RS 226-5

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 99297

DOCKET NUMBER PROPOSED RULE PR50
45FR65474

(29)

POWER SYSTEMS

December 31, 1980
LD-80-075

DOCKETED JAN - 6 1981 Office of the Secretary Docketing & Service Branch

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

ATTN: Docketing and Service Branch

Subject: Consideration of Degraded or Melted Cores in Safety Regulation 45 Federal Register 65474, October 2, 1980.

Reference: C-E letter from A. E. Scherer to Secretary of the Commission, LD-80-060, November 6, 1980.

Gentlemen:

Combustion Engineering (C-E) has reviewed the subject Federal Register notice and has participated in the formulation of, and is in general agreement with, the AIF comments on that notice. We would like to take this opportunity, however, to provide you with some further general comments on the Advanced Notice of Rulemaking (ANR) and to provide specific comments on the questions presented in that notice.

C-E is very concerned with the apparent focus of the Commission as evidenced by this ANR. In particular, we disagree with the Commission's apparent orientation towards requiring specific hardware fixes at this early stage of the rulemaking. This approach, in our opinion, goes far beyond the recommendation of the President's Commission on the Accident at Three Mile Island (The Kemeny Commission), which is referenced in the Historical Background section of the ANR. The President's Commission recommended that "in depth studies be initiated on the probabilities and consequences (onsite and offsite) of nuclear power plant accident, including the consequences of core meltdown". However, in many cases this ANR addresses whether the NRC should "require" particular hardware fixes. Indeed, the primary focus of this ANR is towards evaluation of those plant modifications which could be used to mitigate the consequences of a core-melt accident. We believe that the recommendations of the President's Commission would more properly be implemented by conducting the recommended studies and by evaluating the results in the context of an overall safety goal which is being developed as part of a separate rulemaking.

C-E recognizes that the assessment of various hardware modifications to mitigate the consequences of core melt accidents may very well be the next step towards reducing the risk associated with operating a nuclear power plant, if such modifications are needed to meet an overall safety goal. However, requiring specific hardware features to mitigate an event presupposes that a particular feature provides the most cost-effective method of risk reduction for an individual plant.

L-4, 17.50

8102090021

One of the significant lessons to be learned from TMI is that the regulatory process has traditionally devoted far too much attention to low probability events. As an example of this, the ECCS Rulemaking resulted in requirements for ECCS analyses of large, double-ended primary system pipe breaks. The Water Safety Research effort was also directed toward the investigation of phenomenon associated with large breaks. Only recently has the LOFT test series L3-5 and L3-6 investigated a phenomenon (Reactor Coolant Pump Operation) significant to small breaks. While it was accepted that small breaks were more probable, the regulatory process required an overemphasis on the large break because the consequences were thought to be more severe. It is from this experience that we express a concern about undue emphasis on degraded core mitigation without the proper use of the quantitative risk assessment techniques to assess its true value.

In our referenced letter, C-E has previously provided comments regarding the NRC's proposed interim requirements related to degraded cores. We feel it is important, however, to restate our belief that in order to provide a stabilized licensing process until a final rule is issued, the interim rule should be the basis for licensing decisions and should clearly state that compliance with the interim rule is a sufficient basis for licensing approval. When a final rule is subsequently implemented, additional features should not be required unless they are clearly needed to achieve an acceptable level of safety and are justified by a rigorous cost/benefit analysis.

In addition, no requirements for plant modifications related to degraded cores should be established in the absence of a safety goal, and no new requirements should be implemented without allowing for proper evaluation of their effect on integrated plant operation. Any new requirements which are ultimately developed as a result of these proceedings should be expressed as criteria to be met by the applicant, as opposed to requirements for new systems or prescriptive approaches to meeting those criteria.

In closing we would like to state that if the goal of the NRC in these proceedings is to assure that operation of nuclear power plants are conducted at an acceptable level of risk to the public, then the thrust of this rulemaking proceeding should be integrated with the development of a safety goal and the assessment of nuclear power plants against that goal.

Our comments on individual questions are provided in Attachment (1).

Very truly yours,

COMBUSTION ENGINEERING, INC.

Thomas J. Rice /

A. E. Scherer /fa
Director
Nuclear Licensing

AES:cw

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 heath danger thereby reduced substantially, by practicable design improvements? If not, why not, or, if so, what design inprovements can be made and at what estimated cost? How would your recommendations affect other safety considerations?

RESPONSE TO QUESTION #1

It is the view of Combustion Engineering that the emphasis should be on prevention of core damage via operator training for emergency situations and the use of high reliability equipment. At this time Combustion Engineering recommends against design modifications, until reasonable analytical methods have been developed to assess this condition and a clearly defined safety goal exists against which the results can be judged.

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?

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 casulty 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?

QUESTION #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 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?

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?

RESPONSE TO QUESTIONS #2, 3, 4, & 5

Each of these questions is concerned with whether additional analyses should include the plant assessment under core melt conditions.

It can not be reasonably determined to what extent core melt accidents should be evaluated and designed against in the absence of an established safety goal and realistic assessments of nuclear power plants against that goal. Rather than randomly pursuing additional analyses of various types, it is recommended that the emphasis be placed on the operating staff becoming familiar with the range and recognition of various accident situations and that they be trained in the proper corrective actions. It is felt that this type of activity which stresses prevention will yield the highest returns in terms of overall plant safety.

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 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 a structure offset potential increases in risk that many materialize from incidents such as inadvertent operation or the concentration of hydrogen in the filtering apparatus?

RESPONSE TO QUESTION #6

Nuclear reactor plants already have substantial ability to limit fission products released to the environment following accidents. Construction of a containment filter vent system capable of accomodating severe accident fission product release would be justifiable as safety enhancement only if it can be demonstrated that the accident is of sufficient likelihood and containment integrity is likely to be lost. Since current industry studies indicate that typical containment integrity will be maintained even without hydrogen control systems, it appears that pursuit of containment filter vent systems which merely transport the location of the fission product problem to a less secure location would not be advantageous. Keeping in mind that the ultimate goal of a containment filter vent system is to help maintain containment integrity, it appears at this time that efforts in the area of controlling the hydrogen that threatens containment integrity would be more cost effective.

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?

RESPONSE TO QUESTION #7

The current licensing requirements for nuclear power plants include incorporation of systems that are designed for collecting and controlling post accident hydrogen. These systems were sized based on hydrogen generation rates that were considered reasonable prior to the TMI event. The TMI event would have resulted in hydrogen generation consistent with current assumptions if safeguard systems had been allowed to function as intended. Significant improvements have been made in the area of operator training and procedures in an effort to eliminate recurrence of the problems encountered at TMI.

Large hydrogen generation rates would be difficult to handle with conventional hydrogen control equipment and therefore burning has been considered a candidate for hydrogen control. Burning is not a comfortable solution in that it is difficult to define the system's performance and the impact of the burning on equipment within the containment. A multi-faceted approach to the hydrogen control problem is likely to constitute an optimum solution. For example, realistically sized hydrogen removal systems used in conjuction with a systematic elimination of hydrogen igniting sources may result in the best solution to the problem. Risk studies may show that it is better to remove hydrogen from the containment over a period of several hours with power secured from all equipment in the containment capable of igniting the hydrogen than to burn the hydrogen as it is produced, especially if the production rate can not be defined.

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?

RESPONSE TO QUESTION #8

No. Containment inerting for this low probability type of occurence may not result in a net risk reduction for all nuclear power plants. The limititations and restrictions imposed on operations could increase the likelihood that minor events may develop into a situation for which inerting would then be useful. Before determining a specific approach, a risk assessment analysis must be made.

POOR ORIGINAL

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?

RESPONSE TO QUESTION #9

Combustion Engineering believes that discussions as to core retention systems are totally premature. The current emphasis, as noted previously, should be placed on accident prevention and mitigation via operator training. In addition, before such designs could be realistically addressed, more information would be needed regarding the actual threat posed by the core debris and its interaction with the vessel materials and any proposed core retention system.

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?

RESPONSE TO QUESTION #10

The increased radioactivity in systems normally functioning with much lower levels will depend, to a large extent, on the operating guidelines and the allowable alternative systems. Possible design changes could not be made judiciously until much more data is available including determination of new activity levels.

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?

RESPONSE TO QUESTION #11

More extensive operator training and improved tools for operating, such as improved procedures and improved control room information display equipment are being developed as part of the TMI action plan and should significantly improve the effectiveness of the operator in dealing with an emergency situation. Improving operator effectiveness will also affect minimum manning requirements. An evaluation of the effectiveness of these actions should be part of rulemaking in this area.

Previous difficulties appear to be in assuring that the operators recognize the tasks that need to be performed. The emphasis in operator training should be on the understanding of the processes involved as opposed to the strict literal compliance with procedures under all circumstances.

Increasing the reliability of emergency cooling or decay heat removal capability should be considered at the point when quantitative objectives of overall risk reduction are established so that an appropriate cost/benefit analysis can be used to evaluate possible system changes.

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?

RESPONSE TO QUESTION #12

It is noted that this is one of the few places in the Advanced Notice of Rulemaking where it is indicated that prevention is a part of the solution to the problem of degraded cores. Combustion Engineering is supportive of this approach.

We have evaluated alternate self-contained decay heat removal systems in the past and will continue to do so. Most current systems and equipment for removing decay heat during normal operations do not serve the reactor vessel and core region directly but rely on the Reactor Coolant System to provide a heat transfer circuit. New add-on systems, while solving some problems, could introduce others. The nuclear power plant has been designed with a complement of safeguard systems that can accomodate many accidents and failures including those things that will make the reactor coolant system heat transfer circuit inoperable. It appears that upgrading these systems and upgrading the operator's ability to use them could have more value than adding a new system with a duplicate function and making it self-contained.

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?

RESPONSE TO QUESTION #13

Current makeup and purification systems that are associated with normal service in the plant are not intended to be employed in an accident situation in such a way that they would require a leak-tight building. Make-up systems can be operated so that they do not encounter highly contaminated fluids. Purification systems in this category are not sized to provide service during an accident. Providing leak-tight buildings for these systems and thereby increasing the likelihood of contamination as a result of use during an accident is not clearly beneficial. In the case of the purification system, the limited performance available is not likely to be of much value.

QUESTION #14

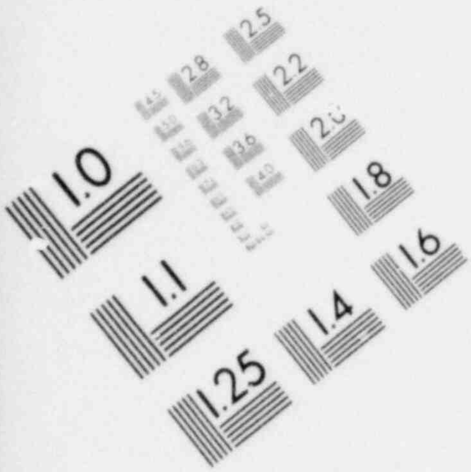
What design, quality and seismic criteria would you recommend for any additional systems to prevent the potential breeching of containment such as systems for controlled filtered venting, hydrogen combustion control, and core retention mentioned in previous questions? Do you favor evaluating designs 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 redundancy, diversity, testability, inspectability, and structural design limits (including seismic requirements).

RESPONSE TO QUESTION #14

Combustion Engineering is in agreement with the response developed by the AIF Committee on Reactor Licensing and Safety.

IMAGE EVALUATION TEST TARGET (MT-3)

6"



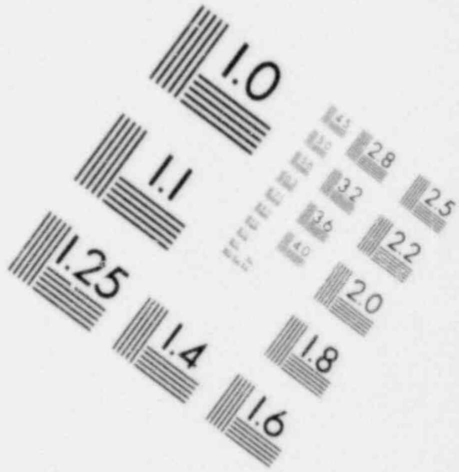
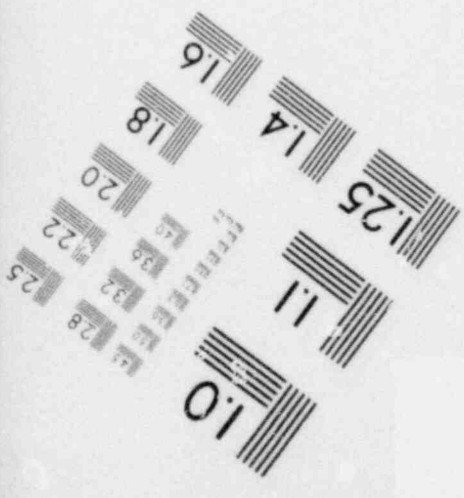
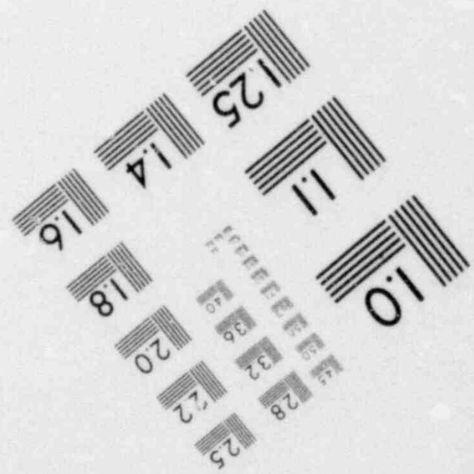


IMAGE EVALUATION
TEST TARGET (MT-3)





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?

RESPONSE TO QUESTION #15

Combustion Engineering is in general agreement with the response developed by the AIF Committee on Reactor Licensing and Safety.

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?

RESPONSE TO QUESTION #16

Combustion Engineering is in agreement with the response developed by the AIF Committee on Reactor Licensing and Safety.