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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 16, 1985

The Honorable Edward Markey, Chairman
Subcommittee on Energy Conservation and Power
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

Enclosed is an insert for page 39, line 827 of the record of the Subcommittee's April 17, 1985 hearing on the NRC Budget. This insert provides additional views of the NRC staff on the estimate of the crude cumulative probability of a core melt accident and was prepared under the direction of William Dircks, Executive Director for Operations.

Sincerely,

Edward Fay

Carlton Kammerer, Director
Office of Congressional Affairs

Enclosure:
As stated

cc: The Honorable Carlos J. Moorhead

Insert request for the record of:
April 17, 1985 NRC FY 86 Budget Authorization Hearing held by the Subcommittee on Energy Conservation and Power of the House Committee on Energy and Commerce

ADDITIONAL VIEWS OF THE NRC STAFF ON THE LIKELIHOOD AND SIGNIFICANCE OF CORE MELT ACCIDENTS

The Likelihood of Core-Melt

The staff estimate of the average core-melt frequency of 3×10^{-4} per year utilized in responding to Congressman Markey's first question is based upon six recent PRAs covering nine reactor units. These studies were originally performed by the owner-operators of the plants. In three cases, the NRC staff revised the owner-calculated accident frequency upward, and the average of 3×10^{-4} core-melt accidents per year reflects these NRC revisions. In each case, changes made in plant design and operation to reflect post-TMI orders were considered in the PRAs. The estimate is not inconsistent with inferences drawn from the frequency of events, known as precursors[1] to severe reactor accidents,

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1 Precursors refer to operational events that have taken place and which when taken in combination with the postulation of additional failures could lead to a core damage accident. For further insights on likelihood estimates derived from actual operational experience, reference is made to NUREG/CR-3591 (Vol. 1), entitled Precursors to Potential Severe Core Damage Accidents: 1980-1981 (A Status Report). Chapter 6 therein provides the analysis techniques used and sets forth an estimate of about 1×10^{-4} per reactor year as the derived mean frequency for a potential core damage accident. Chapter 7 briefly discusses that this estimate is likely to overstate the frequency of a large-scale core-melt accident. Other studies have suggested that the PRA techniques may overstate the expected frequency of the large-scale core-melt accident by factors of roughly 4 to 8 (e.g., Analysis transmitted May 24, 1984; from G Kolb, Sandia National Laboratory to M. Cunningham, Division of Risk Analysis, RES, NRC). It is believed, however, that the PRA techniques are not yet sufficiently developed to permit confidence in such a discrimination between the arrested core damage states and those that would proceed into a large-scale core-melt accident.

and the spectrum of results from other PRAs. The core-melt frequency for individual plants is not a static property. The likelihood of a core-melt accident varies with time as safety improvements are made, as the operating staff gains more experience with the plant, and because of plant aging phenomena typical of complex mechanical systems.

Under the assumption that 3×10^{-4} core-melt accidents per unit year remains the industry average in the future, one can calculate that the probability of one or more such accidents is 0.45 in a population of 100 plants serving for 20 years each. Another way of expressing the calculation is to say that there is 55 percent chance of no core-melt accidents, and 33 percent chance of one such accident, and a 12 percent chance of more than one accident over the 20-year period.

There are several reasons to believe that this assessment is pessimistic.

- Regarding the conservatism in the above PRA estimates, it is useful to keep in mind that no large scale core melt accident has been observed in the world history of operating experience with commerical nuclear power reactors. Based on a 1985 IAEA mid-year update, about 3,650 reactor years of experience has been accumulated.[2] Based on these data, we conclude

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2 Refer to: IAEA Reference Data Series #2 (data as of the end of 1984), Nuclear Power Reactors in the World; Contact: Robert Skjoeldebrand, Head of Reactor Engineering Section, Division of Nuclear Power, IAEA, Vienna, Austria.

with 95 percent confidence that a frequency of full scale core meltdown would be less than 8.2×10^{-4} and with 50 percent confidence that a full scale core meltdown would be less than 1.9×10^{-4}. This operating experience, therefore, provides considerable evidence that if uncertainties calculated in current PRA analyses are taken into account rationally, these should not exceed the experience base in spite of a calculated result which indicates otherwise. For example, it would be very unlikely that a large scale core melt frequency of 3×10^{-3} per reactor year is a valid frequency to use as representative of a typical or industry average for the frequency of full scale core meltdown accidents.

- Limited allowances have been made for improvements in safety due to the accumulation of experience in reactor operations. Both the NRC and the industry are continuously evaluating plant operating experience to identify improvements in design and operation.

- No allowance has been made for improvements in safety that may be found by performing additional PRAs, which often identify changes in design or operation that would lower vulnerabilities to core-melt. The NRC is implementing a severe-accident policy that would entail screening all operating plants for prominent vulnerabilities, called outliers, and rectifying these vulnerabilities. Each of the probabilistic risk assessments done to date has identified areas where risk can be reduced through plant design or operational procedure improvements, and in most, if not all cases, improvements have been made.

- PRAs may exaggerate the likelihood of accident vulnerabilities they find. For example, they routinely underestimate the potential for operators to arrest accidents in the early stages, or as was the case at TMI-2, even after core melt had begun.

There are two factors that could tend to cause the estimated average core-melt frequency to be too low:

- No allowance has been made for wear-out effects that might cause the plant's accident vulnerability to increase. Wear-out trends thus far have been more than compensated by improvements such as the TMI orders and other lessons of experience, but we cannot be certain this will always be the case.

- All PRAs are incomplete. Further intensive study might show that one or more of the six PRAs the staff used to develop the estimate missed a subtle but significant vulnerability.

On balance, the NRC feels that the current projection of accident likelihood is more likely to be pessimistic than optimistic especially because of the continuous process of learning from operating experience.

It is worth noting that it is the few plants with particularly prominent vulnerabilities, or outliers, that control the industry average accident frequency. This effect is a strong reason to focus attention on the search for outliers; that is, plants that may be vulnerable to a specific sequence that

can lead to severe core damage. We believe our regulatory process is aimed at continually reducing the likelihood of severe reactor accidents by requiring plant-design improvements (e.g., the ATWS improvements), improved operator training and improved management of plant operations. Such activities are a continuing part of our regulatory process.

In any event, as pointed out by Commissioners Bernthal and Asselstine during testimony before this subcommittee, it is inappropriate to focus regulatory applications of PRA on the bottom line numbers resulting from PRA evaluations. The staff is of the view, as are most PRA practioners, that one of the least reliable products of PRA is its bottom line estimate of risk. On the other hand, the major strengths of PRA are the disciplined methodology and logical framework for examining the safety of nuclear power plants and the unique insights it offers of the strengths and weaknesses of a plant's design. It is these strengths of PRA that should be emphasized in the regulatory process.

If Core Melt Is Likely How Can Plants Be Safe Enough?

If there is a distinct possibility of a severe core-damage accident in the next 20 years, are reactors safe enough? Overall we believe the answer is "yes." Reactors are safe enough to protect the public from the radiological releases that might result from reactor accidents. Only a small fraction of the large-scale core-melt accidents are predicted to have major off-site consequences, since only those accidents that involve the early failure or bypass of the reactor containment structure could result in significant releases of radio-activity. Research results on severe accident technology are showing that

containment structures have somewhat more safety margin than previously thought and that somewhat less radioactivity will be released from containment structures in the event of a core-melt accident than had previously been postulated.

The Three Mile Island involved an accident in which the reactor core began to melt, yet most of the radioactive materials released from the fuel in the accident remained within the plant (mostly inside the containment). Recent analyses of core-melt challenges to containment buildings show that even if the TMI core had melted down completely, and melted through the reactor vessel, the TMI containment would have held.

This is not to say that there are no accident sequences that could breach containment. Analysis shows that there are some kinds of accident sequences that could cause failure of any containment design, although a substantial fraction of the radioactive material would be trapped within the plant.

Still, we can be confident that for the great majority of core-melt-accident scenarios, containment and other features will be very effective in limiting the offsite radiological consequences, and will be fairly effective in moderating the consequences in almost all situations.

Our research into PRA, the effectiveness of containment systems, and the magnitude of radiological releases when containments fail is advancing rapidly today. We cannot be certain what future developments will show. However, the trend is toward lower estimates of the radiological consequences of accidents.