July 20, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
)
HOUSTON LIGHTING & POWER COMPANY) Docket No. 50-466
)
(Allens Creek Nuclear Generating)
Station, Unit No. 1))
)

TESTIMONY OF GUY MARTIN, JR. ON BEHALF OF HOUSTON LIGHTING & POWER CO. ON DOHERTY CONTENTION 40 PART 100 RELEASES

Q. Please state your name and job position.

A. My name is Guy Martin, Jr. I am employed by Ebasco as Supervising Engineer of Envirosphere's Radiological Impact Assessment Department.

Q. Please state your educational background, work experience and professional qualifications.

A. A statement of my qualifications is attached to this testimony as Attachment GM-1.

Q. What is the purpose of your testimony?

A. My testimony will address Doherty Contention 40, in which it is alleged that:

> "...the Allens Creek site is unsuitable for the proposed nuclear plant, because the assumed fission product release from any accident considered credible will exceed the limitations of radioactivity dose to the low population zone stated in 10 CFR 100.11, (a) (1), (2), and (3).

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This Intervenor contends this because the actual release of radioactivity from the Three Mile Island accident exceeded calculated release for any accident considered credible by a factor of 22, using the calculation suggestions of Regulatory Guide 1.4. The proposed ACNGS and Three Mile Island are sufficiently similar in design such that the miscalculation in the ill-fated reactor's case is the same for the subject of these proceedings, in regard to source terms, and other factors.

This contention is particularly relevant to ACNGS construction license proceeding because the Applicant's proposed NPSS will use the largest BWR core attempted, with the highest power core density, and greater minimum critical heat flux ratio than any functioning BWR plant. Construction of the plant at the proposed site will injure Intervenor's health and safety interest by exposing him to radiation in excess of the guidelines of 10 CFR 100.11."

Q. Are the TMI and Allens Creek designs substantially similar as Intervenor Doherty alleges?

A. No. TMI is a Babcock and Wilcox PWR with a reactor and containment design substantially different from the General Electric BWR at Allens Creek. These differences will result in different isotopic source terms and different containment release rates during an accident. These differences are reflected in the fact that the NRC uses different Regulatory Guides to evaluate the radiological consequences of a design basis accident (LOCA) for BWR's and PWR's (Regulatory Guide 1.3 addresses BWR's; Regulatory Guide 1.4 addresses PWR's).

Q. The Intervenor has based his allegation that radiation releases for TMI were 22 times greater than estimated on Board Notification BN-79-23. Would you discuss your evaluation of BN-79-23?

A. BN-79-23 estimated the radioactive releases from TMI to be 13 million curies of Xenon-133. The concern raised by the notification was that it had been previously estimated on the basis of Regulatory Guide 1.4 that the amount of Xe-133 released during a design basis accident would be 600,000 curies. Thus, the actual release of this isotope was reported to have been substantially in excess of that predicted under the Regulatory Guide.

Q. Were 10 CFR 100 limits in fact, ever exceeded for the Three Mile Island plant?

A. No. A detailed study in NUREG-0558 indicated that an average dose of only 1.5 millirem was received by the population surrounding TMI during the entire course of the incident. The study also indicated that the maximum estimated dose to one individual outside the exclusion area was less than 100 millirem, or 1/250th of the 10 CFR Part 100 limits.

Q. Please explain how the number of curies of Xenon-133 could be larger than obtained under Regulatory Guide 1.3 or 1.4 analyses, yet result in doses still acceptable and lower than 10 CFR 100 limits.

A. Because it assumes immediate release of fission products to the containment atmosphere, the model used by Regulatory Guides 1.3 and 1.4 is a conservative estimate of doses that could result from a design basis accident with a high degree of core damage. This assumption results in a higher ratio of shorter lived high energy gamma emitting noble gases such as Krypton-88 being released. At TMI, releases to the environment did not occur until several hours into the incident, thus allowing a significant decay time for the short lived high energy isotopes and resulting in a higher ratio of the low energy gamma emitter, Xenon-133, being released. Although the total number of curies released from TMI exceeded that of the Regulatory Guide source term, the effect of the TMI source term was less because the high energy gamma emitting isotopes are the major contributors to the total dose.

Q. Has the NRC calculational method, that is Regulatory Guides 1.3 and 1.4, with respect to post-accident

doses, been revised since the TMI incident to reflect different calculational techniques due to the higher than estimated release of XE-133 at TMI?

A. No.

Q. Would you please discuss the method used for determining Allens Creek's compliance with 10 CFR 100 limits?

A. The analytical procedure used to determine such compliance is detailed in Regulatory Guide 1.3 and in the Allens Creek PSAR Section 15.1.39. Basically the method assumes that 25 per cent of the iodine and 100 per cent of the noble gas core inventories developed during equilibrium maximum full power operation are immediately released to the containment and are available for leakage from containment. Next the containment is assumed to leak at the maximum rate allowed by the plant technical specifications for the duration of the accident. Then plant specific values for meteorology are applied to determine the resulting doses.

Q. Does Allens Creek meet the 10 CFR Part 100 limits?

A. Yes. Using the methodology of Regulatory Guide 1.3 the resultant doses are given in PSAR Table 15.1.39-3 and summarized below:

1) The individual 2 hour dose at the exclusion area is 4.9 rem whole body and 150 rem thyroid

2) The individual accident duration dose at the low population zone is 1.2 rem whole body and 71 rem thyroid.

Part 100 establishes limits of 25 rem whole body and 300 rem thyroid for either an individual two hour dose at the exclusion area boundary or an individual accident duration dose at the low population zone. The above doses are well within the 10 CFR 100 limits.

In addition, it should be noted that gross fuel failure to the point of melting would be required to achieve the fission product inventories in containment that are assumed in Regulatory Guide 1.3.

An analysis has also been completed for a less conservative amount of fuel damage and resultant fission product release that would occur during a design basis accident. Under this more realistic analysis (PSAR Table 15.1.39-3) the following doses would be received:

1) Individual 2 hour doses at the exclusion area of 5×10^{-6} rem whole body and 3.4×10^{-5} rem thyroid.

2) Individual accident duration dose at the low population zone of 9.8×10^{-7} rem whole body and 1.8×10^{-5} thyroid.

Q. How was the large amount (approximately 13 million curies) of Xe-133 released to the environment at TMI?

A. According to the Kemeny Commission Report, most radioactivity escaping from TMI-2 to the environment was in the form of fission gases transported through the reactor coolant let-down/make-up system into the auxiliary building and through the building filters, then to the vent header and to the outside atmosphere. The major release of radioactivity on the morning of March 30 was caused by the controlled, planned venting of the make-up tank into the vent header.

Q. Is a scenario for radioactivity release of this type included in the dose assessment analyses of Regulatory Guides 1.3 or 1.4?

A. No. Containment isolation is assumed to occur, and the containment is then assumed to leak at the maximum allowable plant technical specifications leak rate.

Q. Are any design improvements incorporated in the Allens Creek design that would preclude the occurrence of a release such as that which occurred at TMI?

A. First of all, ACNGS does not utilize a coolant let-down/make-up system, as provided at TMI, so a release path of this type is not possible at ACNGS. Moreover in response to the TMI incident, design modifications were developed for the containment isolation system. These modifications are required by NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits And Manufacturing License." As detailed in Sections II.E.4.2, II.E.4.4, and III.D.1.1 of the ACNGS PSAR, Appendix O, Allens Creek has incorporated all suggested modifications, such as containment isolation for non-essential systems, that were not already part of the plant design.

These design modifications were developed in response to the TMI incident to contain radioactive contaminants within the containment building. Incorporation of these modifications will help assure that releases such as occurred at TMI will not occur at Allens Creek.

Q. What are your conclusions?

A. The release mechanism which occured at TMI cannot be duplicated at Allens Creek, nor will the estimated doses from a design basis accident at Allens Creek exceed the 10 CFR Part 100 limits.

Attachment GM-1

Guy Martin, Jr.

I received a ME from the City College of the City of New York in 1974. I received a MS in Nuclear Engineering from Polytechnic Institute of New York in 1976. I have been employed by Ebasco since 1973. I have eight years' experience in preparation of Safety Analysis and Envirionmental Reports sections dealing with the impact analysis of toxic chemicals and radiological releases. Such analyses are performed for both routine plant operation and accident conditins. In this regard, I conduct reviews of radwaste handling systems, air handling and cleanup systems and estimate radionuclide releases from plant effluents and calculate and calculation of implant dose rates to equipment and personnel from air borne radionuclide exposure and I have performed ALARA of air cleanup systems. I have performed safety reviews of engineered safety systems, which included a review of the specifications and operation from the radiation protection viewpoint and have provided design recommendations based on assessed radiological doses and established nuclear safety criteria. I have performed analyses of the transport of toxic chemicals postulated to be released accidentally and calculated the concentration in critical locations of the power plant. I have provided technical feedback to the designers on required protection levels. In this regard I have assisted in making the determination of toxic chemical detector specifications based on worker and equipment protection criteria.

I have responsibility for the preparation of radiological envirionmental surveillance programs wherein I have prepared detailed surveillance program description based on site specific critical pathways of exposure. I have established the sampling requirements of the frequency and types of analyses to be performed.

I have also participated in preparation of a study regarding the establishment of a comprehensive data base regarding high level waste disposal and I have supervised the health physics activities related to decontamination work at the Kellex Laboratory.

Prior to my employment with basco, I was employed as a cost analyst by Equitable Life Assurance Socity of the US.

I am a member of the American Society of Mechanical Engineers, a member of the Health Physics Society, and a member of the American Nuclear Society, and Intern Engineer of New York State. I have written the following publications:

Marti., G. and J. Thomas 1978. Meeting the dose requirements of 10CFR100 for site suitability and general design criteria 19 for control room habitability: a parametric approach. Transactions of American Nuclear Society 24th Annual Meeting. Vol. 18.

Martin, G. D. Michlewicz and J. Thomas 1978. Fission 2120: a program for assessing the need for engineered safety feature grade air cleaning systems in post-accident environment. Proceedings of 15th DCE Nuclear Air Cleaning Conference.

Letizia, A. P., G. Martin and J. F. Silvey 1979. - Implications for nuclear facilities of changes being initiated in the NRC standard atmospheric diffusion model. Proceeding of the 41st Annual Meeting of the American Power Conference.

Bhatia, R. K., Mauro, J., Martin, G. - Effections of Containment Purge on the Consequences of a Loss-of-Coolant Accident. Transactions of the American Nuclear Society 1980 Annual Meeting.