clearly written; their content is limited to safety related requirements; and, they are closely related to he reactor operator's job. The Authority strongly endorses this effort and is fully participating in industry activities in this area. The Authority has previously commented on the NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Plants and supported the adoption of a final policy statement (Reference 1).

Technical Specification Surveillance Requirements

Section 1.2.4 identifies the need to evaluate NRC testing requirements as contained in the technical specification surveillance requirements. The Authority agrees that this evaluation needs to be done in order to assure that the positive and negative impacts on safety have been correctly evaluated and reflected in the specifications. The Authority considers the TSIP to be the appropriate vehicle for accomplishing this evaluation. The Authority requests that the NRC take whatever steps are necessary to expedite completion of the TSIP, including the priority allocation of NRC resources to complete the program.

Licensed Operator Requalification

Section 1.5.2 of NUREG-1251 briefly describes the NRC's program for the requalification of licensed operators and states that it has resulted in improvements in operator knowledge and performance. The Authority believes that the impact of this program as implemented by the Staff is more negative than positive. The Authority agrees with the recent NRC decision to suspend this program due to its negative impact on safety (see the NRC letter to All Power Reactor Licensees dated September 18, 1987).

Degree Requirement for Senior Reactor Operators

Section 1.5.2 also refers to the NRC's proposed rule which would require senior reactor operators to have a bachelors degree in engineering or science. The Authority is strongly opposed to this requirement and has commented on this proposed rule at length (Reference 2). Those comments pointed out that the degree would not increase the operating and accident management expertise on shift since this is gained only by utility provided training and actual operating experience. In addition, the degree requirement will have a negative impact on shift crews because it presents a formidable stumbling block on the career path of many highly qualified individuals.

Severe Accident Policy

The NRC's severe accident policy and the Industry Degraded Core Rulemaking Program (IDCOR) are discussed in Section 1.7.2. The NUREG restates the Commission conclusion that the existing plants present "no undue risk to the public" and that their is no need for immediate 1.7
regulatory action as a result of the severe accident risk presented by the plants. The Authority agrees with the NRC's conclusion. The NRC published NUREG-1150, "Reactor Risk Reference Document," in February 1987. The Authority has done extensive research in this area and has provided detailed evaluations of, and comments on, the NUREG (Reference 3).

Nuclear Design

The NUREG stated (Chapter 2) that the NRC Staff found that the nuclear design of domestic reactors preclude a Chernobyl-type superprompt critical reactivity excursion. The Authority agrees with this Staff conclusion.

However, the NUREG goes on to discuss design basis reactivity insertion events which have already been evaluated for domestic reactors. The discussion recommends expansion of these accident sequences to include multiple diverse failures or errors which have an extremely low probability of failure. The NUREG concludes that "conceivable reactivity accidents are not likely to lead to a 2.1,
Chernobyl-type event." 2.2

It then goes on to recommend probability studies of these very low probability events. The Authority believes that the ongoing efforts to evaluate severe accidents are adequate and that it is unnecessary to continue to expand the events to be evaluated until the current work is complete, has been subjected to a thorough technical review, and has been accepted by the technical community. Rather then begin new studies, both the NRC and the nuclear industry should concentrate their efforts on completing the review of NUREG-1150 and related work.

Containment

Chapter 3 focuses on the containment as a barrier to the release of fission products during an accident. The Staff again references the conclusion of the Severe Accident Policy Statement that existing plants pose no undue risk to the public. The Staff also refers to recent studies that show that domestic containments can withstand pressures as high as 2 to 3 times the design pressure. The Authority agrees with these positive findings, however this chapter should be expanded to include a description of the Chernobyl "containment vessel" since it is so much different then those in the United States. This would make the NRC conclusion concerning containments more understandable to readers who are not familiar with evaluations of domestic containments.

Emergency Planning

NUREG-1251 reviewed the following four areas of emergency planning in
light of the Chernobyl accident: emergency planning zone size;
medical services; ingestion pathway measures; and decontamination and 4
relocation. The NRC Staff concluded that emergency planning in the
first three categories is adequate and that Soviet data on the last
should be reviewed as it becomes available. The Authority concurs
with the Staff's conclusions in these areas.

References

1. NYPA letter, John C. Brons to the NRC, dated March 27, 1987 (JPN-87-016/IPN-87-018), transmitting Authority comments on the NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Plants

2. NYPA letter, John C. Brons to the Secretary of the Commission, dated September 24, 1986 (JPN-86-43), transmitting Authority comments on the Advance Notice of Proposed Rulemaking for Degree Requirement for Senior Operators at Nuclear Power Plants.

3. NYPA letter, John C. Brons to the NRC, dated September 28, 1987 (JPN-87-051/IPN-87-045), transmitting Authority comments on Draft NUREG-1150 - "Reactor Risk Reference Document."

North Carolina Department of Crime Control and Public Safety

James G. Martin, Governor
Joseph W. Dean, Secretary

Division of Emergency Management
116 W. Jones St., Raleigh, N.C. 27611
(919) 733-3867

October 29, 1987

Rules and Procedures Branch
Division of Rules and Records
Office of Administration
Room 4000 MNBB
Washington, DC 20555

This is a reply to your request for comments (Federal Register/Vol. 52, No. 170) on NUREG-1251 (Draft), "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States."

The following comments are provided:

1. The conclusions should be written in an Action Item format and a Cognizant Federal Agency, with an action due date, should be assigned to each action item.

2. This report did not substantiate General Conclusion Number (2) that some aspects of Emergency Planning need to be reexamined. Recommend that General Conclusion Number (2) (page 3) be rewritten to delete reference to Emergency Planning. **Summary, 4**

3. This report did not demonstrate the need to reexamine current planning bases for ingestion pathway planning. Recommend that Conclusions about Specific Areas Number (7) (page 5-6) be rewritten to delete reference to the reexamination of Ingestion Pathway Planning Bases. **Summary, 4.3**

Request eight copies of the final NUREG-1251 publication. I understand there is no charge to State Government agencies.

Sincerely,

Joseph F. Myers

Joseph F. Myers
Director

512 N. Salisbury Street • P. O. Box 27687 • Raleigh, N. C. 27611-7687

Nuclear Information and Resource Service
1616 P Street, N.W., Suite 160, Washington, D.C. 20036 (202) 328-0002

November 4, 1987

Secretary
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Chilk,

Enclosed are the comments of the Nuclear Information and Resource Service on Nureg-1251 "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States". Though the comment period expired two days ago, we hope you will consider our remarks in the final report.

Please forward our comments on the appropriate NRC department.

Thank you very much.

Sincerely,

Stephanie Murphy

Stephanie Murphy

Nuclear Information and Resource Service
1616 P Street, N.W., Suite 160, Washington, D.C. 20036 (202) 328-0002

November 4, 1987

COMMENTS OF THE NUCLEAR INFORMATION AND RESOURCE SERVICE ON NUREG-1251 "IMPLICATIONS OF THE ACCIDENT AT CHERNOBYL FOR SAFETY REGULATION OF COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES"

Immediately after the Chernobyl accident, then Chair of the NRC, Nunzio Palladino, warned: "We must not become complacent because of the differences in (U.S. and Soviet) reactors. There may be generic implications, We should look to some of these generic areas." However, the draft of NUREG-1251 makes obvious that NRC intends to ignore this advice. The first general conclusion of the report states "No immediate changes are needed in the NRC's regulations regarding the design or operation of U.S. commercial nuclear reactors." As if in over-reaction to the many regulatory changes brought about by the Three Mile Island accident, the NRC apparently is opting for the opposite extreme--learn nothing from the Chernobyl disaster.

But the draft does claim to learn lessons from Chernobyl--"The most important lesson is that it reminds us of the continuing importance of safe design in both concept and implementation; of operational controls, of competence and motivation of plant management and operating staff to operate in strict compliance with controls; and of backup features of defense in depth against potential accidents."

NIRS' comments will focus on NRC's evaluation and treatment of three issues in NUREG-1251: fire protection standards, containment design and regulations to prevent hydrogen explosions, and the lessons to be gleaned from Chernobyl's emergency evacuation experience. These examples show that regulatory changes should be made as a result of the Chernobyl accident, and further, that NRC is failing to properly regulate even those areas it deemed "most important lessons" from Chernobyl, mentioned above. 2.4, 3, 4, 5.3

EMERGENCY PLANNING

Emergency planning is perhaps the regulatory area deserving the most rapid and obvious changes. The NUREG fails to examine the details of the evacuation that occured at Chernobyl to weigh the benefits of the Soviets' method of evacuation and determine areas in U.S. regulations that need upgrading--such as preparedness beyond the 10 mile EPZs, evacuation transport measures, relocation issues, ingestion zone size and preparedness, etc. The draft report starts with a conclusion--no changes are necessary--and selects facts and events to prove the conclusion. However, many potentially instructive aspects of the Chernobyl evacuation have been ignored and left unanalyzed.

The NUREG fails to weigh any of the factors which lessened the impact of the accident on the public in the evacuation zone, for example, meteorological conditions and the accident sequence. 4
Because of dry weather conditions, radiation stayed aloft and spread over a larger area than it would have had it been raining. This decreased the intensity of exposure to those living near the plant. Such dry weather cannot be counted on at many reactor sites in the U.S. Radiation doses were further reduced by the height of the plume. The explosion and fire at the Chernobyl reactor sent radiation high into the atmosphere, also resulting in lower doses in the evacuation zone than if the plume were closer to the ground.

In discussing exposures to the public, the NUREG also fails to examine the differences in sheltering ability between Soviet and U.S. housing. While the Soviets had no warning of an impending accident, the concrete housing in the area gave officials several hours to plan orderly evacuations while citizens were sheltered in their homes. Such sheltering benefits would not be available to many U.S. citizens, whose homes often are constructed of wood.

Typically, the report does not acknowledge these dose reducing factors as such, casting these factors, instead, in an unfavorable light. For instance, the NUREG takes credit for the fact that people in the U.S. generally have cars, unlike the Soviet population around the Chernobyl plant, who were evacuated by buses. The report postulates that doses in the U.S. therefore 4.1
would be lower because evacuating populations would not have to wait for buses to take them out of the contaminated area. However, the NUREG does not examine the potential benefits of fewer vehicles congesting roads, making for speedier evacuations, nor of the decreased ability to prevent radiation from spreading

out of the evacuated area through uncontrolled, contaminated automobiles. Further, the Soviets used a polymer substance over the roads to limit radiation spread via transport vehicles out of the evacuation zone. Such methods are surely responsible for some reduction in doses to the public. By not mentioning such dose reduction measures, the NUREG appears to downplay the dangers of radiation by creating the appearance that even ad hoc evacuation plans kept doses to the public low.

Further, the draft report does not treat in any significant way, the Chernobyl data which would indicate the need for larger evacuation zones. In arguing to maintain the status quo of 10 mile EPZs, the report offers no justification for believing utilities, and state and local governments would be capable of carrying out an evacuation beyond 10 miles, given the current level of planning and training. It ignores the Chernobyl experience, and, instead, simply refers to NUREG-0654 which says "the choice of the size of the Emergency Planning Zones represents a judgement on the extent of detailed planning which must be performed to assure an adequate response base." NUREG-0654 notes that for some accidents, actions would need to be taken beyond the 10 mile zone. This, however, offers small assurance such a capability exists. 4.1

In discussing U.S. evacuation planning, the draft report states "(u)tility, State, local and Federal emergency plans were developed, reviewed, and exercised...The populations within the plume exposure pathway EPZs for U.S. plants have been informed of the risks of an accident ahd have been instructed on protective actions during an emergency." (p. 4-3) Unfortunately, this is no longer the case. The NRC's recent approval of emergency planning changes which allow utilities alone to submit plans, removes the benefits of knowledgeable, trained state and local evacuation personnel for some plants.

There is also debate over public knowledge of evacuation plans, indicating the public may not respond as planned. A recent General Accounting Office report found that "no federal agency assesses public knowledge of radiological emergency procedures." Additionally, a Massachusetts Public Interest Research Group study of public knowledge of the Pilgrim plant's evacuation plan found that residents had only a limited knowledge of the plan. Only 56% of those surveyed said they had even received the utility's information booklet on the evacuation plan. This and other surveys have shown many residents would respond incorrectly to an emergency, for instance, trying to pick up their children from school, or evacuation to incorrect locations. Much of the public confusion is due to insufficiently detailed planning, confusing changes in plans, and poor notification of the public.

Just as Three Mile Island demonstrated the need for coordinated planning between utilities, states and local governments, Chernobyl demonstrated that large scale evacuations of greater than ten miles with large amounts of radiation present may be necessary in a severe accident. The NRC should toughen its emergency planning standards--prepare for evacuations beyond ten miles, ensure greater public knowledge of the dangers of radiation and possible protective actions, and penalize utilities not in compliance with regulations, instead of extending time allowances for compliance. Further, since the Chernobyl accident demonstrated that protective measures will have to be taken hundreds of miles from the accident site, all states should be required to submit emergency plans including radiation limits for food, air and water. Such a regulation would significantly reduce confusion in an accident, by requiring state officials to be knowledgeable and prepared to take protective actions. 4.1

FIRE PROTECTION

Section 2.4 of the NUREG discusses the Soviet firefighting techniques during the Chernobyl accident, and compares them with the NRC's current regulatory practices. Citing the Appendix R regulations, the NUREG concludes "that the (NRC's) programs provide an adequate level of defense in depth for all anticipated events." (p. 2-19)

While the commission's fire protection regulations are quite lengthy, few plants meet all the requirements. In fact, NRC's fire protection standards have become perhaps the most exempted of NRC's regulations. Immediately after the Chernobyl accident, then chair of the NRC, Nunzio Palladino, stated that the first lesson learned from Chernobyl should be the importance of fire protection regulations, and stressed the need to speed up plant compliance with fire protection and alternate shutdown system requirements. 2.4

But the facts show that compliance with these regulations has moved at a snail's pace. While the Commission's regulations were adopted more than six years ago, only some 20 plants claim to be in full compliance or have been found in full compliance by the NRC. Several older plants still lack alternate shutdown systems. So while NRC's fire protection regulations may seem adequate, the reality is many plants may not be able to cope with a large fire. And instead of enforcing the requirements, NRC seems to be caving in to industry pressure to reduce the standards, either through plant-specific exemptions or through increased "flexibility" in complying with the regulations on a generic basis.

One example is particularly telling of NRC's degraded sense of responsibility to the public health and safety. In September 1986, NRC granted an exemption to the FitzPatrick plant, allowing it to operate without a proper alternate shutdown system. The utility requested an exemption from certain requirements "to the extent that the reactor coolant level would be permitted to drop below the top of the core during use of alternate safe shutdown procedures following a postulated fire whicch renders the control room uninhabitable." To this request, the Staff made a finding of "no significant environmental impact" though it was authorizing the uncovering of the reactor core for an estimated period of time.

In addition to industry reluctance to meet the requirements, the fire protection rules themselves are riddled with legal and technical loopholes. For example, the rule is based on the assumptions that a fire will not occur at the same time as any other accident and that a fire will not be among the consequences resulting from some other accident, such as a loss-of-coolant accident. However, the potential for fire is higher during some types of accidents, For example, a hydrogen fire or explosion inside the reactor building is possible following a severe loss-of-coolant accident. Another potential cause of a fire during an accident is that the equipment in safety systems, such as large pump motors, draw the most electrical current during an accident--the condition when a loose connection is most likely to initiate a fire. The list of loopholes goes on. 2.4

The NUREG boasts that 95% of all plants licensed before January 1, 1979 have made the required fire protection changes, and that all these plants will be in compliance with Appendix R by 1989. However, in adopting its fire protection rules, the NRC decided this category of plants need only meet three of the rule's 15 standards. Taking ten years to meet such requirements only demonstrates the industry's and NRC's lax attitude toward these safety requirements.

CONTAINMENTS

Throughout the draft report, NRC takes credit for the strength of U.S. reactors. While some containments are undoubtedly stronger than Chernobyl's, many U.S. containments are of questionable strength, in particular the GE plants. The GE plants use the pressure suppression system of containing accidents pressures, similar to the Chernobyl design. And though a Chernobyl-style accident is unlikely at these plants, an accident which similarly challenges the containment is possible.

The debate has raged about the likelihood of GE Mark I containments failing in an accident--percentages range from 5 to 90 %. What is certain, however, is that these containments are not as sturdy as others such as the large dry containments. That is precisely why the Commission adopted its inertion rule for Mark I and II containments, which the draft report mentions.

However, the report does not mention the loopholes in the rule, or actions by the Commission and/or utilities which further reduce the ability of these containments to contain radiation in an accident. For instance, the NUREG does not discuss the dangers 3
involved during the de-inertion periods plants are allowed during start-up and shutdown. However, accidents may be more likely to occur during start-up or shutdown when the reactor systems are being challenged or manipulated.

If only real-life was as predictable as it looks on paper. The draft report details the benefits of the hydrogen control systems, but fails to recognize NRC and utilities do not always stick to the regulations. For instance, the NRC granted Detroit Edison an exemption from inerting the containment of the Fermi II reactor. The rationale for doing so, itself, was an example of illogical reasoning--Fermi's history of management and design problems should have required strict compliance with safety rules, not a relaxation of them. Further, the exemption was granted for nearly an indefinite period of time. NRC set these regulations aside to speed full-power operation of the plant.

Other degradations of containment reliability are caused by utilities simply ignoring the regulations. Recently, workers at the Oyster Creek plant tied its vacuum breaker valves open to de-inert the containment more quickly with the reactor at 23% power. Such actions demonstrate a failure to recognize that accidents do happen. NRC's failure to shut down the reactor for extensive management and operator review points to a flaw in the

Commission's regulations and its regulatory philosophy. 3

The General Accounting Office recently criticized the NRC for the lack of guidelines to identify safety violations severe enough to require shut down of a plant. This criticism is in keeping with NIRS' criticism of draft NUREG-1251. The NRC's veritable tome of safety regulations serves little function if it is not adequately implemented and adhered to by the NRC and its licensees. NUREG-1251 is flawed in that it fails to relate the Chernobyl events and consequences to the U.S. system in any depth. While the report takes credit for U.S. safety regulations, it fails to recognize flaws in the application of these regulations. 1.1

NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

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Byron Lee, Jr.
President & Chief
Executive Officer

November 3, 1987

U.S. Nuclear Regulatory Commission
Rules and Procedures Branch
Division of Rules and Records
Office of Administration
Room 4000 MNBB
Washington, DC 20555

Re: Draft NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States" (52 Fed.Reg., September 2, 1987)

Dear Sir:

These comments are submitted on behalf of the Industry Technical Review Group on Chernobyl in response to the Nuclear Regulatory Commission's invitation for comments on the above-captioned document. The Review Group, which I chair, was formed in May 1986, to serve as a focal point for the industry assessment of the Chernobyl accident. We appreciate this opportunity to comment on draft NUREG-1251.

As you know, the nuclear industry has studied the Chernobyl accident extensively. INPO, EPRI, regulatory owners groups, and individual utilities have assessed the impact of Chernobyl, looked for lessons that could be appropriate to U.S. reactors, and prepared numerous reports and presentations. The Industry Review Group coordinated much of this activity and tapped these resources to prepare and carry out an Industry Plan of Response. My committee issued an Industry Position Paper along with that Plan of Response in February 1987. These documents, and subsequent industry activities and analysis constitute a thorough review of the accident and its implications.

Based on this independent evaluation of the event, we concluded that "The design and institutional differences between the Chernobyl-type, water cooled graphite reactor and U.S. light-water nuclear power plants are so fundamental that the Soviet accident should not impact the processes of design and regulation of U.S. nuclear reactors...." In addition, "The Chernobyl accident confirms U.S. choices in nuclear technology, supported by our public regulatory program...."

These major conclusions are in close agreement with the general conclusions in NUREG-1251. Our review group also concluded, as you did, that the most important lesson of Chernobyl is that we must continue to sustain Summary our high standards of safe design, operational controls, management and staff motivation and dedication. This vigilance, in combination with the defense in-depth that is inherent in our designs and operations, are in sharp contrast to the design and operation of the Chernobyl reactor. Our review concluded that the RBMK reactor exhibits many serious design weaknesses that permitted the accident to occur. This conclusion, now very clear from over one and a half years of careful study, is an important perspective that is relevant to the regulatory implications of Chernobyl. We believe it should be stated more explicitly in NUREG-1251.

Overall, NUREG-1251 is an excellent document. We believe its areas of initial focus were appropriate, and that the conclusions were generally sound. The background information, definitions, "current regulatory practice" and "work in progress" sections were thorough and extremely useful to the reader. Summary

We are concerned that in spite of its overall conclusions, NUREG-1251 identifies a very large number of areas that warrant further study or analysis. We would generally recommend that ongoing programs and studies should be sensitized and sometimes modified to account for Chernobyl implications. However, we believe that in most cases new independent studies of these areas would not be productive or beneficial. In almost all cases, existing programs in combination with Probabilistic Risk Assessments (PRAs) or Individual Plant Evaluations (IPEs) conducted in response to the Severe Accident policy, will address the subtle implications of Chernobyl.

NUREG-1251 searches exhaustively for potential implications from Chernobyl, even indirect ones based on inference or analogy. We attempted to use the same approach in our review, and found that there is a practical limit to how far one can stretch for lessons that are meaningful and relevant from a reactor so fundamentally different. For example, we also studied the containment issue, and concluded, as you did, that current programs addressing containment survival in severe accidents are adequate and that no new programs or initiatives are needed. However, we did not find from our study of Chernobyl, any logical basis for considering adding filtered vents to containment. The containment lesson from Chernobyl is that reactors and their reactor 3
coolant pressure boundary should be fully contained, a standard that U.S. LWRs meet. There is no basis in the uncontained Chernobyl accident for questioning the adequacy of western containment designs. Consideration for improving U.S. containment performance should be based on ongoing U.S. studies, not indirect inference from Chernobyl. Other implications we believe are too oblique include the use of dry materials dumped from helicopters for LWR accident mitigation, steam explosion research, and cold shutdown safety assessment.

Detailed comments on NUREG-1251 are provided in Enclosure 1. Four specific comments are singled out as most significant and are presented here.

First, in Section 1.2.2, "Work in Progress" related to approval of tests
and other unusual operations, it states that, "The Atomic Industrial Forum
(AIF) has accepted a task .... for licensees conducting 10CFR50.59 reviews."
AIF has merged with the U.S. Committee for Energy Awareness to form the U.S.
Council for Energy Awareness. The guidelines are now being developed in a
joint effort of NUMARC and the Nuclear Safety Analysis Center (NSAC) of the
Electric Power Research Institute (EPRI). The purpose of the guideline docu-
ment is to clarify the language in 10CFR50.59 in order that adequate and
consistent 10CFR50.59 implementation programs may be developed. A preliminary 1.2
draft guidance document was written and provided to the NRC staff for review
and comment. A public meeting was held on September 23, 1987, between the
staff and the NSAC/NUMARC Working Group to discuss comments on the preliminary
draft. These comments have been considered and the appropriate changes made.
We plan to distribute the revised guidelines to each NUMARC member as well
as the NUMARC Working Group on Technical Specification Improvements for broader
review and comment. The final industry guidelines should be available by
April 1988. We will continue to support this effort.

Second, the need for a high level, on-site, nuclear safety manager is
briefly discussed in Section 1.6.3, page 1-22. We firmly believe that safe
operation is the responsibility of everyone associated with the operating
plant. An analogy is found on page 3 of SECY-87-220, "Assurance of Quality,"
which states in part, "The assurance of quality rests with the line organiza-
tion responsible for the work product function." Similarly, safe plant opera-
tions rests with the line organization responsible for plant operations.
This responsible line organization is supported by many other groups that
are accountable to line management for safety in their functional area.
This not only includes operations but all support functions such as engineer- 1.6
ing, maintenance, quality assurance, health physics, security, personnel,
spare parts stores, document control, etc. The industry stresses the team
effort approach to achieve safe plant operation. Considering the present
programs in place to monitor and support safe plant operation, we do not
believe an additional level of management would improve safety. We agree
with the concern expressed in NUREG-1251 that it would be counterproductive
if the presence of a dedicated high level on-site nuclear safety manager
decreased the sense of responsibility by other site personnel. If additional
action is considered on this proposal, we would like the opportunity to discuss
this further with the staff.

Third, Sections 2.1 and 2.2 of the document discuss reactivity acci-
dents and accidents at low power and at zero power. NUMARC will provide the
overall coordination of the generic aspects of this issue, and work with the
four vendors, the vendor owners groups and EPRI. Results from this work 2.1,
will be integrated into the technical specification improvement program 2.2
(TSIP) as appropriate. NUMARC is working very closely with the four owners
groups on technical specification improvements. Each group has committed to
major programs to develop topicals that revise the present Standard

Technical Specifications. In addition, application of the improvements on four plant specific technical specifications on operation plants is going forward in parallel. These plants are North Anna (Westinghouse), Crystal 2.1,
River (Babcock & Wilcox), San Onofre (Combustion Engineering) and Hatch 2.2
(General Electric). These efforts include developing improved bases and utilizing human factor considerations to make them more operator friendly.

Fourth, Chapter 3, containment, is written with a perspective that containments, as currently designed, are unable to withstand the challenges of severe accidents and that modifications such as venting are required to reduce the uncontrolled release of radioactive material. NUMARC containment integrity evaluations are portrayed as initiatives to identify such modifications for BWR containments. In fact, it is not a foregone conclusion that modifica-
tions to containments are necessary. The NUMARC effort on containment in- 3
tegrity started by assessing BWR Mark I containment integrity rather than assuming an answer. Results to date show no overriding generic vulnerabilities. Rather, they confirm the wisdom of performing plant specific severe accident evaluations of entire plant systems, including containment, as called for in the NRC's Severe Accident Policy Statement. To assume a conclusion and implement generic modifications creates the risk of unnecessary, and perhaps even improper design changes.

We appreciate this opportunity to comment on the draft NUREG-1251. NUMARC will coordinate industry involvement in the disposition of the final NUREG-1251 recommendations. Through industry organizations such as EPRI, NSAC, INPO and the four Vendor Owners Groups, a significant source of technical, operational and management experience and expertise is available. We support a mutual goal -- safe commercial nuclear plant operation to protect the health and safety of the public in the United States and internationally.

Sincerely,

Byron Lee Jr.

Byron Lee, Jr.
President and
Chief Executive Officer

Chairman, Industry Technical
Review Group on Chernobyl

BL:tpg
Enclosure

Enclosure 1

SPECIFIC COMMENTS ON NUREG-1251

NOTE: The following comments include some selected examples of administrative controls employed in U.S. utilities to govern the adherence to procedures, the approval and conduct of tests, the bypassing of safety systems, and the availability of required safety equipment. Full implementation of and adherence to these and other controls, together with U.S. plant designs and protection systems should provide the desired level of safety.

SECTION 1.1 ADMINISTRATIVE CONTROLS

This section identifies the value of symptom based procedures, and notes the importance of full implementation at all plants. We agree with that conclusion.

The following are examples of complementary administrative controls used by U.S. utilities to ensure procedures are followed:

1. Training programs focus on adhering to procedures. Operator licensing programs, in particular, mandate adherence to procedures.

2. Controls also include routine supervision, monitoring, and oversight to ensure procedures are followed.

SECTION 1.2 APPROVAL OF TESTS

In Section 1.2.2, "Work in Progress" related to approval of tests and other unusual operations, it states that, "The Atomic Industrial Forum (AIF) has accepted a task ... for licensees conducting 10CFR50.59 reviews." AIF has merged with the U.S. Committee for Energy Awareness to form the U.S. Council for Energy Awareness. The guidelines are now being developed in a joint effort of NUMARC and the Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Institute (EPRI). The purpose of the guideline document is to clarify the language in 10CFR50.59 in order that adequate and consistent 10CFR50.59 implementation programs may be developed. A preliminary draft guidance document was written and provided to the NRC staff for review and comment. A public meeting was held on September 23, 1987, between the staff and the NSAC/NUMARC Working Group to discuss comments on the preliminary draft. These comments have been considered and the appropriate changes made. We plan to distribute the revised guidelines to each NUMARC member as well as the NUMARC Working Group on Technical Specification Improvements for broader review and comment. The final industry guidelines should be available by April 1988. We will continue to support this effort.

Many administrative controls are in place at U.S. nuclear utilities to control testing. Some of these controls are governed by regulatory requirements.

1. Tests are controlled by procedures, and in many cases, technical specifications.

2. Test procedures are prepared, reviewed, and approved with formal controls.

3. Test procedures are reviewed and approved by the on-site safety review committee, or are reviewed by subcommittees with approval by the main committee. Results of these reviews are documented and the documentation is subsequently reviewed by the off-site safety review committee.

SECTION 1.3 BYPASSING SAFETY SYSTEMS

Administrative controls are in place at U.S. nuclear utilities defining when bypassing of safety systems is allowed.

1. Procedures and technical specifications govern bypassing of safety systems.

2. Bypassing safety systems requires deliberate action by an operator or technician. The bypass condition is usually accompanied by a status light or annunciator to indicate that a bypass has occurred.

3. Routine operations allow bypass of safety systems by procedure, provided that redundant circuits/channels are not bypassed or are in a trip condition.

4. Emergency procedures allow bypassing (or blocking) of certain safety systems if certain criteria are met (e.g., temperature, pressure, level).

SECTION 1.4 AVAILABILITY OF ENGINEERED SAFETY FEATURES

In addition to the defense-in-depth concept incorporated in U.S. plant designs, the plant technical specifications are one of several stringent administrative controls in place which should prevent a plant from being placed in an unsafe condition such as that which occurred at Chernobyl. It is our opinion that the technical specifications currently in place, are adequate to insure that such an occurrence would not take place. Implementation of the NRC Policy Statement, 52FR13788, dated February 6, 1987, and the industry's technical specification improvement effort further supports this position.

The review being conducted to apply the criteria of the NRC Policy Statement, and the industry effort in the development of improved bases for technical specifications should provide further confirmation that the required equipment is available to respond to design basis events and maintain plant safety during all modes of operation. A review of the information developed at this stage of the effort has not uncovered any safety concerns due to equipment unavailability during plant operation in the low power/shutdown modes.

Paragraph 1.4.3 states that "the following questions must be addressed for all modes of operation:

(1) Do the TS allow engineered safety features to be inoperable during modes of operation when they may be needed?"

Should one conclude from this question that in the future Tech Specs will exclude allowing loops, legs or channels of redundant safety subsystems to be disabled or removed from service for test or maintenance while at power?

Question number "(1)" should be restated --

"Do the TS allow <u>entire</u> engineered safety features....?"

In general, the content of Sections 1.3 and 1.4 are nearly identical and could be combined. Both deal with bypassing safety systems.

Controls for availability of engineered safety features are very similar to those for bypassing safety systems.

1. Technical specifications and related procedures define the maximum safety systems and components expected for various plant conditions (certain safety systems and components are defined in the safety analyses as being "engineered safety features" (ESF)).

2. Periods of unavailability of ESF that exceed allowances stated in technical specifications and procedures are reported to NRC and result in LERs which receive extensive review with corrective actions defined.

SECTION 1.5 OPERATING STAFF ATTITUDES TOWARD SAFETY

The industry as a whole is striving towards excellence in safety as evidenced by the following:

1. Operator licensing process constantly stresses plant safety.

2. The concept of plant safety first is reinforced by continuous extensive operator retraining, including the use of simulators.

3. 1985 INPO CEO conference focused on control room professionalism as a key factor in overall improvement efforts.

4. 1987 INPO CEO conference focused on professionalism within the licensed operator community.

SECTION 1.6 MANAGEMENT SYSTEMS

The need for a high level, on-site, nuclear safety manager is discussed in Section 1.6.3, page 1-22. We firmly believe that safe operation is the responsibility of everyone associated with the operating plant. An analogy is found on page 3 of SECY-87-220, "Assurance of Quality," which states in part, "The assurance of quality rests with the line organization responsible for the work product function." Similarly, safe plant operations rests with the line organization responsible for plant operations. This responsible line organization is supported by many other groups that are accountable to line management for safety in their functional area. This not only includes operations but all support functions such as engineering, maintenance, quality assurance, health physics, security, personnel, spare parts stores, document control, etc. The industry stresses the team effort approach to achieve safe plant operation.

Considering the present programs in place to monitor and support safe plant operation, we do not believe an additional level of management would improve safety. Present programs that provide a multi-layered safety review include the off-site and on-site review committee, quality assurance and control, shift technical advisors (STAs), independent safety engineering groups, etc. Also, although not part of the utility organization, the NRC resident inspector does provide an additional, independent safety monitoring function. Additionally, we agree with the concern expressed in NUREG-1251 that it would be counterproductive if the presence of a dedicated high level on-site nuclear safety manager decreased the sense of responsibility by other site personnel.

With regard to the Assessment discussion on page 1-22, NUMARC believes the concept of an on-site nuclear safety manager is only remotely related to Chernobyl and should be evaluated independent of NUREG-1251. There is no solid basis for claiming that Chernobyl did not have such a manager, nor any clear basis for demonstrating that having one would have prevented the accident. The proposal's linkage to the Chernobyl accident is not well supported.

In Section 1.6.2, Nuclear Utilities Management Resource Committee (NUMARC) should be changed to Nuclear Management and Resources Council.

Examples of management systems that provide controls include:

1. Administrative controls and approval cycles are in accordance with administrative technical specifications.

2. Independent review mechanisms include QA/QC, NRC, INPO.

3. The corporate Safety Review Board and Plant Review Board review activities and provide oversight functions.

4. Paths of communication exist from operations management to senior executives and to NRC for safety concerns.

5. INPO evaluations focus on plant and corporate organization and administration.

SECTION 1.7 ACCIDENT MANAGEMENT

The Soviets developed a unique and heroic procedure for putting out the Chernobyl graphite fire. They dumped sand, clay, dolomite, boron carbide, and lead on an uncontained, uncovered reactor core from helicopters. As discussed later in Section 5, we believe that method of accident management is unique to uncontained graphite reactor accidents and has little relevance to LWR accident management. In graphite reactor accidents, adding water can be counterproductive; in LWRs, water is always the appropriate coolant, and the only one that can be used on an intact reactor pressure vessel.

SECTION 2.1 REACTIVITY ACCIDENTS

EPRI and other industry bodies have performed work in the area of reactivity accidents, low power operation, and positive moderator reactivity coefficients. There also have been interactive meetings between EPRI and vendors concerning this area. To date the results reported support the NUREG-1251 words in Chapter 2 which state:

> "The nuclear design of U.S. reactors, ... provides assurance against a Chernobyl-type super-prompt critical reactivity excursion."

NUMARC believes that the focus of NRC review of this subject should be for accidents initiating at low power. Thus, Section 2.1 and 2.2 could be consolidated and focussed on low power tests or transients that could cause reactivity excursions. Most of the accidents listed in Section 2.1 have already been studied at length. Many are included in FSARs and/or EOPs. For those sequences beyond the design basis that have not undergone extensive reactivity analysis, probabilistic analysis should validate the credibility of the sequence prior to detailed reactivity analysis.

The BWR transient "multiple safety relief valve failure to operate" is not truly a reactivity accident.

Section 2.1.4 suggests that a reexamination of accident sequences and design approvals of the older plants may be warranted if only to "reconfirm their validity." The basis for this conclusion is that more "sophisticated tools" are now available. It should be noted that reload analyses utilize approved, state-of-the-art methods to support core reloads whenever there have been plant modifications or changes in the methodology to correct an analysis deficiency. Therefore, the "more sophisticated tools" are utilized when required to assure the safety of the plant. These "tools" include both the NRC-approved methodologies and the awareness of current events which could change the conclusions of an existing accident sequence. Further attention to this area of concern is not warranted.

SECTION 2.2 ACCIDENTS AT LOW POWER AND ZERO POWER

As discussed above in comments on Section 2.1, a direct implication of the Chernobyl accident is that reactor safety studies should assure that the design provides protection against reactivity events that initiate at low power, as well as high power. In that regard, Sections 2.1 and 2.2 could be consolidated.

NUREG-1251 appears to define zero power accidents as accidents that initiate in modes 4, 5, or 6. As such, the zero power concern appears to be directed at decay heat removal, not reactivity transients. These zero power (decay heat removal) LWR accident sequences bear no relation to the Chernobyl accident sequence. Although studies of such sequences may be appropriate, their linkage to Chernobyl appears tenuous.

Under the discussion of Accidents at Low Power and Zero Power, NUREG-1251 states that "the entire subject of decay heat removal is being addressed in Unresolved Safety Issue A-45." This appears to imply that A-45 is addressing explicitly these low power and zero power sequences. This is not the case. However, these sequences have been studied by both NRC and industry, and are the subject of a number of EPRI, INPO, and AEOD reports. Generic Issue #99 is considering some of the zero power sequences already analyzed by EPRI, INPO, and AEOD. NRC Generic Letter 87-12 also addresses zero power sequences on PWRs. Taken together these industry and NRC programs adequately address the zero power accident question.

SECTION 2.3 MULTIPLE-UNIT PROTECTION

To treat this question in its proper perspective, we need to remember that a major reason that the Unit 4 accident was a threat to operators and equipment at Chernobyl Units 1, 2 and 3 was that the Chernobyl accident was a severe, uncontained core dispersal accident. U.S. LWRs, with full containments and no credible means of creating a super-prompt critical excursion, would not be expected to place operators at a radiological risk of equivalent magnitude to Chernobyl. We must be careful not to presume Chernobyl-type accident consequences as the basis for our conclusions on this multi-unit question.

The principal question with respect to multiple unit sites appears to be control room habitability in severe accidents. In this regard, extensive work has already been done to address this area. NUREG-0737 (TMI Action Plan) placed extensive requirements on control room designs, ventilation systems, and operator protection. Additional requirements to deal with an uninhabitable control room were imposed under Appendix R (Fire Protection). Some other programs, such as USI A-17 (Systems Interactions) and high energy line break reviews, have included reviews of other multi-unit effects.

SECTION 2.4 FIRE PROTECTION

It should be noted that the major cause of fires at Chernobyl were the large amounts of hot graphite and other core materials expelled from the reactor core, with no containment present to protect personnel and other structures from burning debris and high radiation. This situation does not exist in U.S. LWRs, since LWR cores do not burn, and are completely contained inside a strong pressure-tight containment. Also, since virtually all highly radioactive material resides in the core of both RBMKs and LWRs, but only RBMK cores can burn, it stands to reason that RBMK not LWR operators need to be prepared to fight fires that could simultaneously release extremely dangerous amounts of radioactivity. LWR fires can threaten reactor safety, as evidenced by the 1975 Browns Ferry fire, but the firefighting aspects of such an event are separated in both time and distance from the reactor core. Firefighters at U.S. LWRs should be trained and equipped to fight the fire scenarios that are credible for our reactor design. We do not believe this should include the extremely high radiation fields associated with uncontained burning reactor cores.

Finally, Appendix R has resulted in major reviews of plant fire protection beyond the original licensing studies of fire protection. Many fire protection initiatives and plant modifications resulted. Many ongoing programs are continuing to upgrade fire protection capability. EPRI is expanding prior work on "cool suits" to enhance the heat and radiation protection afforded by firefighting equipment. We believe all these efforts have resulted in a very high degree of fire protection and mitigation capability at U.S. reactors, and that new programs are not warranted.

SECTION 3.1 CONTAINMENT PERFORMANCE

Chapter 3, containment, is written with a perspective that containments, as currently designed, are unable to withstand the challenges of severe accidents, and that modifications such as venting are required to reduce the uncontrolled release of radioactive material. NUMARC containment integrity evaluations are portrayed as initiatives to identify such modifications for BWR containments. In fact, it is not a foregone conclusion that modifications to containments are necessary. The NUMARC effort on containment integrity started by assessing BWR Mark I containment integrity rather than assuming an answer. Results to date show no overriding generic vulnerabilities. Rather, they confirm the wisdom of performing plant specific severe accident evaluations of entire plant systems, including containment, as called for in the NRC's Severe Accident Policy Statement. To assume a conclusion and implement generic modifications creates the risk of unnecessary, and perhaps even improper, design changes.

NUREG-1150 should be referred to as draft NUREG-1150.

SECTION 3.2 FILTERED VENTING

The Chernobyl accident did not involve the failure of a complete containment. Therefore, we did not find from our study of Chernobyl, any logical basis for doubting the adequacy of complete containments, or for considering adding filtered vents to containment. The containment lesson from Chernobyl is that reactors and their reactor coolant pressure boundary should be fully contained, a standard that U.S. LWRs meet. There is no basis in the uncontained Chernobyl accident for questioning the adequacy of western containment designs. Consideration for improving U.S. containment performance should be based on ongoing U.S. studies, not indirect inference from Chernobyl.

Studies to date indicate that filtered vents have advantages in some sequences while presenting competing risks for other sequences. The net benefit of adding filtered vents may not be as positive as perceived by its proponents.

Also, this section tends to use the terms "filtered venting" and "venting" for BWRs interchangeably. Filtered vents are specific backfit structures, whereas "venting" as currently defined as an Emergency Procedure Guideline (EPG) strategy uses the BWR suppression pool for scrubbing.

SECTION 4.1 SIZE OF THE EMERGENCY PLANNING ZONE

Industry studies support the conclusion that the present emergency planning zones (EPZ) provide an adequate size for emergency planning. In fact, these studies demonstrate that the present zone sizes are very conservative. Two specific industry studies address the plume exposure EPZ.

The Industry Degraded Core Rulemaking Program (IDCOR) reassessed the basis for emergency planning requirements, particularly the plume EPZ, around commercial light water reactor nuclear power plants in IDCOR's Technical Report 85.4(5.3), "Reassessment of Emergency Planning Requirements with Present Source Terms." Using the approach of a "Exposure Risk Guideline," the report supports a reduction in the plume exposure EPZ.

The EPRI/NSAC study, reported in NSAC-100, addressed the following question: If the IDCOR source term estimates are used and the NUREG-0396 logic is followed, what would one conclude about the appropriate size of the EPZ?

The major conclusion stated in NSAC-100 is that a plume exposure EPZ radius of three miles or less would be justified.

These two studies provide further confirmation that the present plume exposure EPZ is adequate and very conservative.

SECTION 4.2 MEDICAL SERVICES

The discussion of Potassium Iodide (KI) includes reference to the U.S. policy established by FEMA in 1985. It also discusses the Soviet experience with KI following Chernobyl which apparently was positive. However, no mention is made of Poland's experience with the use of KI, which we understand was not as successful. Inclusion of the Polish experience with KI would provide a more comprehensive view of the use of this substance as a thyroid blocking agent. This would add an important perspective to suitability of the U.S. policy.

SECTION 4.3 INGESTION PATHWAY MEASURES

The report should mention that FDA did not use the regular PAGs but developed more conservative levels for imports of certain foods into the U.S. This reportedly was done because suitable food alternatives were readily available in this country.

The report concludes that present guidance, planning and preparedness in U.S. plants is adequate for ingestion pathways. It further states that direct comparison with the Soviet actions cannot be made because of the differences in the source term. For the enlightenment of the reader who may be called upon to discuss these positions, it would be helpful to describe these differences, even if only qualitatively, so that the basis for the conclusion is clearly understood.

SECTION 4.4 DECONTAMINATION AND RELOCATION

Some readers may not be familiar with the "Mariel boat-lift" relocation effort or its relevance to emergency planning. This subject should be clarified.

The report refers to the next Federal field exercise scheduled for June 1987. We assume this refers to the Zion exercise in Illinois. Since this exercise has been completed, the report should reflect this status and the excellent experience gained.

SECTION 5.1 SOURCE TERM

A recent DOE sponsored study concluded that the first day's releases were approximately 10%, not 25% of the total release.

The discussion in Section 5.1.1 of the role of the 1962 AEC report TID-14844 on current regulatory practice, particularly site suitability, should be clarified. The TID-14844 source term assumes 100% noble gas release, 50% iodine release (a "surrogate" for other volatile fission products) and 1% of the remainder of the core. While not inconsistent with the magnitude of the Chernobyl releases, an almost complete "disassembly" of the uncontained Chernobyl reactor was required to reach such levels. These levels are considered very conservative bounding limits on credible LWR source terms. This is due in part because reactor containment buildings exist in the U.S., and the design of U.S. reactors provides assurance against a Chernobyl type super-prompt critical reactivity excursion. The Chernobyl source term tends to validate the conservative nature of the NRC source term research.

While NUREG-1251 speaks at length about the NRC severe accident research, it should be noted that there also have been industry efforts in this area. It is appropriate to consider the IDCOR research program (which has invested $20M, work by the various owners groups on various safety questions (hydrogen, ATWS issues, etc.) and EPRI's source term research program (which has invested over $25M) in understanding these issues. Of particular importance is industry supported work on containment integrity, direct heating, steam explosions and corium interactions. When taken as a whole, these industry supported initiatives serve to demonstrate the conservatism of the NRC position on severe accidents.

The detailed discussion in Section 5.1.3 is interesting background information and interesting Chernobyl source term analysis, but has little relevance to LWR source terms. It could be shortened or eliminated.

Section 5.1.3(3)(b) entitled "Effects of Materials Deposited on the Core" (p.5-9) should be eliminated, as discussed in comments above on Section 1.7. This subsection indicates that it would be interesting and illuminating to study the role of dumping dry materials on an exposed core during an LWR severe accident. The section acknowledges that an open path for aerial deposition would be required, and "such large openings of the containment appear unlikely at U.S. reactors." We are not aware of any LWR containment testing or analysis that would indicate containment failure modes that would permit aerial dumping. More importantly, even if containment access were possible, the possibility that aerial dumping could reach core debris is very remote.

The fundamental strategy of LWR accident prevention and mitigation is keeping the core contained within the reactor vessel. If we are successful at this step, as we were at TMI, then aerial dumping would be useless. We believe that the accident management strategies used at Chernobyl are not directly applicable to LWR designs. We agree that accident management is an important consideration for future severe accident studies in the U.S., but do not believe the Chernobyl experience provides any significant insights that are directly relevant to minimizing LWR source terms.

SECTION 5.2 STEAM EXPLOSIONS

This four page discussion should be shortened or eliminated, based on the fact that the "steam explosion" discussed in Vienna is unrelated to the specific steam generated shock-wave definition associated with the term "steam explosion" as used by U.S. reactor safety experts. In Vienna, the term "steam explosion" was a general expression for mechanical failure of the reactor vault due to steam overpressure, such as in a fossil boiler explosion. This definition is quite different from the "alpha" containment failure mode discussed in WASH-1400. Again, the discussion is interesting but of little relevance to LWR severe accidents due to this important semantic difference.

SUMMARY, PAGES 3-6

Many of the conclusions listed in the front of NUREG-1251 should be modified to reflect changes in the details of Chapters 1-6.

November 2, 1987

COMMENTS OF OHIO CITIZENS FOR RESPONSIBLE ENERGY, INC. ("OCRE") ON NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States"

The basic conclusion of NUREG-1251 is that there are few lessons to be learned from the Chernobyl accident with respect to safety regulation of U.S. reactors. No immediate changes are needed; the current reactor designs, operations, and regulations provide an adequate level of protection. OCRE believes that these are the wrong conclusions to be drawn from the Chernobyl accident. The NRC appears to have put on blinders to avoid seeing the most obvious lessons of Chernobyl. These are discussed in the detailed comments below. However, the NRC has correctly perceived that "we cannot relax the care and vigilance" in U.S. nuclear power regulation. (NUREG-1251, p. 3) Yet, the NRC has programs planned or in place which would do just that. The major impetus of the source term reassessment program is to relax regulatory requirements, principally those relating to emergency planning and reactor siting. The changes to the ECCS rule and broad-scope GDC-4 rule (leak before break) which would remove the conservatism now present in the requirements, principally for the economic benefit of reactor licensees, and the changes to the emergency planning rule to allow the licensing of nuclear power plants where State and local authorities have refused to cooperate in emergency planning, are other examples of the NRC's march down the path of deregulation. Perhaps the most pernicious program threatening to erode this "care and vigilance" is the examination of present regulatory requirements to determine their risk-effectiveness and to eliminate those having marginal importance to risk (see NUREG/CR-4330). The NRC should take its own advice and halt these programs and instead devote its resources to resolving the unresolved and generic safety issues and improving the operation of the existing nuclear reactors. **Summary**

SPECIFIC COMMENTS

A. ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES

The fundamental lesson of the Chernobyl accident is that administrative controls can be violated. They are no guarantee of safety. We cannot take comfort in attempts to distinguish between Soviet and U.S. procedures, management, etc. In fact, one could surmise that such violations might more readily occur here than in the Soviet Union's repressive and authoritatian society. Our legal system and Constitutional prohibition against cruel and unusual punishment make it unlikely that a control room operator or plant manager would be sentenced to years of hard labor in Siberia for causing an accident. **1.1**

-1-

Another lesson is that administrative controls cannot compensate for basic design weaknesses. Had there been no violations, the Chernobyl accident would not have occurred. As pointed out in NUREG-1250, "Report on the Accident at the Chernobyl Nuclear Power Station", p. 2-3, the Soviets were well aware of the stability problems of the RBMK-1000, particularly at low power levels, but thought they could compensate for it. In fact, their compensatory measures were hardware-oriented (increased reliance on automated control systems, higher fuel enrichment, and lower moderator density), but humans managed to defeat the machines.

The solution to the problem of administrative controls is twofold: (1) make reactor designs inherently safe and "foolproof", such that controls are highly automated, cannot be defeated easily, and the consequences of human errors are minimal; and (2) reduce the likelihood that administrative controls will be violated. The former may involve such fundamental design changes in basic reactor concepts as not to be practicable for existing plants; however, this concept should be paramount for future reactors. In addition, it may well be possible to backfit onto older reactors the interlocks which would thwart an attempted bypass of safety systems by operators (NUREG-1251, p. 1-14). This should be required for all reactors, where possible. The NRC may also wish to inquire whether, given the weaknesses of administrative controls, currently operating reactors should be allowed to continue operating. The latter concept recognizes that operator control cannot be avoided and indeed is necessary for maintenance and repair and prevention of safety system actuations during operational mode changes. 1.1

How can human errors and deliberate violation of procedures be minimized? Clearly, good procedures and effective training play a role. But of equal or even greater importance is attitude. Plant personnel need to have a realization that nuclear power is an inherently dangerous technology, and that safety is of the utmost importance. Did the Chernobyl operators have the proper attitude? The February 1986 issue of Soviet Life may provide some clues. In an article featuring the Chernobyl station, plant personnel are quoted as saying that nuclear power is perfectly safe, safer than driving a car. Such statements are remarkably similar to those routinely made by the U.S. nuclear power industry. These statements reflect a mindset that nuclear power is basically safe; an obvious corollary is that accidents don't happen. When one operates within this belief system, recognizing and responding properly to an accident or other dangerous condition is hampered. It may not be possible to replace this belief system with one more attuned to reality. It is probably a natural result of working in a hazardous industry. "Familiarity breeds contempt" is one factor causing this. Another is that 1.1, 1.5

it becomes necessary psychologically to deny the danger to 1.1, simply be able to work in a hazardous industry. 1.5

The only solution to this problem is to substitute strict, vigorous enforcement by the NRC for the proper attitude. If operators won't respect reactor physics, then perhaps they will respect enforcement action entailing significant economic penalties. This means that the NRC's enforcement policy and actions must be severe, swift, and consistent. The NRC should not hesitate to order a plant to shut down or to impose meaningful monetary penalties should conditions warrant. The present limit of $100,000 per penalty per day is a joke. Utilities routinely complain that each day a plant is shut down costs them $1-2 million. This means that noncompliance is cheaper than compliance, if compliance requires a shutdown (and especially if the licensee does not get caught). This also raises the question of whether enhanced NRC monitoring of licensee operations is necessary.

OCRE believes that it is. Contrary to the impression conveyed in NUREG-1251, operating events demonstrate that plant personnel and management do not always act with the utmost concern for safety. For example, the operational history of the Fermi-2 plant is a lesson on how not to run a nuclear power plant. Operators there have thoroughly demonstrated their inability to follow procedures by the criticality incident of 1.5 July 1985 and the heatup incident of June 26, 1987. Incredibly, a control room operator at Fermi-2 had never even read the plant Technical Specifications. The Peach Bottom plant is another example of plant personnel disregarding administrative controls. On March 17, 1986 an operator withdrew a control rod out of sequence. The second operator monitoring the rod withdrawals did not notice the error. The Rod Sequence Control System blocked an attempt to withdraw additional control rods, and was bypassed by the operators. See Abnormal Occurrence Report, 52 Federal Register 4428 (February 11, 1987). That the NRC has found it necessary to repeatedly remind licensees of the necessity of maintaining a professional attitude and atmosphere in the control room is a strong indication that these expected standards are not always met. See Information Notice 87-21; Information Notice 85-53; IE Circular 81-02.

The NRC has shut down the Peach Bottom facility due to the problem cited in IN 87-21, operators sleeping in the control room, and earlier incidents including the out of sequence control rod withdrawal. Fermi-2, however, continues to operate, albeit at a restricted power level. The inconsistency of the NRC's enforcement actions has been noted by the U.S. General Accounting Office, report GAO/RCED-87-141, August 1987. GAO also found that the NRC has allowed plants with marginal inspection records and operating experience to operate for many years without requiring an improvement program. GAO recommends that the NRC develop guidelines for deciding whether to order a shutdown of a plant. Such guidelines, if followed religiously, would put licensees on notice of what performance is expected of them and that severe penalties will be imposed if they do not meet that performance standard. 1.5

The NRC has been far too lenient with licensees. The utilities need to be regulated by an NRC that means business. There must be a constant expectation that licensees will meet all regulatory requirements at all times, with swift and severe penalties for noncompliance. The NRC should grant exemptions from the regulations extremely sparingly, only when a safety benefit will occur from the exemption. "Living integrated schedules" and allowing operation while in noncompliance with regulatory requirements must be discontinued. If licensees do not have the monetary, technical, and human resources to meet all of the NRC's requirements then they do not deserve to be licensees. 1.5, 1.6

It is noted in NUREG-1251 (pp. 1-20, 1-23) that EOPs and operator training fall short of what is required. The NRC should establish specific criteria for these areas and set a date certain (relatively soon) by which acceptable EOPs and effective training are to be in place. Licensees which do not meet the deadline should expect a shutdown order. 1.5, 1.7

It is stated at p. 1-21 of NUREG-1251 that the NRC has terminated work to provide a technical basis for new requirements for licensee management and organization and instead intends to use "performance indicators" and SALP ratings to assess management performance. One of the disturbing aspects of the Chernobyl accident is that Chernobyl-4 had the best operating and safety record of any of the RBMK reactors. It thus appears that performance indicators and SALP ratings, had such been monitored for Chernobyl, would not have indicated that the plant was at risk of a severe accident caused by gross operator error and procedural violations. The NRC should re-establish work to provide a technical basis for assessing management and operator performance, as it is apparent from Chernobyl that the approach now in use by the NRC is certainly not capable of detecting all risk outliers in the realm of management performance. 1.6

Problems are noted in NUREG-1251 with the reviews conducted under 10 CFR 50.59 for tests, changes, and experiments. The fundamental problem is that the licensees themselves are to make the determination of which changes, tests, or experiments need NRC review and approval. Obviously, if a licensee determines that such an item need not be reviewed by the NRC, when in fact it should be, the NRC will never know about it, unless it is detected during one of the NRC's sampling inspections. It is stated at p. 1-10 that it would not be manageable for the NRC to undertake prior review and approval of these items. Why not? Isn't that the NRC's job? 1.2

If the NRC does not have the resources to do its job, then it should seek increased appropriations from Congress or increase license fees to enable it to hire the personnel needed. Self-regulation is not the answer; self-regulation means no regulation. The NRC must take a stronger and dominant role in the regulation of nuclear power reactors, rather than depending on licensee self-reporting and evaluation. 1.2

B. REACTIVITY ACCIDENTS

The Chernobyl accident should prompt the NRC to reopen the ATWS rulemaking. The final ATWS rule fell far short of the Staff's 1980 recommendations in NUREG-0460, Volume 4, particularly for BWRs. Unmitigated ATWS in BWRs poses a threat not unlike the reactivity excursion occurring at Chernobyl. A pressurization transient (e.g., MSIV closure) without recirculation pump trip would result in an autocatalytic pressure-power spiral, with the negative void coefficient. The pressure pulse would collapse the voids, increasing power, which increases pressure, which collapses voids, which increases power, etc. Such an accident is, in the Staff's judgement, apparently of such low probability that it might be ignored (NUREG-1251, p. 2-8). "Probability", and our assessment of it, has been demonstrated by the Chernobyl accident not to be a useful concept in addressing reactor risk and safety. The Chernobyl event was caused by a unique combination of design, operator errors, violation of test and reactor operating procedures, and specific events (the 9 hour hold at 1600 MW(t), delaying the test, with resultant faster power decline ramp, leading to the xenon poisoning (Nuclear Safety, Vol. 28, No. 1, January-March 1987, p. 4)). Had a risk analyst been asked prior to April 1986 to estimate the probability of this sequence of events leading to a severe accident at a RBMK reactor, the analyst would no doubt have replied that such a combination of events, errors and deliberate procedure violations is so unlikely that it need not be considered. BUT IT HAPPENED. 2.1

The NRC should examine the events listed in Section 2.1 of NUREG-1251 and implement appropriate design changes or operating limits regardless of the perception of the probability of these events.

C. FIRE PROTECTION

It is stated that the fire brigade's "typical protective equipment" provides "a measure of protection" against radioactivity. It would be useful to compare the "typical protective equipment" used in the U.S. with that (if any) used in Chernobyl. The Chernobyl firefighters suffered the most severe health consequences. Thought might also be given to the 2.4

use of robotics in fighting fires in high radiation areas, if this can be done without a reduction in fire-fighting capability. 2.4

D. CONTAINMENT AND SEVERE ACCIDENT PHENOMENA

Even though NUREG-1251 admits that the Chernobyl accident "has graphically demonstrated the effect of containment performance on the overall risks of nuclear power operation" (p. 3-3), it is concluded that no new programs or initiatives are needed in this area. OCRE would disagree. The basic problem is that containments are not designed for severe accidents, on the theory that such accidents are "incredible" and need not be considered in the design basis. NUREG-1250 shows that the Soviets too had a design-basis concept; the RBMK, while incorporating the pressure suppression features of U.S. BWRs, did not have a U.S.-style containment around the reactor core, no doubt because an accident requiring such a feature was deemed incredible. See NUREG-1250, p. 2-3: "a serious loss-of-coolant accident larger than that considered as 3.1
design basis (was) thought to be virtually impossible because of the use of numerous pressure tubes rather than a single pressure vessel"; p. 2-43: "the reactor vault overpressure system is not designed to accomodate multiple pressure tube failures." The "incredible" accident has occurred.

The NRC must, without delay, formulate containment performance REQUIREMENTS for severe accidents. Such a requirement should establish that containments are expected to be a leaktight barrier to fission product release for all postulated accidents, from design-basis, to degraded core, to severe core meltdown. Plants which cannot meet this requirement should be required to shut down until they can demonstrate compliance. Unfortunately, the NRC would not even formulate a containment performance GUIDELINE last year.

Under the proposal above, containment venting, as is now proposed in the BWR EPGs, is considered containment failure. An engineered filtered venting system, which would prevent catastrophic failure of the containment while at the same time performing an equivalent dose reduction as an intact 3.2
containment without venting, might be an allowable option and should be given further study. Such a filter would have to be bypass-proof and would need to have a high-capacity filtering medium capable of withstanding the extremely harsh environment expected in a severe accident.

NUREG-1251 ignores the fundamental lesson of the Chernobyl accident, that a severe accident is simply unacceptable. We Summary,
must devote every effort to preventing the occurrence of such 3

accidents and to mitigating their consequences should they occur. As such, there must be expedited consideration and implementation of all the suggestions and recommendations in NUREG-1251, as well as backfits and procedural measures previously identified to reduce the risk of severe accidents (such as those identified, and rejected on a cost-benefit balance, in NUREG-1150). **Summary 3**

The flawed and pernicious Severe Accident Policy Statement, which asserts that existing reactors are safe enough as is (but "requests" that licensees perform evaluations of their plants to look for risk outliers, evaluations which will apparently fall far short of a complete plant-specific PRA) should be abandoned. The NRC should immediately require existing plants to meet all applicable and feasible standards of the CP/ML rule (10 CFR 50.34(f). The Station Blackout rule should be enacted without further delay. The NRC must also revitalize its research program, start resolving the growing list of unresolved and generic safety issues, and enact regulations requiring nuclear power plants to successfully withstand a spectrum of accidents, including severe core melt accidents. **3.1**

With regard to combustible gas control, the NRC should require all containments to be inerted. "Controlled" ignition is too uncertain to be considered a solution (see, e.g., NUREG/CR-2530, Review of the Grand Gulf Igniter System, which found the igniter concept to be "marginal"). Moreover, the igniters would not even work in a station blackout degraded core accident, which, according to NUREG-1150, dominates risk for the BWR/6 Mark III (Grand Gulf, 99%). Industry arguments against inerting should be disregarded; General Electric's Advanced BWR will use the Mark III containment concept and will inert the containment. If the ABWR's Mark III containment can be inerted, there is no reason why existing plants cannot also inert their containments. Concerns about personnel access for equipment maintenance might be addressed by restructuring the maintenance outage to the scheme used in Japan. According to a speech by Commissioner Bernthal on September 2, 1987 at the International Meeting on Nuclear Powerplant Operations, the Japanese have a mandatory three-month shutdown every year. Incorporation of this plan would allow for reduction in decay heat and de-inerting the containment before performing maintenance and would eliminate the time pressure associated with outages as now utilized in the U.S. **3.1, 5.3**

NUREG-1251, p. 5-15, states that the degraded core hydrogen control measures are interim in nature, pending completion of longer-term efforts regarding severe accidents. It would be more truthful to admit that they are probably final requirements, given the pronouncement of the Severe Accident Policy Statement, that existing plants are safe enough as is, and the finding of NUREG-1150 that risk-reduction measures are **5.3** not cost-beneficial. As indicated above, OCRE finds this situation unacceptable. 5.3

Regarding the Individual Plant Examinations "requested" by the Severe Accident Policy Statement, the Advisory Committee on Reactor Safeguards in its letter of June 9, 1987 stated that licensees "will be so mystified that they will have no recourse but to retain an outside group to carry out the analysis. They will thereby miss one of the more important benefits of the IPE, that of becoming familiar enough with system performance to be able to recognize vulnerabilities in their plants, and of becoming aware of expected system performance in a severe accident." This is a disturbing statement which does not square with the assessment in NUREG-1251 that U.S. plant personnel have a good understanding of their plants. This statement also demonstrates the need for prompt action to reduce the risk of severe accidents in existing reactors. 3

The statement at p. 5-9 of NUREG-1251 that "large openings of the containment appear unlikely at U.S. reactors" has no factual basis. The 1/8 scale steel containment model tested by Sandia in 1984 failed catastrophically; this failure mode was not predicted by analytical methods, which predicted "leak before break". This underscores the importance of establishing containment performance requirements for severe accidents, and in the interim, studying the accident-management strategies employed by the Soviets and establishing plans for their use here if necessary. 3.1, 5.1

As noted above, the Chernobyl accident challenges us to rethink our concepts of risk assessment and plant design basis. The Soviets did not design the RBMK to withstand the accident which occurred because it was considered too improbable. We likewise have limited reactor designs to design basis accidents which are considered more probable. NUREG-1251 does not recognize, let alone address, this problem. PRA would probably not have predicted the Chernobyl accident to be risk-dominant, relying as PRA does on human judgement, the same judgement that assigned such accidents to the "incredible" bin in the first instance. It is not apparent that Soviet engineers are less infallible than those in the U.S. It is apparent from NUREG-1250 that the RBMK-1000 is not the crude, primitive design which the nuclear industry would like the public to believe. It is the Soviet's state-of-the-art reactor with modern computerized control systems. The Chernobyl accident was simply not anticipated by the Soviet designers, just as the TMI-2 accident was not anticipated here, because such accidents were thought to be so improbable as to be deemed incredible. Chernobyl teaches us that no accident can be classified incredible and that we don't know enough about severe accident phenomena, human behavior, material behavior, etc. to be able to say with any confidence that some accident sequences can be disregarded. We must therefore change our thinking on reactor Summary 2.1, 2.2, 3 safety and design from probabilistic and design-basis concepts to worst-case planning.

Section 3 of NUREG-1250 provides an interesting contrast between U.S. and Soviet safety analysis and regulation. The contrast is not so much their present state but rather in the different directions in which they are moving. It is apparent that the Soviets have tried to improve design and operation of their reactors. Earlier RBMK-1000 plants have fewer safety features than later models such as Chernobyl. For example, the first-generation RBMK-1000 plants did not consider large pipe breaks to be credible accidents; later models do (NUREG-1250, p. 3-14). (The Soviets considered pressure tube ruptures within the reactor vault to be beyond the design-basis, based on the expectation of leak-before-break and monitoring for leakage (NUREG-1250, p. 3-52).) The Soviets appeared to have a program of revising and upgrading their nuclear safety regulations (including the use of more prescriptive requirements) and backfitting the new requirements onto older reactors (NUREG-1250, p. 3-7). It is obvious that the Soviets looked to the designs and standards of foreign countries, undoubtedly the U.S., for the pressure suppression concept and the criteria for fuel performance following an accident (compare p. 3-17 of NUREG-1250 with 10 CFR 50.46).

The U.S. NRC is moving in the opposite direction under the current administration. As noted above, the NRC has a number of programs in place intended to relax safety regulations. The U.S. is moving from postulating instantaneous double-ended guillotine pipe breaks to using leak-before-break evaluation. The NRC has enacted the Backfit Rule, designed to hamper the imposition of new regulatory requirements on older reactors. The NRC is moving toward less prescriptive requirements. The Soviet design philosophy (NUREG-1250, p. 3-4) appears to be more comprehensive and stringent than the NRC's Safety Goal. The NRC has refused to look to the practices of other countries to incorporate their more rigorous requirements.

The Chernobyl accident has proven that the NRC is on the wrong path. Increased regulation and enforcement, and not deregulation, is the proper response to Chernobyl. Ironically, the nuclear industry in its public relations propaganda following the accident pointed to U.S. regulation as one of the reasons it believes that such an accident is impossible here. If the NRC continues its march down the path of deregulation (at the urging, of course, of the industry) the occurrence of a severe accident in the U.S. is likely to become a reality.

E. EMERGENCY PLANNING

Chernobyl teaches us that emergency planning is of critical importance and that we must plan for the worst-case, **4.1** severe accidents. As such, expansion of the Plume Exposure Pathway EPZ to the distance beyond which the EPA's protective action guidelines will not be exceeded is essential. The argument in NUREG-1251, that the 10 mile EPZ provides the basis for response in areas outside the EPZ, is not logical. The purpose of planning is to avoid an ad hoc response, which is what would be needed for protective actions outside the zone. Clearly it makes more sense to plan for the worst case, which will envelop all others, than to plan for something less and be confronted with something worse. 4.1

It is also claimed that the radioactivity release experienced at Chernobyl was unique to that plant, and that fast-moving reactivity insertion accidents are unlikely here. Again the fundamental lessons of Chernobyl are ignored. Containments should be taken credit for only if they are expected to remain intact. As noted in NUREG-1150, early containment failure cannot be ruled out for any of the plants studied. And, Chernobyl has shown us that probability of an accident should not be a factor in design and planning. The Chernobyl accident happened largely because it was not deemed credible and thus was not considered by designers and operators. Power excursion accidents (like ATWS without recirculation pump trip in BWRs) are possible in U.S. designs, and thus must be considered in the planning process.

The Soviets used some innovative methods to mitigate doses to the population, such as cloud dispersion by spraying with silver iodide from aircraft to prevent rain and coating of relocation routes with a polymeric substance to prevent resuspension of deposited radionuclides. Such measures should be studied for possible use here. They also indicate that the Soviets possibly did have extensive pre-planning for radiological emergencies, as noted by one source in NUREG-1250 (pp. 7-5, 7-7). We should attempt to learn more of the Soviet's pre-planning for nuclear power plant emergencies for insights which could be useful here. It is myopic to assert that U.S. planning is adequate and that there is little or nothing to learn from Chernobyl. 4

OCRE finds the statement at p. 4-3 of NUREG-1251, that the population within the plume EPZ in the U.S. are informed of the risks of an accident and have been instructed on protective actions, to be misleading. While the utilities may have distributed information to persons in the plume EPZ, there has never been a test of the public's comprehension of this information, i.e., do people really know what to do in an emergency? FEMA had planned to distribute a questionnaire to test this, but it was rejected by OMB as too burdensome. The Soviets found that peasants would not evacuate unless their livestock was evacuated too. Would U.S. farmers leave their livestock? This has never been established. 4.1

With respect to potassium iodide, its successful use by the Soviet public should lead the U.S. to require predistribution to the public around nuclear power plants. KI must be taken before or simultaneously with radioiodine exposure to be effective. The cost-effectiveness of KI distribution examined in NUREG/CR-1433 is flawed in that it assumes a one-year shelf life for KI, and thus, annual distribution. If unopened, KI should have an indefinite shelf 4.2
life. Thus, annual distribution need not be required. Provisions to distribute KI to persons moving into the EPZ after the initial distribution can be made by the utility, such as by giving KI to all persons who seek connection of electricity to their homes. Public education on the use of KI (and its importance, so that people will not discard it) can easily be incorporated in the emergency planning information periodically distributed anyway. It is encouraging that the Soviets reported no side effects from KI.

F. GRAPHITE MODERATED REACTORS

The NRC should study severe accidents beyond the design basis of both the HTGR and the MHTGR, regardless of their perceived probability. To neglect these accidents would be missing the prime lesson of Chernobyl. The NRC should take a 6
proactive approach with the MHTGR (and all proposed advanced reactor designs) so that unresolved safety issues are identified and resolved before the plants are built and operating.

Respectfully submitted,

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OCT 2 3 1987

Mr. Eric S. Beckjord, Director
Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Beckjord:

Thank you for asking us to comment on the NRC staff report "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," NUREG-1251.

We found the general and detailed assessments by the staff to be incisive and well written. We were in general agreement with the findings and recommendations contained in the report.

We strongly agree with the staff's recommendation calling for a systematic approach by NRC and industry to develop and implement accident management programs. The report notes that further research is needed on core debris heat removal and other phenomenological issues affecting accident management. It also points out the need to develop radiation-hardened diagnostic instrumentation and safety equipment. These are areas in which Sandia has particular expertise, and we would welcome the opportunity to help resolve these important issues. 1.7

Additional specific comments from the Sandia staff are being transmitted by David J. McCloskey under separate cover. Please contact me or Mr. McCloskey if we can be of further assistance.

Sincerely,

Irwin Welber

D. J. McCloskey
Director
Nuclear Regulatory Research

Sandia National Laboratories
Albuquerque, New Mexico 87185

November 16, 1987

Mr. Eric S. Beckjord, Director
Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Beckjord:

Following up Mr. Irwin Welber's letter of October 23, 1987, enclosed are additional comments from the Sandia staff on the NRC staff report "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," NUREG-1251.

Sincerely,

David J. McCloskey

Enclosure

November 16, 1987

Sandia National Laboratories Staff

Comments on

"Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," NUREG-1251

General Comments:

The report is incisive and well written. Our overall impression of the report is quite favorable. We strongly agree with the staff's recommendation calling for a systematic approach by NRC and industry to develop and implement accident management programs.

Page 3, General Conclusions

We strongly agree with the sentiment expressed in Item 1 that the Chernobyl accident is a reminder of the continuing importance of safe design and operation, and the need for defense in depth against accidents which exceed the design basis. Summary

Pages 3-6, General Conclusions

It would appear that accident consequence data should be listed as an area in which the Chernobyl accident provides important information. The Chernobyl accident is a rich source of information about radionuclide transport in the biosphere, uptake of radionuclides by plants, impact of contaminated food on wildlife, etc.

Page 5, General Conclusions

We strongly agree with the need for analysis of "Accidents at Low and at Zero Power." 2.2

More needs to be said about "Multi-Unit Protection." Avoiding shared safety systems is an element of considering multi-unit sites. But, shared safety systems should not be dismissed out of hand. For instance, a viable concept for a multi-unit site might indeed be a shared filter vent system for containments. The ability to cross connect emergency power sources from one unit to another seems a useful safety measure. On the other hand, careful review of how an accident at one unit might initiate accidents at other units needs consideration. Summary, 2.3, 3.2

Clearly, any energetic process that could cause early containment failure merits close scrutiny in light of Chernobyl. For LWR severe accidents, these processes include hydrogen combustion 3 steam explosions, direct containment heating, and direct melt impingement. It is unclear that most filtered venting strategies provide much protection from these processes. The importance of energetic processes which could threaten containment early in the accident progression should be highlighted. 3.2

Pages 5-6, Emergency Planning

A premier lesson learned from the Chernobyl accident is that accident management and effective emergency response are possible. A second lesson is that the best laid plans for emergency action may be disrupted by the emergency itself. For instance, the evacuation routes from Pripyat planned before the accident would have taken the residents into the plume. Plans must be sufficiently flexible to accommodate disruptions likely to develop in an emergency, particularly for accidents initiated by external events. Summary, 1.7, 4

Page 6, Severe-Accident Phenomenology

The Chernobyl accident phenomena and source term were specific to the RBMK design. But, the events of the accident do remind us of some approximations made in accident analyses that may merit re-examination. For instance, the release from a plant is usually treated as a short duration "puff." A low-level protracted release caused by core debris/concrete interactions or revaporization from RCS surfaces occurring under varying meteorological conditions is normally not considered. Further, the ingress of air into degraded fuel regions late in an accident following vessel failure is not considered in source term analyses to date. Summary, 5.1

Page 1-2, First Full Paragraph

We question the statement that "Significant effort has been expended to prepare for events involving degraded-core cooling..." We believe that it is more accurate to say that we have expended significant effort to prevent such accidents and that research is in progress to understand degraded-core phenomena. Training and procedure development for coping with severe core damage and effective management of containment will depend strongly on our understanding of the underlying phenomenology and on the performance of plant equipment in severe accident environments. 1.1

Page 1-2, "Administrative Controls..."

Is it not a lesson from Chernobyl that hazardous plant configurations need to be avoided not just by administrative guidelines but also by hardware controls? The Russians are blocking control rods so that the rods physically cannot be fully withdrawn. They are making other hardware changes to make it impossible to get into very hazardous plant configurations.

Page 1-4, "Training on Procedures"

Even the most advanced simulators available for training operators have limited capability to enter the domain of severe
accidents. It would appear that Chernobyl would suggest that 1.1
efforts should be made to incorporate our knowledge of severe accident phenomena into simulators.

Page 11, Section 1.3.1

Design-basis events should not be referred to as "bounding" or "worst case" unless those terms are more clearly defined.
Design-basis analyses make assumptions concerning operability of 1.3
equipment (e.g., availability of electrical power) which may not be "bounding" or "worst-case" from a risk point of view.

Page 1-20, Second Paragraph

Some human factor experts have taken exception to frequent shift
rotations and other utility working practices. The sleeping 1.5
operators at Peach Bottom may have been influenced by the nature of the shift rotations.

Page 1-22, Section 1.7

The ongoing examination of plants for risk dominant accident sequences is identifying many accident scenarios which develop slowly. For instance, loss of containment heat removal capabilities in BWRs do not produce core damage for tens of hours. There is confidence in the industry that "something could be done" to arrest these slowly developing accidents. NRC and industry should work to develop more explicit procedures for this class of accidents.

Page 1-23, Bottom of Page

Neither NUREG-1150 nor IDCOR go as far as implied here. These
studies considered a limited number of alternative prevention 1.7
and mitigation options. Neither study has plans for considering all dominant accident sequences and recommending accident management actions in any integrated sense. However, we agree with the desirability of taking the actions indicated in the report.

Page 1-22, Section 1.7

This section indicates the need for reliable instrumentation and control equipment to allow the operator to take appropriate actions for severe accidents. The recommended program areas include the development of new rad-hard (and hopefully other severe accident environments) equipment. The statements imply a need to provide equipment reliability by assessment or qualification processes. However, no program need was identified to develop an appropriate qualification procedure (testing basis) for the severe accident area.

Page 1-25, Section 1.7.4

"There is no need to increase or alter the scope of the ongoing programs" (to develop and implement accident management programs) is contrary to other statements in the report regarding the need to improve response to degraded-core accidents. 1.7

Page 2-5, Third Paragraph and Fifth Paragraph

The statement "... this event, where it may be pertinent, is being reviewed on a case-by-case basis" appears to contradict "No work is currently being done on any events considered for this issue." 2.1

Page 2-8, Top of Page

Power excursions during refueling operations should be included for completeness.

Page 2-10 2.2

Again, these design-basis analyses depend upon assumptions concerning the operability of plant systems (e.g., electrical power). They may not always be bounding from a risk standpoint.

Page 2-11, Section 2.3.1

The design of filtration systems for control rooms is now based on a source term consisting of primarily iodine gas and little particulate. The filtration system consists of HEPA filters to remove particulate and charcoal beds to absorb iodine. Current best-estimate source terms for severe accidents are quite different than those used to design control room filters. They consist, in general, of huge amounts of particulate and very little iodine gas. The HEPA filters have limited capacity for particulate loading (around 1-4 kg/m2).

Page 2-14, Section 2.4

We have pointed out some newly identified issues in this area. These include effects of fire or smoke on control room habitability, the lack of data or analysis on smoke movement, the fact that fire brigade does not train with actual fire environments, and questions about control system functionality after the operators leave the control room because of a fire. Codes available for fire analysis have not been validated and are known to include many questionable assumptions.

Page 2-14, Section 2.3.4

Sharing of systems should not be arbitrarily precluded for new plants. The goal should be to design systems so that a single common cause cannot compromise both units. However, the ability to cross-tie systems does not necessarily have to result in these problems. For example, two units may have diesel generators in separate buildings, with completely separate maintenance crews, etc. If the diesel generators at one unit fail, the ability for the operators to cross-tie should only enhance safety.

Page 2-16, First Paragraph

The Soviets made specific recommendations concerning both protective apparel and fire fighting equipment. Do the existing 2.4
capabilities and minimum equipment conform to these recommendations?

Page 2-14, Section 2.3.4, Last Sentence 2.3

The sentence appears to contradict itself.

Page 2-19, Section 2.4.4

We suggest that this conclusion be re-examined. For example, possible adverse effects of fire suppression have received little attention. The recent event at Surry indicates the importance this issue. Other newly identified issues need to be incorporated into detailed risk-based estimates for operating plants.

Page 3-3, Section 3.1.2

Current research and the NUREG-1150 analyses provide only a very cursory look at equipment survivability and operability in severe accident environments.

With current analytic tools which may be used in the IPEs, important phenomena may not be resolved or be overlooked. For instance, current knowledge on severe accidents does not allow high confidence against early failure of containment or steam generator tubes rupture leading to a bypass of containment. These issues need to be resolved to provide a sound technical basis for the IPEs.

The reference to GE BWR containments should be altered to specifically address Mark I designs.

Filtered venting is only one aspect of overall accident management to minimize release of fission products from containment. A more integrated approach is needed for severe accident mitigation concepts.

Page 3-3, Section 3.1.4

The conclusion "that current programs in the US are adequate... new programs or initiatives are not needed" appears to deserve more justification than it has been given in the document.

Chapter 4

The thrust of this chapter appears to be that we are better prepared than the Soviets and would respond more quickly and appropriately to severe accidents. In the accident at Chernobyl, heroic actions were taken because the facilities of the military and state could be quickly marshaled to address the crisis. When the Soviets decided to act, they did so very rapidly and in what appears to be both a flexible and organized manner.

Page 4-1, Section 4.1

Soviets at Vienna indicated they had planned evacuation routes which, unfortunately, would have taken the evacuated population into zones of higher radioactivity. Within 4 hours, an emergency response team was on site, and in about 6 hours, classic emergency actions were underway. The necessary revisions to previously made plans may indeed have been ad hoc.

This section appears to miss an essential lesson of Chernobyl - that emergency plans must have sufficient flexibility and decision-making capability to adjust to the emergency circumstances. Does the US emergency planning encompass the required flexibility to make necessary changes when circumstances dictate?

Page 4-6, Section 4.2.4

The conclusions and recommendations in this section should be assessed in terms of the Chernobyl experience. Few of the US plants have only 135,000 people within 18 miles. Yet 1240 physicians, 920 nurses, and 360 physician assistants were used for the Chernobyl accident. The magnitude of the required medical response to a severe nuclear reactor accident at a highly populated site is uncertain and may exceed the resources given in this section.

Page 5-1, First Paragraph

Although the conclusion "the specific accident mechanisms involved at Chernobyl have no exact parallel in US reactors" is true, we should concentrate on commonality of the potentially destructive phenomena, not on the differences in accident initiation. The effects produced are similar (e.g., mechanical disruption of the core, rapid water vaporization, fuel fragmentation, and hydrogen generation).

Page 5-4, Section 5.1.3

In this and several other points in the text, it is noted that the initial releases during the Chernobyl accident occurred with no warning and that such sudden releases would not be expected for accidents at Western plants. However, there is a lack of coordination between US evacuation plans and the analyses of severe accident progression. We should use what we know about severe accidents to optimize accident management and emergency plans.

The Source Term Code Package does not include models that address important radionuclide release phenomena such as revaporization, direct containment heating, and residual fuel oxidation. These phenomena need to be quantified and included in our predictive analyses.

Page 5-5, Section 5.1.3

A release of ten days is not impossible, regardless of what WASH- 1400 says. IDCOR predictions of revaporization showed releases of iodine from some plants beginning after almost two days. Releases of iodine from water pools could be very protracted according to analyses done at ORNL. Mechanical aerosol generation by core debris/concrete interactions could cause releases of 10^3 to 10^5 curies/day some ten days after accident initiation.

Page 5-6

Resuspension of fission products due to energetic hydrogen combustion is an issue which we have raised and briefly investigated. However, we are not aware that this resuspension mechanism has been considered in any "source term evaluations," and certainly there has been no prior experimental confirmation.

The report gives credence to the oxidation release mechanism. The characteristics of the release--notably high barium release--are inconsistent with oxidation. But, if it were the dominant cause of release late in the Chernobyl accident, it has real implications for US reactors. Air can be drawn into the RCS in an accident once the vessel fails. Evidence from TMI and advanced code calculations show some fuel will be present. This fuel would then be susceptible to the oxidation release. 5.1
Currently, this is not considered in US accident analyses.

Page 5-7, Item 2

Another possibility is that the graphite slumped (which is known to have occurred) about this time and the slumping enhanced cooling or brought cool materials onto the hot debris.

Page 5-9, First Paragraph

Steel model containments have failed catastrophically suggesting that a large containment opening may conceivably occur in some plants during a severe accident.

Page 5-9, Third Paragraph

What is the basis for the speculation that the hydrogen-air mixing could not have occurred "rapidly enough?" Are they talking about an explosion within the core or in the room above? Rates of mixing depend on system dimensions and gas velocities and directions, which in turn would be determined by the precursor steam explosion(s). Mixing could easily take place on time scales given by the characteristic dimensions (say 10-20 m) divided by the characteristic velocities (say 20-100 m/s or faster), i.e., from milliseconds to seconds. 5.1, 5.3

Page 5-11, Paragraph (a)

This paragraph appears to argue that overpower accidents produce "near-ideal" conditions for efficient steam explosions. This is not supported by experimental comparisons between the RIA-ST-4 experiment (conversion ratio of about 0.3 percent) to many FITS experiments (conversion ratio of from 3 percent to perhaps 15 percent). This suggests that more energetic explosions are possible in LWR lower plenum geometries with melt jets pouring through an array of small holes.

Page 5-12, Last Paragraph

This and the preceding and following paragraphs concentrate on "the potential for highly energetic steam explosion events as a consequence of RIAs..." A more pertinent issue is the effects of scale and geometry on steam explosion energetics in severe accidents in LWR accidents. 5.2

Page 5-13, Section 5.2.1

We do not know of experimental data or validated models that support the statement that "an initially separated configuration of molten core material and water" has a "vastly reduced potential for highly energetic behavior." As stated above, the exact opposite may be true. Furthermore, to our knowledge no one has been able to conclusively demonstrate the existence of "limitations on pour rates during contacting, and on associated rates of coarse mixing..." that has practical significance to reactor accidents.

Page 5-13, Section 5.2.2

Much of the discussion relates to classic steam explosions causing alpha-mode containment failure. Attention should also be given to fuel/coolant interactions within the vessel leading to possible rupture of steam generator tubes and a bypass of containment.

Page 5-17, Section 5.3.3

The "official" Soviet position taken at the Vienna meeting was that 10 percent of the graphite oxidized. In further discussions it was found that some of the Soviet experts believed substantially more graphite burned--perhaps as much as 50 percent.

Page 5-13, Section 5.2.2

The reference given does not exist. The SARRP Sequoyah Containment Event Analysis report became NUREG/CR-4700, SAND86-1135. This report does not examine the sensitivity of risk to alpha-mode failure. Neither Statements (1) or (2) are contained in the "SNL/SARRP analysis."

Page 5-14, First Paragraph

We do not know of additional technical assessments being made of "alternative contact modes, multiple steam explosions, and the potential effects of steam explosions on safety systems and/or functions." There is no ongoing research in the US that will reduce uncertainties with respect to alpha-mode failure. 5.2

Page 5-17, Section 5.3.3

The graphite may not have contributed at all to the hydrogen generated in the first few seconds after the power excursion. The primary source was probably zirconium-steam reactions, probably due to the steam explosion fragmentation rapidly increasing the surface area available for chemical reaction.

Page 5-18, First Paragraph

We question the analysis in this section. What experimental data or calculations support the belief that mixing could not take place in 7 seconds or less? Several seconds, or possibly tens of milliseconds, might be sufficient time to mix large quantities of hydrogen and air. Steam-explosive generation of hydrogen could easily produce velocities of tens or even hundreds of m/s. The momentum-dominated hydrogen-steam jets could rapidly mix with surrounding air. Only a few seconds separated the first and second explosions. To escape from the reactor building, the "leading edge" of the hydrogen jet would have to travel about 30 to 40 m in this time, implying an "average" velocity of about 10-20 m/s; the initial escape velocity from the reactor would have been much higher. Hence, it is almost certain that a great deal of entrainment and mixing would have occurred at lower elevations and that little hydrogen would have escaped. Most of it would have contributed to a well-mixed hydrogen-air cloud within the reactor building. 5.3

The possibility of a hydrogen explosion or detonation contributing to the destruction of the Chernobyl containment, altering the chemistry of the fission products, and affecting their dispersion in the atmosphere by thermal and mechanical augmentation is very relevant to LWR accidents and should not be dismissed.

Page 5-18, Section 5.3.4

Chernobyl suggests the importance of understanding many phenomena and the need for additional research on: the rates and magnitudes of hydrogen generated during explosive and nonexplosive fuel-coolant interactions, the mixing of high-velocity steam-hydrogen jets with surrounding air, the potential for steam explosions to directly initiate detonations in hydrogen-air mixtures, the influence of high-velocity-flow induced turbulence on deflagration-to-detonation transition (DDT), and the possibility that hot hydrogen-air mixtures are more prone to DDT than cold mixtures. 5.3

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November 2, 1987

NS-NRC-87-3286

Rules and Procedures Branch
Division of Rules and Records
Office of Administration
Room 4000 MNBB
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NUREG-1251, Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States (Draft for Comment)

Dear Sir:

In the September 2, 1987 Federal Register, it was announced that a draft for comment on NUREG-1251, Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States, was issued requesting comments by November 2, 1987.

In response to this request, please find attached the comments from the Westinghouse Electric Corporation. Westinghouse provided a major response following the Chernobyl incident as an integral part of the overall United States Government and industry response.

Based on this action and our continuing attention to the Chernobyl incident, Westinghouse considers the subject matter (NUREG-1251) to be an important document.

We are pleased to offer the attached comments for your consideration for addressment in preparation of the final version of NUREG-1251.

If you should have any questions on the enclosed material, please call (412/374-4868).

Yours truly,

W. J. Johnson, Manager
Nuclear Safety Department

Attachment

WESTINGHOUSE ELECTRIC CORPORATION

COMMENTS ON draft NUREG-1251

Section 1.1.2 (2) Work in Progress; Symptom/Function Based Emergency Operating Procedures

The last sentence states that,"...The ability of operators to successfully implement EOPs is directly related to their knowledge of whether or not the plant is initially operating within the safe operating envelope".

The symptom based Westinghouse EOPs, which were developed under the auspices of the Westinghouse Owners Group after the Three Mile Island accident, do not require a knowledge of the operational state of the plant immediately preceding an event. The Functional Restoration Guidelines identify several critical safety parameters and require the operators to monitor their values throughout the event. If the value of one or more critical safety parameter is found to be outside an acceptable range, the operator is given priority instructions related to returning the value of the parameter(s) to an acceptable range. This methodology was developed to enhance the operators response to an event due to any number of reasons, including the case of the plant being outside of the safe operating envelope at the time of the event.

Therefore, the indicated statement is incorrect for plants which have implemented the Westinghouse EOP package.

Section 1.3 Bypassing of Safety Systems; Definition of Bypass

The safety systems included in the plant design are generally referred to as the Integrated Protection System, which is composed of two parts: the Reactor Protections System and the Safeguards Actuation System. These systems are intended to provide automatic initiation of emergency systems in the event that certain specified plant parameters are outside of a prescribed range, thereby maintaining the reactor in a safe, stable condition.

In dealing with the issue of "Bypass of Safety Systems", one must be very careful in defining the meaning of the word "BYPASS". The general definition that is commonly used is "...to make a system unavailable", irrespective of the circumstances. This is a very broad definition and not one endorsed by Westinghouse. We prefer to use three separate definitions which specify the conditions under which the protection is unavailable:

> Bypass refers to the blocking of safety functions at plant conditions which are attained during normal startup and shutdown of the reactor and are justified by detailed safety analysis studies. The blocking of safety systems is specified by administrative procedures and is possible only if system interlocks are satisfied. In this situation, if the plant conditions exceed prescribed limits, the safety functions are AUTOMATICALLY restored.

> Defeat refers to operator actions to preclude the operation of a safety system during the recovery from an accident or off-normal event and are justified by detailed safety analysis. The defeating of safety systems is controlled by administrative procedures called Emergency Operating Procedures. In this situation, the safety functions must be reset by the operator in order to become reinitiated automatically.

Out-of-Service refers to premeditated operator actions to remove a PORTION of a safety function from service for testing or maintenance and are justified by safety analyses. The placing of systems in an "Out-of-Service" state is controlled by administrative procedures called the plant Technical Specifications and requires that a fully available alternate safety function be available during the period of the outage. This situation is one in which the safety function is not totally removed; only the degree of redundancy is decreased. In this situation, the system must be placed back into service by the plant operations staff in order for the function to become initiated by automatic means.

Section 1.4.1 Current Regulatory Practice

It should be noted that the Engineered Safety Features are generally subject to the same type of protection against bypass as the safety systems discussed in Section 1.3 of NUREG-1251. Thus, the requirement for the availability of the Engineered Safety System function during plant operation is identical to those for safety systems. The only deviation from this is during operational modes other than full power modes where the availability of such systems is governed by the plant technical specifications. This represents only a small fraction of the total operating history of the plants. Thus, the issues described in this section are only a small subset of the overall consideration of Engineered Safety Feature "bypass".

Section 1.4.3 Assessment

This section unduly focuses on recently identified issues associated with less than full power operation in which the technical specification requirements are alleged to not be consistent with the plant safety analyses. The discussion in this section fails to mention that at the plant conditions identified where the technical specification requirements may not be consistent with the safety analysis, a significantly longer time period (compared to power operation mode) exists for operator action to restore the safety function before any serious consequences to the plant occur. Thus, these inconsistencies do not represent a serious threat to safety since the operators are generally able to identify the these events of concern and take appropriate manual actions to maintain the plant in a safe condition.

Section 2.1.1 Current Regulatory Practice

In the U.S., the General Design Criteria for light water cooled reactors specified in 10 CFR Part 50 Appendix A provide the overall regulatory requirements for reactivity control. In particular, GDC-11 requires that light water reactors have a self-limiting power feedback behavior, without any protection system or operator action. This self-limiting power feedback is one major difference between the RBMK type of reactor and the LWR reactors in the U.S. The most significant contributing difference between the reactor types is that the RBMK reactor is in its most reactive configuration and therefore the highest potential for power generation with a complete lack of coolant. This behavior is in fundamental contrast to the LWR which is least reactive in this condition. Therefore with regard to coolant induced reactivity excursions, U.S. LWRs bear no comparison to the RBMK reactor. We believe that the documentation of the consequences of reactivity insertion accidents for Westinghouse PWR reactors already exists in various plant FSARs and that these consequences are within the acceptable limits, as specified in NUREG-800.

Section 2.1.3 Assessment

One may always postulate more severe consequences for reactivity insertion events (or any other accident) by assuming combinations of failures of increasing lower probability of occurrence. Therefore, the primary purpose of any additional investigations of the reactivity insertion accidents in the U.S. should only be to ensure that reactivity insertion events with very severe consequences are well beyond the current design basis envelope; that is, the consequences of events just beyond the design basis have consequences similar to, or less than, those which define the design basis envelope. By ensuring that reactivity insertion events with consequences significantly more severe than those within the design basis have a probability of occurrence which is significantly less than those which define the design basis, we establish an adequate safety margin for operation of LWRs.

Section 2.1.3 Assessment

In assessing the reactivity feedback behavior differences between the RBMK reactor and the LWR reactor, NUREG-1251 does not point out a very fundamental characteristic of the LWR design. While local reactivity coefficients may be positive within a limited range of conditions, the overall integral power feedback is always strongly negative, including under postulated accident conditions. This is an inherent characteristic of the LWR design as required by the General Design Criteria (10 CFR Part 50 Appendix A). Due to the intimate coupling of the coolant and the moderator in an LWR, the reactivity feedback is more realistically treated in terms of defects where either large changes in dependent variables affecting the reactivity are present or phase changes in the coolant are possible. The reactivity coefficient approach, while adequate for conservative analyses, does not give a complete picture of the reactivity state of the core. An example of this is the case of PWR positive moderator temperature coefficient (PMTC). PMTC implies positive feedback for an increase in temperature and could be extrapolated to imply positive (local) void feedback. However, this implication is only correct for subcooled and near saturation coolant boundary conditions. Any significant voiding, which implies partially or completely saturated conditions, will always result in the inherently negative void feedback of the LWR. This is difficult to ascertain when only reactivity coefficients are used to express the reactivity state of the core.

Section 2.1.3 (3) Assessment; Positive Moderator Coefficient in a PWR

The potential reactivity addition at operating conditions with a local PMTC is on the order of 50 - 150 pcm as opposed to the 500 pcm quoted in Section 2.1.3 of NUREG-1251. The step insertion of 150 pcm (corresponding to the maximum amount of available reactivity for an unlimited reactor heatup) would result in an asymptotic stable reactor period at BOL of roughly 1.3 DPM which is well within the ability of the control/protection system to mitigate any adverse outcome. Further, in the event of no external action, the PWR under these conditions will eventually come to a stable, zero power condition at elevated temperature.

Also, the maximum potential reactivity insertion from cold conditions due to moderator heatup is estimated to be much less than the 2.0% dK/K quoted in Section 2.1.3 of NUREG-1251 and, in fact, is less than the 1.0 % dK/K shutdown margin requirements in the shutdown modes thus precluding the addition of any nuclear heating. Even if criticality (and nuclear heating) were allowed at these low temperatures, moderator heating cannot be accomplished at any significant rate without nuclear heating and any nuclear heat will bring with it prompt negative Doppler feedback.

Section 2.1.3 Assessment

In this section, a number of accidents are identified as candidates for further study. We believe that sufficient safeguards have been incorporated into the plant design, technical specification limits for operation and emergency operating procedures for the plant staff to preclude the occurrence of serious consequences from these events. Some examples for a few of the events listed for PWRs include:

Multiple rod ejection - The operating history to date does not include any rod ejection accidents or any precursors to a rod ejection accident. Thus the probability of a single rod ejection accident is extremely small. Since no mechanism has been identified to cause a multiple rod ejection, the probability of concurrent random failures would be negligibly small.

Unlimited boron dilution - Specific Emergency Operating Procedures have been written for Westinghouse PWRs to provide recovery instructions for the plant operating staff for this event. Due to physical limitations in the plant, a very long time period is required to for this to become a serious event; adequate instrumentation and instruction is available to the operators to terminate the event before serious consequences can occur. Thus this event does not merit further investigation.

Opening of loop stop valves in a loop containing unborated water - Presently, no Westinghouse PWR is licensed for operation with loop stop valves in the closed position. Analyses of this event have been provided in license applications for operation with stop valves closed in one loop. Thus, this event does not merit further consideration.

LOCA or other injection with unborated water - Plant technical specifications require that the Emergency Core Cooling System water source contain a minimum boron concentration (generally greater than 2000 ppm). In addition, the Emergency Operating Procedures for Westinghouse plants provides specific instructions to the operating staff to ensure that unborated water is not used for core cooling. Thus, this event should not merit further investigation.

Section 2.3.4 Conclusions

This sections concludes that the severe accident policy for new plants should restrict the sharing of systems forming part of the shutdown capability at multi-unit sites. We do not believe that sufficient evidence exists to arrive at this conclusion, particularly a detailed study of alternatives designs for sharing of system important to shutdown. We believe that it would be appropriate to conclude that, for future plant designs, severe accident considerations should be taken into account when designing shared systems, particularly with respect to shutdown of the non-affected unit and eventual restart of the non-affected unit.

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'87 NOV -2 P1:42

October 30, 1987

United States Nuclear Regulatory Commission
Rules and Procedures Branch
Division of Rules and Records
Office of Administration
Room 4000 MNBB
Washington, DC 20555

Subject: Draft NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," (52FR33304)

Dear Sir:

Yankee Atomic Electric Company (YAEC) appreciates the opportunity to comment on draft NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States." YAEC owns and operates a nuclear power plant in Rowe, Massachusetts. Our Nuclear Services Division also provides engineering and licensing services for other nuclear power plants in the Northeast, including Vermont Yankee, Maine Yankee, and Seabrook.

Summary

NRC has concluded that no immediate changes are required in the design or operation of U.S. light water reactors as a result of the Chernobyl accident. We believe that even a stronger case exists to conclude that neither immediate nor future changes are necessary. Knowledge of the Chernobyl plant design and results from the evaluation of Chernobyl-accident data and information clearly show that whatever connection does exist between Chernobyl and U.S. plants, it is so remote that further NRC initiatives are decidely unjustified.

As a result of the overwhelming evidence that clearly differentiates Chernobyl from U.S. plants, we believe that the staff's use of the Chernobyl accident as a basis for pursuing new research and licensing activities is unwarranted. In NUREG-1251, the staff is recommending further research, evaluations, and activities in many areas that NRC and industry have already demonstrated to be adequately concluded in terms of public health and safety.

We recommend that the NRC not proceed with new initiatives based on what "seems to be justified," or what "may be worthwhile." Throughout NUREG-1251, the staff admits that current regulations, research, and generic programs have or will adequately address any issues that may be related to the Chernobyl accident. We contend that Chernobyl is so remotely tied to the U.S. plants that no further research or evaluations need to take place based on the Chernobyl accident. Although we believe that research and other intitatives play an important role in ensuring continued overall nuclear plant safety, we urge the NRC to carefully screen proposed activities to determine if they are likely to result in merely an enhancement of an already acceptable, conservative level of safety or are, indeed, needed to eliminate what the NRC and industry perceive to be an unacceptable risk to the public health and safety.

Summary

With regard to lessons-learned, we disagree with the staff's conclusion that the Chernobyl accident has important lessons for us. What did have important lessons for the U.S. nuclear industry was the incident at the TMI-2 reactor. This resulted in extensive actions by the NRC and the industry (and the consequential expenditure of large resources) to conclusively address all concerns. Chernobyl only confirmed the appropriateness of those actions.

In conclusion, we urge the NRC to more accurately and clearly reflect in NUREG-1251 the dramatic, yet real distinction between Chernobyl and U.S. light water reactor plants. Failure to do so will result in not only a disservice to the U.S. nuclear industry, but more importantly to the public.

Truly yours,

D.W. Edwards

Donald W. Edwards
Director of Industry Affairs

JMG/12.217

ATTACHMENT

Our comments on specific sections of NUREG-1251 follow.

Chapter 1 - Administrative Controls and Operational Practices

The staff has suggested in Section 1.6 that an evaluation be conducted to consider the need for a dedicated high-level, on-site, nuclear safety manager, whose only responsibilities are the safety of the facility.

We believe that such an evaluation is unnecessary. Unlike Chernobyl, safety is adequately addressed in the U.S. through design and design change programs, procedures, Technical Specifications, and NRC-required assignment of individuals and groups to continuously review plant activities in terms of safety during both normal and upset conditions.

In Section 1.4, the staff recommends that each licensee should perform a comprehensive review of its specific design (including design-basis-accident analyses) and Technical Specifications to determine if, for each mode of operation defined in the Technical Specifications, (1) all equipment required to mitigate the design-basis-accident has corresponding operability requirements, and (2) sufficient equipment is available to ensure that safe shutdown cooling can be maintained with redundancy while the reactor is shut down. The staff notes that such a review and corresponding changes will be conducted under the Technical Specification Improvement Program (TSIP). Furthermore, the staff recommends that future proposed changes to the Technical Specifications be accompanied by a justification that the proposed change to the Technical Specifications is consistent with the safety analysis.

Contrary to the inference of NUREG-1251, the objective of the TSIP is not to redo the kind of analysis that has already been done. As we understand the NRC Interim Technical Specification Policy Statement, the purpose of the TSIP is to streamline the Standard Technical Specifications; that is, to eliminate those requirements unrelated to either the specific characteristics of the facility or the conditions for its operation, and to remove nonessential details that are more appropriately contained elsewhere. As implied by the voluntary nature of the TSIP, the NRC recognizes that not all facilities require such streamlining to achieve the objectives set forth in the TSIP Statement. Indeed, it has been our experience that plants with custom Technical Specifications have expended a great deal of time in developing and maintaining their Technical Specifications to ensure that they provide an optimal means of meeting both regulatory requirements and also the needs of their plant and operations staff.

Controls placed on Technical Specification changes by 10CFR Section 50.59 ensure that changes to the Technical Specifications are consistent with the safety analysis. That is, the margins of safety incorporated in the Technical Specifications are derived from a facility's safety analysis. The NRC through its review of Technical Specification changes ensures that such margins of safety are maintained. Therefore, further justification as suggested by the staff is unnecessary.

ATTACHMENT
(Continued)

Chapter 2 - Design

In NUREG-1251, Section 2.1, "Reactivity Accidents," the staff concludes with the recommendation that the selection of reactivity events for analysis and the actual analyses done in the past be re-evaluated as a result of the Chernobyl accident.

Although we believe it is always prudent to reconfirm the validity of past analyses, we believe that the issues raised in Section 2.1 are of very low priority. Basic design differences between Chernobyl and U.S. plants, including the absence of positive void coefficients and the presence of fast-acting control rods at U.S. plants, assure that the superprompt critical reactivity excursion that occurred at Chernobyl will not occur at U.S. plants.

A large number of reactivity insertion events are already considered for U.S. plants with results that are far removed from potentially destructive energy levels. Additional events identified in Section 2.1 are events with low probability and require multiple failures or errors. The very low probability of these events, which are beyond the design basis, coupled with the favorable reactivity characteristics of U.S. plants, should make any examination of these events a very low priority.

Furthermore, if the NRC indeed has the resources to spare on a re-evaluation of such low-probability events, we contend that such valuable resources should be redirected to resolution of the source term issue which is significantly more important to the resolution of major ongoing NRC/industry programs. Given what we believe are the limited resources of both the NRC and licensees, we reiterate our belief that re-evaluation of low probability reactivity events should remain a low priority activity.

Chapter 3 - Containment

We agree with the staff's conclusion in reference to containment that "new programs or initiatives are not needed as a result of the accident at Chernobyl." However, we believe that the supporting evidence for this conclusion is much stronger than indicated in NUREG-1251, Chapter 3.

The staff touches on many of the severe accident issues related to containment performance and the many NRC programs currently in place to address those issues. However, the staff fails to mention the single most important containment issue related to the Chernobyl accident. That is, U.S. plants have substantial containments while Chernobyl did not. This point should be stressed and made the focal point of Chapter 3.

All of the ongoing programs are secondary to the facts that:

- o U.S. plants have substantial containments,
- o U.S. containments were built to satisfy design-basis requirements, and

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ATTACHMENT
(Continued)

- Recent studies have shown that U.S. containments can survive pressures that are several times design levels.

These facts taken together support the conclusion that containments at U.S. plants provide an effective barrier to the release of fission products to the environment and that no new programs or initiatives are required as a result of the Chernobyl accident.

Chapter 4 - Emergency Planning

In Section 4.1.3, the staff refers to the United States as having experienced its "Chernobyl" accident at TMI-2 in 1979.

Regardless of the emergency planning context in which such a statement was used, equating Chernobyl and TMI in any way is blatantly irresponsible and results in an absolute disservice to the public. We urge the Commission to remove such a reference.

Chapter 5 - Severe Accident Phenomena

The staff notes in Section 5.1.1 that the most severe release categories from WASH-1400 entail releases of volatile fission products of comparable or greater magnitudes than were released at Chernobyl, although the releases of low-volatility species were higher for Chernobyl.

To preclude public misunderstanding, we recommend that the staff delete or clarify the statement concerning releases of low-volatility species. The releases of low-volatility species were higher at Chernobyl; however, an uncontrolled explosion and subsequent carbon fire as existed at Chernoybl cannot be considered a release as treated in WASH-1400.

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This report was prepared by the Nuclear Regulatory Commission (NRC) staff to assess the implications of the accident at the Chernobyl nuclear power plant as they relate to reactor safety regulation for commercial nuclear power plants in the United States. The facts used in this assessment have been drawn from the U.S. fact-finding report (NUREG-1250) and its sources. The report consists of two volumes: Volume I, Main Report and Volume II, Appendix -- Public Comments and Their Disposition.

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