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December 30, 1980

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

Attention: Docketing and Service Branch

Subject: 10CFR Part 50, Domestic Licensing of Production and Utilization Facilities; Consideration of Degraded or Melted Cores in Safety Regulation, Advance Notice of Proposed Rulemaking (45FR65474), October 2, 1980.

Dear Sir:

Your proposal for a rulemaking on the subject of degraded or melted cores has been reviewed and comments are presented below. These comments have been generated to assist the Nuclear Regulatory Commission in establishing a meaningful, workable rule, if such a rule is ultimately considered necessary.

It is recommended that an integrated approach be taken to the rulemakings presently under evaluation by the NRC. Therefore, a complete program plan should be developed that integrates these contemplated rulemakings, describes the basis upon which rules will be established, treats licensing in the interim, and defines the implementation schedule. This program plan will ensure that the generic efforts are properly sequenced, evaluation in proposed areas are removed from individual plant hearings, and plant licensing decisions can be made during the interim. A preferred sequence of rulemakings is safety goal, degraded or melted cores, minimum engineered safety features, emergency planning, and siting criteria, as suggested in the AIF response.

It is strongly recommended that this program plan stress the utilization of probabilistic risk assessment (PRA) techniques in the evaluation of accident scenarios and that an integrated risk,safety target or goal be established to measure the extent to which a given plant meets or exceeds the goal. Included in the PRA should be a realistic or "best estimate" treatment of the course of a given accident such that the consequences are assessed properly. Uncertainty bounds could then be placed on the individual accident evaluations and also the total integrated risk for a plant. This process would provide the necessary quantitative evaluation of plant safety as compared to ultra conservative design basis evaluations for predetermined accident scenarios. The ultra conservative design basis evaluations technique has served the nuclear industry well in the past, but it is now time to consider, with the increased attention on the very low

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probability Class 9 type accidents, a shift in the evaluation basis of plant safety.

The assessment of a nuclear plant against an integrated safety target or goal will be a difficult task. Establishment of the technical bases for accident evaluation will require a significant amount of additional research, in some cases. Definition of sequence probabilities would be limited to existing data bases. In addition, the methodology would need to be better established. However, using PRA techniques will allow the designer and operator to pursue and receive "credit" for accident prevention techniques and systems that are generally bypassed in the design basis evaluations. Factors considered in accident prevention techniques and systems should include system reliability and availability, accident prevention systems, periodic testing, periodic inspections, and operator training. Thus, with the use of PRA, emphasis could be focused on accident prevention rather than accident mitigation, which could result from a design basis evaluation. It is generally better to alleviate the cause rather than mitigate the effect. Further, using PRA will lead to the evaluations of higher probability events which could have a larger contribution to integrated risk than the lower probability events.

The design basis evaluation concept can lead the designer and operator to choose setpoints, parameters, systems, and interface criteria that are less than optimum for the overall plant. Some previous examples of this for the design basis evaluation of the loss of coolant accident include the choice of internal fuel pressure, accumulator water volume, and containment heat transfer capability. Performing a PRA using realistic or "best estimate" accident evaluations for the various accident sequences and integrating these results together will allow the designer and the operator to choose the optimum configuration and method of operation from an overall plant viewpoint. Further, the operating procedures will be enhanced with a realistic accident evaluation methodology which will contribute significantly to overall plant safety.

Several of the attached questions deal with the addition of a required system to mitigate the consequences of a degraded or melted core. One should first assess the risk associated with a degraded or melted core and compare this risk to other higher probability events to judge if the risk of degraded or melted core is sufficiently low that no additional systems are needed. If it is determined that the core melt risk should be lowered, then one should evaluate prevention systems or techniques that tend to lower the probability and hence, the risk, against systems that tend to mitigate the accident and associated consequences, with a subsequent lowering of risk. Further, before one can judge whether an additional system is beneficial, one should compare the risk without the add on system against the risk and benefit after the system is added. In some cases, the addition of a system could increase the integrated plant risk. Cost/benefit analyses should also be employed in the decision making process of whether to add another system. Hence, no additional systems should be required by the NRC until the above steps have been completed. It is premature at this time to require additional systems when the safety goal or target has yet to be established.

The enclosure contains brief answers to the specific questions that were posed in the advance notice of rulemaking, whereas the discussion above presents the philosophy and underlying rationale behind these answers. I would be pleased to discuss these comments with you at your convenience.

Sincerely,

James O. Cermak

James O. Cermak, Manager
of Nuclear Safety

JOC/amw

cc: Dr. Milton Plessett, Chairman, ACRS

Attachment

Question

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 practicable design improvements? If not, why not, or, if so, what design improvements can be made and at what estimated cost? How would your recommendations affect other safety considerations?

Answer

1. The consequences of a loss of core cooling and resultant core damage occurring in a nuclear power plant, are not clearly predictable. Therefore, the predictions of consequences have tended to be conservative. Before one addresses the mitigation of consequences, one should determine if the plant risk from a degraded or melted core is suffiently low that the currently installed systems are adequate. Further before one can judge whether an additional system is beneficial, one should evaluate the risk without the add on system and the risk and benefit after the system is added. In some cases, the addition of a system could increase the integrated plant risk. Cost/benefit analyses should also be employed in the decision making process of whether to add another system. Hence, no additional systems should be required by the NRC until the above steps have been completed. It is premature to require additional systems when the safety goal or target has yet to be established.

Question

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?

Answer

2. The lesson of Three Mile Island is that the NRC should focus on accident prevention, not the development of another design basis evaluation based on the Three Mile Island scenario. Historically, it has been shown that accidents generally do not follow the prescriptions that have been developed ahead of time for the design basis evaluations. Safe design and operation is ensured by a combination of defense in depth safety analysis, design reviews, NRC reviews, quality assurance, surveillance, training, and ISI. These features, when adequately implemented, should preclude the occurrence of accidents characterized by a degraded or melted core. The Three Mile Island accident indicates there were deviations from a number of these features. Even though the Three Mile Island accident went beyond the normal design basis concept, the release of radiation to the public was minimal.

Using PRA techniques will allow the designer and operator to pursue and receive "credit" for accident prevention techniques and systems that generally bypassed in the design basis evaluations. Factors considered in accident prevention techniques and systems should include system reliability and availability, accident prevention systems, periodic testing, periodic inspections, and operator training. Thus, with the use of PRA, emphasis could be focused on accident prevention rather that accident mitigation, which could result from a design basis evaluation. It is generally better to alleviate the cause rather than mitigate the effect.

Question

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?

Answer

3. No, explicit consideration of the capability of current designs to cope with core melt accidents should not be part of the safety analyses performed. Casualty procedures are written to preclude a core-melt scenario. Consideration of the degraded or core-melt sequences should be included as part of the PRA and included in the integrated risk assessment against the safety goal or target to measure the extent to which a given plant meets or exceeds the goal. Included in the PRA should be a realistic or "best estimate" treatment of the course of a given accident such that the consequences are assessed properly. Uncertainty bounds could then be placed on the individual accident evaluations and also the total integrated risk for a plant. This process would provide the necessary quantitative evaluation of plant safety as compared to ultra conservative design basis evaluations for predetermined accident scenarios. The ultra conservative design basis evaluations technique has served the nuclear industry well in the past, but it is now time to consider, with the increased attention on the very low probability Class 9 type accidents, a shift in the evaluation basis of plant safety.

Question

4. Recognizing that there can never be complete assurance that only analyzed events as delineated in a Safety Analysis Report will occur, what ditional analyses, procedures, or design features would you propose mitigate fuel damage accidents in the range from extensive clad rforation 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?

Answer

4. Realistic analysis of accidents in the PRA analysis and emphasis on accident prevention should be the preferred approach in safety evaluation and in meeting of safety goals. Hence, the generation of arbitrary groundrules for design should be strongly discouraged. Discussion of design features to cope with a degraded or melted core is contained in the answer to question 1.

Question

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 should probabilistic risk assessment be used to determine these limits?

Answer

5. Probabilistic risk assessment should be the recommended approach to account for the reliability of the engineered safety features in combination with realistic accident analysis. The success of the plant design could be measured against an integrated risk target or goal. The design basis evaluation concept can lead the designer and operator to choose setpoints, parameters, systems, and interface criteria that are less than optimum for the overall plant. Some previous examples of this for the design basis evaluation of the loss of coolant accident include the choice of internal fuel Pressure, accumulator water volume, and containment heat transfer capability. Performing a PRA using realistic or "best estimate" accident evaluations for the various accident sequences and integrating these results together, will allow the designer and the operator to choose the optimum configuration and method of operation from an overall plant viewpoint. Further, the operating procedures will be enhanced with a realistic accident evaluation methodology which will contribute significantly to overall plant safety. Discussion of the limitations of the design basis evaluation approach is contained in answers to questions 2 and 3.

Question

6. Should the NRC require construction, at each nuclear reactor plant site, of a new structure for controlled filtering 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 include adsobption 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 a structure offset potential increases in risk that may materialize from incidents such as inadvertent operations or the concentration of hydrogen in the filtering apparatus?

Answer

6. No, the NRC should not require construction of a new structure for controlled filtered venting of the reactor containment structure. As stated previously, this design approach of mitigating new design case scenarios with required systems does not in general yield a good, prudent, reliable, and the most optimum overall plant design. In addition, as the question implies, risk increases are possible in utilizing this approach. See further discussion in the answer to question 1.

Question

7. Should the 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 zirconium oxidation in the core and at what rate would you design for? Would you respond differently for different reactor or containment types? If so, what differences would you recommend?

Answer

7. No, the NRC should not require incorporation into containment design, any further systems beyond the hydrogen recombiner systems for controlling combustion of hydrogen. See further discussion in the answers to questions 1 and 6.

Question

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?

Answer

8. Nitrogen enriched containments are not recommended. Plant risk may increase with this approach, since access to the containment for maintenance and inspection would tend to more restrictive. See further discussion in the answers to questions 1 and 6.

Question

9. Should the NRC require incorporation 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 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 affect other safety considerations?

Answer

9. No, the NRC should not require incorporation into the containment design, a core retention system to mitigate the consequences of a core-melt. See further discussion in the answers to questions 1 and 6.

Question

10. Should the NRC require design changes to account for increased radioactive material that may be transported during an accident by systems normally functioning with much lower levels of radioactivity such as the steam and residual heat removal systems and the containment drainage system?

Answer

10. One should assess as part of the integrated risk of the overall plant, the contributions to increased risk from radioactive material that may be transported by systems normally functioning with much lower levels of radioactivity. This should not require any design changes, but if there is a need, they should be oriented toward the system reliability approach or credit given for more extensive operator training. See further discussion in the answer to question 2.

Question

11. Should the NRC require more extensive operator training, strict literal compliance with new and improved detailed operating procedures, increased reliability of emergency cooling or decay heat removal capability, and expanded control room minimum manning as alternatives or supplements to degraded cooling design improvements?

Answer

11. Additional operator training and improved operating procedures are definite enhancements to safety. Operating procedures, however, are very dependent on the analysis input used to generate them. Hence, realistic analysis methods and models should be generated to form the basis for the development of these operating procedures. Other areas, such as the reliability of the emergency core cooling system, should be assessed to ensure that they are in consonance

with the integrated risk target goal that is advocated in the answers to other questions. See further discussion in the answer to question 2.

Question

12. Should the NRC require 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 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?

Answer

12. No, the NRC should not require an alternate, add-on, self contained decay heat removal system. Several means already exist to remove decay heat from the primary system. These include the normal heat exchange systems, the auxiliary feedwater system, the residual heat removal system, the charging/letdown system, and feed and bleed operation. However, the thinking implied in this question of preventing core-melt is a more appropriate direction to pursue than to assume core-melt has occurred and mitigate its consequences. See further discussion in the answers to questions 1 and 6.

Question

13. Should the NRC require systems such as the makeup and purification systems to be located in a leak-tight building? Would such a requirement add to or detract from overall plant safety?

Answer

13. No, the NRC should not require systems such as the makeup and purification systems to be located in a leak tight building. These systems as well as many others are already located in a controlled leakage area and this configuration is felt to be completely acceptable. These systems should be included in the integrated risk assessment that is described in the answer to question 1.

Question

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

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

Answer

14. None of the systems mentioned in this question are required for LWR plants and further, they are being considered by NRC on the basis of mitigating core-melt instead of accident prevention. It is believed that the emphasis should be on accident prevention rather than mitigation as stated in the answer to question 2. Specification of design, quality, and seismic criteria is premature, at best, since the PRA has not been performed on a realistic basis and the integrated risk safety goal or target has yet to be established.

Question

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?

Answer

15. Probabilistic analysis combined with consequence analysis can most definitely be used as an aid in determining the adequacy and usefulness of a proposed design modification and also the establishment of the design criteria. However, a realistic probabilistic risk analysis should first be performed for a core-melt scenario before one decides that core-melt is a "given" design criteria and attempts to mitigate its consequences. See additional discussion in the answer to question 2.

Question

16. In weighing the costs of design and operational improvements to cope with degraded core cooling against the benefits of their use, what quantitative methods or other guidance 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?

Answer

16. A design improvement or operational improvement should be weighed against the amount that the risk is reduced. Of course, if the integrated risk is already below the target, further reductions are not generally warranted. Comparison between nuclear power risks and other risks is very important and virtually mandatory to establish the integrated risk target for nuclear power.

Question

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

Answer

17. If one postulates a melted core accident, there are many areas in which increased research and experimentation would be useful. Some of these areas include the fission product distribution in the gaseous, liquid, and solid phase, the distribution of these fission products within the containment, heat transfer behavior between the various core-melt products and structures/cooling media, and containment gas distribution and mixing over short time periods.

Question

18. The NRC has under way a separate rulemaking proceeding concerning reactor siting and an emergency planning rule has recently been approved. If you are familiar with these separate activities, how would you modify present and proposed requirements for emergency planning and reactor siting if accidents beyond the present design basis were to be considered in nuclear power plant safety analyses?

Answer

18. The NRC has already included consideration of the degraded core cooling and core-melt scenarios in the emergency planning and reactor siting requirements. Any further modifications, if any, should await the results of plant evaluations against an integrated risk target and the proposed degraded core cooling rulemaking hearing.