UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

PUBLIC MEETING

AFFIRMATION SESSION 80-39

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Nuclear Regulatory Commission
Room 1130
1717 H Street, N.W.
Washington, D.C.

Thursday, September 4, 1980

The Commission met, pursuant to notice, at 4:05 p.m.

BEFORE:

JOHN F. AHEARNE, Chairman

VICTOR GILINSKY, Commissioner

JOSEPH HENDRIE, Commissioner

PETER A. BRADFORD, Commissioner

NRC STAFF PRESENT:

LEONARD BICKWIT, General Counsel

M. MALSCH

J. HOYLE

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P R O C E E D I N G S

CHAIRMAN AHEARNE: We are on the record.

MR. HOYLE: The first affirmation item, Mr. Chairman, is SECY-80-364, the subject is Fees for Withdrawn License Applications for power reactor construction permits, operating licenses and other reviews.

The staff has recommended a rule change to authorize the General Counsel to take action if and when necessary to collect fees for licensing reviews, for denied, withdrawn and suspended or postponed applications. You've all approved the text of the amendment to CFR Part 170, and you've all now agreed that it should be made effective upon publication, rather than a proposed rule. May I have your affirmative votes?

(There was a chorus of Aye's.)

MR. HOYLE: I will speak to the other two for a moment. SECY-80-373, the General Counsel has provided a memorandum which asks that that be deferred for a few days. You've all agreed to that. And 80-365 the Chairman has circulated a memorandum on which you agree.

COMMISSIONER HENDRIE: Good.

CHAIRMAN AHEARNE: Very well, thank you.

(Whereupon, at 4:10 p.m., the Affirmation Session 80-39 was adjourned.)

POOR ORIGINAL

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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in the matter of: AFFIRMATION SESSION 80-39

Date of Proceeding: September 4, 1980

Docket Number: ______________________

Place of Proceeding: Washington, D.C.

were held as herein appears, and that this is the original transcr: thereof for the file of the Commission.

Suzanne Babineau

Official Reporter (Typed)

Suzanne Babineau

Official Reporter (Signature)

NUCLEAR REGULATORY COMMISSION

[10 CFR Part 50]

DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Consideration of Degraded or Melted Cores in Safety Regulation

AGENCY: U.S. Nuclear Regulatory Commission

ACTION: Advance notice of proposed rulemaking

SUMMARY: The U.S. Nuclear Regulatory Commission is considering amending its regulations to determine to what extent ~~, if any,~~ commercial nuclear power plants should be designed to cope with ~~for a broad range of~~ reactor accidents ~~which involve damage to fuel and release of radioactivity, including design for reactor accidents~~ beyond those considered in the current "design basis accident" approach. In particular, this rulemaking would consider the need for nuclear power plant designs to be evaluated over a range of degraded core cooling events with resulting core damage and the need for design improvements to cope with such events.

This advance notice of proposed rulemaking is being issued to invite advice and recommendations on several questions ~~to help the NRC shape its policies~~ concerning design and operational improvements for dealing with degraded core cooling. Therefore, the preliminary views expressed in this notice may change in light of comments received. In any case, there will be an opportunity later for additional public comment in connection with any proposed rule that may be developed by the Commission.

DATES: The comment period expires [90 days after notice in the Federal Register].

ADDRESSES: Interested persons are invited to submit written comments and suggestions to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. Copies of comments received by the Commission may be examined in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. Comments may also be delivered to Room 1121, 1717 H Street, N.W., Washington, D.C., between 8:15 a.m. and 5:00 p.m.

FOR FURTHER INFORMATION CONTACT:

M. S. Medeiros, Jr., Office of Standards Development
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 or telephone (301) 443-5913.

SUPPLEMENTARY INFORMATION:

HISTORICAL BACKGROUND

The Nuclear Regulatory Commission (NRC) is responsible for licensing and regulating nuclear power plants. Before a nuclear power plant can be built at a particular site, a construction permit must be obtained from the NRC. As a major part of the application for a construction permit, the applicant files a Safety Analysis Report. This report presents the design criteria and preliminary design information for the proposed nuclear power plant and provides information on the proposed site. The report also discusses various abnormal conditions and accident situations and describes safety features to be provided to prevent accidents or, if they should occur, to mitigate their effects on public health and safety.

In nuclear power plants, large amounts of radioactive material are generated during fission of nuclear reactor fuel. Although this radioactive material generally remains in the fuel pellets, significant amounts can be released to the reactor coolant during accident conditions. For appreciable amounts of radioactive material to be released from the fuel, it must experience damage from one or more of several possible causes. For example, a hydraulic-mechanical accident at normal fuel temperatures can burst fuel cladding resulting in release of radioactive material normally retained in the gap between the fuel pellets and the fuel clad. A more serious type of accident involving higher fuel temperatures might, in addition to rupturing fuel cladding, cause oxidation of the cladding. This, in turn, would cause hydrogen to be generated and released which would compound the severity of the accident. A still more serious accident might involve very high fuel temperatures and oxidation of a large fraction of the core's zirconium. In this case, not only would large amounts of hydrogen be released to the containment building but other thermal reactions could result in the release of radioactive material normally held captive in the fuel pellets. Finally, an accident so severe that core melting occurs could release large amounts of radioactive material to the environment if reactor containment integrity were also to be lost. Based on these considerations, a broad range of nuclear power plant abnormal conditions and accidents with the potential to cause fuel clad damage and release of radioactive material to the environment has been identified and categorized for analysis. Attempting to prevent abnormal conditions and accidents and mitigating their potential consequences have been the primary objectives of nuclear power plant safety design. The Safety Analysis Report is a key analysis document supporting the adequacy of this aspect of nuclear power plant design.

As discussed in Title 10, Chapter 1, Code of Federal Regulations, section 50.34 (a), in the Safety Analysis Report the applicant is required to determine margins of safety for both normal and abnormal operations and to determine "the adequacy of structures, systems, and components provided for prevention of accidents and the mitigation of the consequences of accidents." To assist the applicant in complying with this regulation, the NRC has published Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants[1], which describes the information to be provided in the Safety Analysis Report. In particular, section 15 of Regulatory Guide 1.70, provides guidance to an applicant concerning "design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 CFR Part 100 criteria." Regulatory Guide 1.70 explains that these design basis assumptions can, for the most part, be found in regulatory guides that deal with radiological releases and suggests use of Regulatory Guides 1.3 and 1.4, Assumptions Used for Evaluation of the Potential Radiological Consequences of a Loss-of-Coolant Accident[1]. Regulatory Guide 1.70 further states that "This analysis should be referred to as the 'design basis analysis'." Operating events corresponding to design basis assumptions are termed "design basis accidents", and satisfactory analysis conclusions concerning them allow a judgement that the facility can be operated without undue risk to the health and safety of the public. It should be noted that these events are analyzed primarily for the purpose of establishing the adequacy of engineered safety features, such features being those structures, systems, and components, designed into a plant to

[1]Available from the U.S. Nuclear Regulatory Commission, Washington, DC 20555

mitigate the consequences of postulated design basis accidents, and which supplement plant features designed to meet performance specifications for normal operations and anticipated abnormal conditions.

In the Safety Analysis Report the applicant is not required, however, to explicitly analyze accidents more severe than the design basis accidents. ~~the most serious types of possible accidents, even though~~ there are other design requirements which would presuppose events where significant core damage and release of radioactivity has occurred. For example, radioactive source terms of Technical Information Document TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites,[2] which imply a major reactor accident, are used to judge design adequacy of various engineered safety features and certain other plant systems and components. ~~Historically, the belief has been that the most serious possible~~ This approach was based on the assumption that such accidents are of sufficiently low probability that ~~evaluation and~~ mitigation of their consequences is not necessary for public safety. ~~is not explicitly required~~. This low probability was thought to result~~s~~ from the ~~a~~ "defense in depth" approach that requires conservative design, multiple physical barriers, quality assurance for design, manufacture and operation, and continued surveillance and testing to prevent such accidents.

Furthermore, in reviewing reactor plant designs using the "design basis accident" approach, the NRC does not review all structures, systems, and components but rather reviews, in varying levels of detail, only those considered "safety grade" by the applicant submitting a Safety Analysis Report. Items considered by the applicant to be outside the scope of design basis accident analyses are generally not considered to be "safety

[2]Available from National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia 22151.

grade" and are not reviewed by the NRC to see whether they will perform as intended or meet various dependability criteria. This method of classification is based on the notion that things credited in the analysis of a design basis event or specified in the regulations are important to safety and thus are "safety grade" while all else is "non-safety grade." Non-safety grade items do not receive continuing regulatory supervision or surveillance to see that they are properly maintained or that their design is not changed in some way that might interact negatively with other systems. Instead, these items simply receive what attention may be dictated by routine industrial codes and by desires to enhance plant availability.

Historically, a further assumption ~~presumption~~ in design review and licensing ~~has been to assume~~ was that if reactor plant systems can handle large-scale design basis accidents, ~~in most cases~~ they can also handle a spectrum of smaller accidents that are regarded as being "within the design envelope."

The accident at Three Mile Island resulted in ~~, which involved a sequence of events causing~~ core damage more severe than that considered in current design basis events, There was no ~~but not causing a~~ release of fission products from the core more severe than that presumed in 10 CFR Part 100 or TID-14844 and has shown the need to re-examine these historical approaches to analyzing reactor plant design and plant accidents. ~~For example,~~ The October 1979 Report of the President's Commission on the Accident at Three Mile Island[3] recommended that in-depth studies be initiated on the probabilities and consequences (onsite and offsite) of nuclear power plant accidents,

[3] "The Need for Change: The Legacy of TMI," available from the U.S. Government Printing Office, Washington, D.C. 20402.

including the consequences of core meltdown. This report recommended that these studies include a variety of small-break, loss-of-coolant accidents and multiple-failure accidents, with particular attention to human failures. The report stated that "from these studies may emerge desirable modifications in the design of plants that will help prevent accidents and mitigate their consequences. For example, consideration should be given to equipment that would facilitate the controlled safe venting of hydrogen gas from the reactor cooling system," and "consideration should be given to overall gas-tight enclosure of the let-down/make-up system with the option of returning gases to the containment building."

Similarly, the January 1980 report, Three Mile Island, A Report to the Commissioners and to the Public[4], states, "...we have come far beyond the point at which the existing, stylized design basis accident review approach is sufficient. The process is not good enough to pinpoint many important design weaknesses or to address all the relevant design issues. Some important accidents are outside or are not adequately assessed within the 'design envelope'; key systems are not 'safety related'; and integration of human factors into the design review is grossly inadequate."

COMMISSION'S INTENTIONS

Accordingly, it is the Commission's intent to determine ~~to~~ what changes, ~~extent~~, if any, in reactor plant designs and safety analyses are needed to ~~should~~ take into account reactor accidents beyond those considered in the current design basis accident approach. Accidents under consideration include ~~including~~ a range of loss-of-core-cooling, core damage,

[4]Copies may be obtained from the GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

and core-melting events both inside and outside historical design envelopes. Furthermore, the Commission will consider whether to require more coherent consideration of this range of core damage events in the design of both normal operating systems and engineered safety features. Therefore, this advance notice of proposed rulemaking is being published to provide the regulated industry and the public an opportunity to provide advice and recommendations to the Commission on ~~advise on the con-~~ what should be the content of a regulation requiring improvements to cope with degraded core cooling and ~~to cope~~ with accidents not covered adequately by traditional design envelopes. The rulemaking proceeding will address the objectives of such a regulation, the design and operational improvements being considered, and the costs of such design improvements compared to expected benefits.

Recognizing the need for prompt action to correct specific deficiencies identified during the Three Mile Island accident and subsequent investigations, the Commission is publishing~~, in parallel with this Federal Register Notice,~~ a proposed rule that would require certain interim improvements to better cope with degraded reactor cores. ~~The~~ That proposed interim rule should not be viewed as prejudging the final action concerning this advance notice of proposed rulemaking, and comments should be framed accordingly.

In addition to this FEDERAL REGISTER Notice, the Commission's Office of Standards Development is making a direct mailing to affected licensees and other known interested persons to ensure that they are aware of this advance notice of proposed rulemaking.

SUMMARY OF FEATURES BEING CONSIDERED FOR PROPOSED RULE

The Commission is considering initiating rulemaking that would ~~have the following features:~~

~~1.~~ Require that a broader range of accidents of both lesser and greater severity than the design basis accidents, including a fully melted core, be considered in plant design, plant operation, and reactor safety ~~protection~~ analyses. ~~design and analysis might be required for a range of loss-of-core-cooling events and resultant core damage,~~ ~~so that certain predictable consequences might be prevented or substantially mitigated.~~

~~2. Require more coherent consideration of core damage and release of radioactive material in design of plant structures, systems, and components and eliminate uneven treatment of accident classes by different parts of the regulations.~~

Comment is also invited on the extent to which any additional measures should be backfitted.

SPECIFIC CONSIDERATIONS

Advice and recommendations on a proposed rule reflecting the foregoing features and on any other points considered pertinent are invited from all interested persons. Comments and supporting reasons are particularly requested on the following questions:

1. If loss of core cooling and resultant core damage occur in a nuclear power plant, there are certain predictable consequences. Can these consequences be mitigated substantially, and the risk of severe public health danger thereby reduced substantially, by practical design improvements? If not, why not, or, if so, what design improvements can be made and at what estimated cost? How would your recommendations impact on other safety considerations?

2. The Three Mile Island accident was terminated after the core was damaged severely but before substantial melting occurred, a condition beyond the current design-basis-accident events considered in the safety analysis. Should the NRC require that events of this type be considered in future safety analyses? If not, why not, or, if so, what criteria would you impose to judge design acceptability?

3. Although the consequences of core-melt accidents have been considered to some extent in assessing nuclear power plant safety, such as in requirements for siting, emergency response plans, and certain engineered safety features, explicit consideration of the capability of current designs and casualty procedures to cope with core-melt accidents has not been a part of safety analysis scrutiny by the NRC. Should core-melt accidents be specifically evaluated in safety analysis reviews, and, if so, to what extent, or, if not, why not?

4. Recognizing that there can never be complete assurance that only analyzed events as delineated in a Safety Analysis Report will occur, what additional analyses, procedures, or design features would you propose to mitigate fuel damage accidents in the range from extensive clad perforation without oxidation, through a few percent clad oxidation, through extensive oxidation to full core meltdown? Would you recommend different and perhaps overlapping design features depending on the severity of core damage to be coped with?

5. To what extent should reactor design and reactor safety analysis account for engineered safety features not working at all, not working well, or being defeated by the operator, resulting in severe core damage? What limits should be placed on multiple failure and operator

error assumptions made in safety analyses and how ~~might~~ should probabilistic risk assessment be used to determine such limits?

6. ~~Are you in favor of a new requirement to construct~~ Should NRC require construction, at each nuclear reactor plant site, of a new structure for controlled filtered venting of the reactor containment structure? Would you limit the function of such a new structure to filtering particulates, elemental iodine, and inorganic iodine or would you ~~extend such an appendage to~~ include adsorption bed systems using charcoal or other processes so that organic iodine and noble gases could also be trapped? What quantities and release rates of gases and particulates would you design such a structure to handle and at what removal efficiency and cost? Do the potential reductions in risk expected from such ~~an appendage~~ a structure offset potential increases in risk that may materialize from incidents such as inadvertent operation or the concentration of hydrogen in the filtering apparatus?

7. ~~Are you in favor of requirements to incorporate~~ Should NRC require incorporation into containment design, systems for controlling combustion of hydrogen? Do you favor methods of control that suppress combustion or do you favor controlled burning? If you favor suppression of combustion, what techniques would you recommend and should they vary as a function of the design capability of current containments? If you favor controlled burning, do you recommend open flames, spark plugs, catalytic combustors, or some other means? What percent of a ~~core's~~ zirconium oxidation in the core ~~being oxidized~~ would you design for and at what rate? Would you respond differently for different reactor or containment types? If so, what differences would you recommend?

8. Would you recommend that all nuclear power plants operate with a nitrogen-enriched containment atmosphere as some BWR plants currently do? Why or why not and, if not, to which types of containment, if any, would you limit required nitrogen enrichment? ~~to?~~

Should NRC require incorporation

9. ~~Are you in favor of a new requirement to incorporate~~ into containment design, a core retention system to mitigate the consequences of core meltdown by, for example, increasing resistance to molten core debris penetration and thereby substantially reducing gas, vapor and aerosol generation to less than that which occurs when core debris is allowed to interact with concrete? Assuming a core retention system is required, do you favor a device that delays melt-through of the containment basemat, OR ~~versus~~ a device that permanently retains core debris within the containment building? If you favor delay of core melt-through, do you recommend refractory materials (such as MgO, ZrO_2) to protect the containment concrete basemat, or do you recommend some other means? If you favor permanent retention of core debris, do you recommend using refractory materials in combination with cooling systems that rely either on natural convective cooling or forced pumping of coolant around the extremities of the refractory material, or do you recommend some other concept? Would you respond differently for different containment types? If so, what differences would you recommend? How do your recommendations impact on other safety considerations?

Should NRC require design changes to account for

10. ~~What design changes, if any, would you recommend to account for~~ increased radioactive material that may be transported during an accident by systems normally functioning with much lower levels of

radioactive material such as the steam and residual heat removal systems and the containment drainage system?

11. Should NRC require ~~Are you in favor of actions such as requiring~~ more extensive operator training, ~~requiring~~ strict literal compliance with new and improved detailed operating procedures, increased reliability of emergency cooling or decay heat removal capability, and expand~~ing~~ed control room minimum manning as alternatives or supplements to degraded cooling design improvements?

12. Should NRC require ~~Are you in favor of a requirement to provide~~ an alternate, add-on, self-contained decay heat removal system to prevent degradation of the core or to cool a degraded core, in contrast to ~~providing~~ the previously discussed schemes which are aimed toward mitigating the consequences of degraded core cooling? How would such a decay heat removal system affect other safety considerations?

13. Should NRC require ~~Are you in favor of a requirement to locate~~ systems such as the make-up and purification systems to be located in a leak-tight building? Would such a requirement add to or detract from overall plant safety?

14. What design, quality and seismic criteria would you recommend for any additional systems to prevent the potential breeching of containment such as systems for controlled filtered venting, hydrogen combustion control, and core retention mentioned in previous questions? Do you favor evaluating designs of such systems on a realistic basis, as opposed to the conservative method used to evaluate engineered safety features? Do you favor establishing design criteria for such systems that are equally stringent, less stringent, or more strin-

gent than those applied to engineered safety features? Please explain your response in terms of criteria you would recommend, including consideration of redundancy, diversity, testability, inspectability, and structural design limits (including seismic requirements).

15. Can probabilistic analysis be used both as an aid in determining and comparing the adequacy and usefulness of the several features mentioned in previous questions and as an aid in determining the design criteria and reliability requirements for these features? How do you view the utility of quantitative risk analysis in better understanding the safety advantages and disadvantages of the several features mentioned in previous questions?

16. In weighing the costs of design and operational improvements to cope with degraded core cooling against the benefits of their utilization, what quantitative methods or other Guidance ~~and rules of thumb~~ would you suggest to facilitate preparation of a useful value-impact assessment? Would you consider useful or appropriate comparisons between nuclear power plant risks and other risks to which people are exposed?

17. What aspects of degraded cooling or melted-core accidents are sufficiently unknown or uncertain ~~so~~ as to impede ~~mitigating system~~ design and analysis of mitigating systems, and thus requir~~e~~ing additional research or experimentation?

18. The NRC has under way a separate rulemaking proceeding~~s~~ concerning ~~emergency planning and~~ reactor siting. If you are familiar with these separate activities, how would you modify present and proposed requirements for emergency planning and reactor siting if accidents

and an emergency planning Rule has recently been approved.

beyond the present design basis were to be considered in nuclear power plant safety analyses?

Dated at Washington, D.C., this ____ day of _________ 1980.

For the Nuclear Regulatory Commission.

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Samuel J. Chilk
Secretary of the Commission