APR 27 1988

The Honorable Lloyd Bentsen
United States Senator
961 Federal Building
Austin, Texas 78701

Dear Senator Bentsen:

Your letter of February 17, 1988 requested pertinent information we might have regarding a constituent's concern over the safety of boiling water reactor (BWR) Mark I containment designs, and the Fermi 2 nuclear plant in particular. The enclosed discussion responds to this matter.

As an aside, the constituent's letter referred to the Fermi 2 plant and "the fifteen other nuclear power plants in Michigan." There are in fact five licensed nuclear plants in Michigan, one of which (Fermi 2) is a Mark I design.

Sincerely,

Original signed by
Victor Stello

Victor Stello, Jr.
Executive Director
for Operations

Enclosure:
As stated

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SEVERE ACCIDENT RISK AT MARK I REACTORS

Containment structures are an integral part of US reactor designs in that they form one level of a structured tiered approach to public safety known as defense in depth. Defense in depth is the process implemented by the Nuclear Regulatory Commission to ensure that multiple levels of assurance and safety exist to minimize risk to the public from nuclear plant operation.

A primary level of assurance are those activities to ensure that the plant is designed and constructed to high quality standards. Guidance on plant design is provided in the Code of Federal Regulations and specified in the General Design Criteria (GDC). Specific information is provided in the NRC's Standard Review Plan (SRP) which details acceptable methods for complying with the requirements established in the GDC.

Early in the development of commercial nuclear power it was recognized that nuclear reactors could not be expected to be immune from various failures and malfunctions, regardless of the quality of design, construction, and operation. Therefore, a further level of defense was established in that the plants were required to be designed for successfully coping with various equipment failures, transients and postulated accidents. The scenarios for postulated accidents, to which all plants (including Mark I reactors, the design in question) are designed to adequately respond, are known as design basis accidents.

Design basis accidents were chosen to represent a wide spectrum of plant problems, some of which were expected to be experienced in the plant lifetime (such as failure of power systems), as well as events considered to be quite unlikely (such as major ruptures of piping systems). The requirements and capabilities of plant safety systems necessary to prevent these design basis accidents from leading to unacceptable radiological releases, as well as guidelines for judging the acceptability of the analytical results in response to these hypothetical scenarios, are specified in NRC regulations. The plant designs resulting from these regulations incorporate multiple safety systems and back up safety systems to protect the reactor during the design basis accidents even with failures of some of these safety systems.

Notwithstanding the above, additional margins are required in the plant design to protect the public even in the event of very unlikely accidents. The reactor containment provides an additional level of safety. Design basis accidents for containment reflect a number of conservative arbitrary accident sequences developed from postulated events. For example, the containment structural design is based upon the effects of a concurrent earthquake and a rupture of major reactor coolant system piping. For the containment design, some independent failures of the safety systems are arbitrarily assumed to occur simultaneously with the occurrence of the accident they are intended to control. While the purpose of other safety systems is to shut down the reactor fission process and provide emergency cooling water to the reactor core, the containment has a required function of providing an essentially leaktight barrier to "bottle up" any radioactive material released to the containment through any rupture or break in the reactor coolant system. Consequently, the assessment of containment design adequacy assumes the postulated release of radioactive material to the containment atmosphere irrespective of the performance of the core cooling safety systems.

In summary the analysis of nuclear reactors, including Mark I reactors, against the acceptance criteria of design basis accidents provides considerable assurance that serious accidents will not occur and even if they should occur that significant radiological releases will be confined within the containment structure.

For the last several years, as part of the NRC's efforts to continually evaluate and improve power plant safety, we have been studying the likelihood and consequences of extremely low probability accidents with attendant higher estimates of core damage and higher radiological releases from the core. This class of accidents is beyond the existing design basis and is known generally as severe accidents (sometimes referred to as core melt accidents). The first comprehensive study of severe accidents was the Reactor Safety Study (WASH 1400); this study used an analytical method known as probabilistic risk assessment (PRA) to evaluate the risks posed by nuclear reactors.

The type of accidents studied in this evaluation are basically those where multiple backup safety systems are assumed to fail, eventually resulting in substantial damage to the reactor core and considerable release of radioactive material inside the containment. Depending on other failures and containment behavior, significant radiological releases into the environment could conceivably occur.

More detailed PRA studies have been conducted since the publication of WASH 1400 to better understand the probability of these unlikely events and also to better predict the magnitude of potential radiological releases into the environment, given a containment failure and attendant consequences. Considerable work has also focused on the behavior of reactor containments following a severe accident where molten reactor fuel could potentially melt through the reactor vessel. Results of such studies have generally confirmed the low risk to the public from severe accidents.

The most recent thorough study of severe accidents has been published in draft NUREG-1150 (February, 1987) "Reactor Risk Reference Document." This study included the analyses of different reactor designs; one of those studied was a BWR Mark I reactor, namely the Peach Bottom nuclear plant. The Peach Bottom plant as a Mark I design is similar to Fermi in reactor and containment design. The results of this study indicated an estimated mean frequency of core damage (i.e., core melt) to be approximately one chance in 100,000 per year of operation.

These results are consistent with NRC's belief that core melt accidents are very unlikely. Draft NUREG-1150 also investigated the probability of early containment failure following a core melt. It is this issue which has attracted considerable attention to Mark I reactors since the study concluded that there is large uncertainty regarding the probability of early containment failure for these extremely unlikely accidents. As a result of the study documented in draft NUREG-1150 it was concluded that the containment failure probability for Peach Bottom, a Mark I reactor, could range from 10 to 90 percent, albeit for highly improbable accidents.

Even allowing the large uncertainties which result in a high upper value for containment failure, the draft NUREG-1150 study estimated that the probability of a large reactor accident that results in 1 or more early fatalities ranged from 1 in one million to 1 in one billion. Given a severe accident, the probabilities of very high radiation exposure and the distances over which they would occur were also estimated to be reasonably small. The risk levels for Fermi 2 or other Mark I reactors would of course depend on its actual core melt probability, containment behavior, the local demography, and could vary somewhat from the results presented in draft NUREG-1150. The results of this and related studies do, however, support our overall conclusion of low severe accident risk of nuclear reactors.

While we believe that severe accident risks are low at operating nuclear plants, our goal is to pursue additional activities to achieve even lower levels of public risk. To assure that our risk conclusions are applicable to all operating units, a number of programs are going forward to assess severe accident likelihood and consequences. These programs include plant specific studies by each utility to determine any severe accident vulnerabilities, both from the perspective of accident frequencies and from containment performance following a core melt. Any problems will be dealt with if identified. This program is known as the individual plant examination (IPE) program which is expected to provide further assessments of severe accidents on a plant specific basis, so that appropriately low risk levels can be maintained. Even though the risk posed by Mark I reactors has been found by past studies to be acceptable, the NRC has continued to investigate means to improve the containment performance for those plants, and thus improve overall plant safety. The impetus for this continued research stems from concern over the uncertainty regarding the Mark I containment performance, as reflected in draft NUREG-1150, as well as a belief that reasonable plant improvements can be identified.

To that end the NRC has initiated the Mark I Containment Performance Program. The principal objective of the program is to evaluate technical issues associated with core melt accident phenomena and evaluate potential improvements to the Mark I design. Among the areas of improvement being considered are 1) combustible gas control, 2) containment sprays, 3) containment venting, 4) core melt debris control and 5) improved emergency procedures and training.

This program is being implemented on a high priority basis and should provide a firm and timely basis for deciding an appropriate course of action. An interim report on these activities is due to the Commission in April 1988. This report will address differences in existing risk studies and indicate whether existing analyses justify changes on the Mark I containment systems or operating procedures in the near term. A final report to the Commission is scheduled for August 1988. In summary, based on existing studies, the calculated failure probability for Mark I reactors in the event of certain severe accident scenarios does not in itself constitute an unacceptable risk to the public health and safety. Nonetheless the Commission, consistent with its defense in depth philosophy, is pursuing methods for improving containment reliability and reducing overall risk.

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Design basis accidents were chosen to represent a wide spectrum of plant problems, some of which were expected to be experienced in the plant lifetime (such as failure of power systems), as well as events considered to be quite unlikely (such as major ruptures of piping systems). The requirements and capabilities of plant safety systems necessary to prevent these design basis accidents from leading to unacceptable radiological releases, as well as guidelines for judging the acceptability of the analytical results in response to these hypothetical scenarios, are specified in NRC regulations. The plant design guidance required as a result of this approach results in the incorporation of multiple and backup safety systems which will protect the reactor during the postulated failures of these various protection devices.

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ENCLOSURE

← Change last sentence to:

The plant designs resulting from these regulations incorporate multiple safety systems and back up safety systems to protect the reactor during the design basis accidents even with failures of some of these safety systems.

3/14/ Thadani
3/25

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

THADANI

ACTION

EDO PRINCIPAL CORRESPONDENCE CONTROL

FROM:
SEN. LLOYD BENTSEN

DUE: 03/~~14~~ 25/88

EDO CONTROL: 003530
DOC DT: 02/17/88
FINAL REPLY:
SPLB 3/9/88

TO:
OCA

FOR SIGNATURE OF:
EXECUTIVE DIRECTOR

**** GREEN ****

SECY NO: 88-145

DESC:
ENCLOSES LETTER FROM RITA J. ECKHARDT CONCERNING SAFETY OF NUCLEAR POWER PLANTS IN MICHIGAN

ROUTING:
DAVIS

DATE: 02/29/88
ASSIGNED TO: NRR **CONTACT:** MURLEY

– Gary
– Murley
– Ted – Action

SPECIAL INSTRUCTIONS OR REMARKS:
REPLY TO AUSTIN, TEXAS OFFICE.
MARK ENVELOPE ATTN: ANNE MOSHER.

NRR RECEIVED: FEBRUARY 29, 1988
ACTION: ~~DRSP/CRUTCHFIELD~~ PMAS: Gillespie, Cecil/Cox

NRR ROUTING: MURLEY/SNIEZEK
MIRAGLIA
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CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-88-0145 LOGGING DATE: Feb 24 88

ACTION OFFICE: EDO

AUTHOR: L. Bentsen--Const Ref
AFFILIATION: U.S. SENATE

LETTER DATE: Feb 17 88 FILE CODE: ID&R-6

SUBJECT: Safety of nuclear power plants in Michigan

ACTION: Direct Reply

DISTRIBUTION: OCA to Ack

SPECIAL HANDLING: None

NOTES: Rita Eckhardt

DATE DUE: Mar 9 88

SIGNATURE: DATE SIGNED:
AFFILIATION:

Rec'd Off. EDO
Date 2-26-88
Time 3:30p

EDO---003530