12/7/81

DOCKETED
USNRC

'81 DEC 10 P4:18

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

IN THE MATTER OF	)	
	)	Docket No. 50-155-OLA
CONSUMERS POWER COMPANY	)	(Spent Fuel Pool
	)	Expansion)
(Big Rock Point Nuclear Power Plant)	)	

TESTIMONY OF ROGER W. SINDERMAN
CONCERNING
O'NEILL CONTENTION IIE-4

My name is Roger W. Sinderman. I am employed by Consumers Power Company as Director of Radiological Services. My business address is 1945 West Parnall Road, Jackson, Michigan. I joined the Company on May 9, 1966, and I have held various positions of increasing responsibility since that date. My educational background and work experience are detailed in a Statement of Professional Qualifications that is attached to my testimony. I have had over fifteen years of experience in calculating and determining radionuclide source terms and the attendent radiation dose levels associated with routine operation of nuclear power reactors as well as the radiological consequences calculated for hypothetical accidents as required by the NRC's regulations in 10 C.F.R. Part 100.

Based on my educational background and work experience, I believe I am qualified to answer O'Neill Contention IIE-4:

> In the event of an accident which results in a substantial release of radioactivity from the expanded fuel pool, the containment building does not provide adequate shielding to protect the public health and safety.

Contention IIE-4 questions the adequacy of the three-quarter-inch steel containment shell at Big Rock Point to shield and protect the health and safety of the public from the effects of radiation. Gamma radiation is the only type of radiation emitted from radioactive materials in the spent fuel pool that exhibits the necessary characteristics and ability to radiate through or penetrate solid masses. Therefore, I presume that Mr. O'Neill is postulating a substantial release from the spent fuel pool to the containment of gamma-emitting radioactive material. Thus, my testimony addresses Mr. O'Neill's concern that the three-quarter-inch thick steel containment at the Big Rock Point plant may not adequately shield the public from the effects of such gamma radiation.

For purposes of maximizing the consequences of any accident involving a substantial release of radioactive material from the spent fuel pool, the spent fuel pool is assumed to be fully loaded in its expanded mode (441 spent fuel assemblies). Of these 441, 84 (a full reactor core off-load) is assumed to have decayed for only five days following a full year of reactor operation. This five-day decay period is the minimum time it can be reasonably expected to require to shutdown the reactor from full power operation, cool the primary coolant, remove the reactor biological shield and vessel head, and finally transfer all 84 fuel assemblies to the spent fuel pool. The remaining 357 fuel assemblies exhibit decay after reactor core off-load of one year through seventeen years in increments of 1/4 core (21 assemblies) added per year, <u>i.e.</u>, 21 x 17 = 357.

With the spent fuel pool filled in this manner, I will then assume, for purposes of this contention, a worst-case, hypothetical, nonmechanistic, instantaneous loss of all cooling water in the spent fuel pool. In addition, this assumption is combined with an assumed inability, for unspecified reasons, to refill the pool or to provide any mode of cooling. Further, all heat generated within each fuel rod is assumed to remain in that rod, resulting in the overheating of all stored fuel and consequent fission product release. In reality, oxidation and failure of fuel or cladding due to overheating with a resultant release of fission products is likely to be limited to freshly off-loaded fuel exhibiting large decay heat. Based on the foregoing set of unrealistic assumptions, a "substantial release of [gamma-emitting] radioactivity from the expanded fuel pool" would occur.

Release of fission products is assumed to be as given in TID 14844, which applies to a loss-of-coolant accident (LOCA) in a nuclear reactor at full-power operation. During the course of a LOCA, the fuel would be operating at an elevated temperature and would contain significantly greater quantities of fission products, thereby reaching higher temperatures than would be the case with respect to the accident scenario assumed for purposes of my testimony (particularly for older, more highly decayed fuel). The latter accident scenario is much less likely to produce a fission product release of this magnitude. Nevertheless, as a conservative assumption, I have used TID 14844 as the source term for the accident scenario assumed for purposes of my testimony. Thus, 100% of the noble gases contained in all the fuel in the pool are assumed to be released to the containment atmosphere along with 50% of all halogens and 1% of the remaining solid fission products. The release begins

36 minutes after the water loss occurs. This 36-minute time period is the minimum period for the most radioactive fuel rod of the freshly off-loaded core to reach cladding failure temperature. Total fuel melt of the freshly off-loaded fuel and, hence, the maximum dose rate, is not reached for several hours. Containment isolation is achieved and is necessary to maintain the source of radiation to expose an individual at the site property boundary from gamma radiation through the containment wall.

Based on the postulated spent fuel pool loading, the postulated nonmechanistic water-loss accident, and the postulated source terms described above, I have made dose calculations taking into consideration the thickness of the containment shell, the shielding afforded by air between the containment and the distance to the nearest site property boundary (2640 feet for overland sectors and 206 feet for the nearest shoreline sector). Radiation attenuation caused by other structures on site, including structures within the containment building, and by forest growth, though considerable, was neglected due to its highly varying nature as a function of direction.

The dose to any hypothetical individual located at the site boundary in overland directions is less than 5 millirems in two hours. The nearest actual residence is located at 1 mile, at which the dose is 0.0024 millirems in two hours. The nearest public highway, US-31, is 2760 feet at its closest approach to the plant. At this location, the dose is 4.4 millirems in two hours. These values are negligible when compared to the 25,000 millirem limit established for accident conditions in 10 C.F.R. 100.

The nearest distance to the shoreline from the containment building is 206 feet. The shoreline at this location is not used for recreational swimming and boating due to its rocky features. Fishermen do, however, from time to time, fish from the shoreline near the plant discharge canal as well as from boats offshore. Hence, radiation doses to members of the public using Lake Michigan resulting from the accident described herein are limited to these fishermen.

Given the assumptions of the accident scenario assumed for purposes of my testimony, a radiation dose of 25,000 millirems over a two-hour period occurs at a distance off-shore of 272 feet (478 feet from the containment building centerline). Fishermen located at distances less than 272 feet off-shore are in a zone in which the limits of 10 C.F.R. 100 are exceeded. However, ample time is available, at least 60 minutes, to warn and/or evacuate any persons fishing along the discharge canal or elsewhere within the area of interest. Evacuation would be effected by the Big Rock Point plant security force.

For both overland and shoreline sectors, well over 90% of the dose in the two-hour period results from the full core off-load with only five days of decay. Less than 1% of the dose results from fuel having decayed more than three years and, as such, from the fuel which is the object of this proceeding.

Based upon the above analysis, the shielding afforded by the Big Rock Point containment building provides adequate protection to the health and safety of the public in the event of the accident postulated by Mr. O'Neill, i.e., one resulting in substantial releases of radioactive material form the expanded fuel pool.

STATEMENT OF PROFESSIONAL QUALIFICATIONS
FOR
ROGER WILLIAM SINDERMAN

EDUCATION:

B.S. Science Engineering, University of Michigan
M.S. Health Physics, University of Michigan
M.P.H. Health Physics, University of Michigan

EXPERIENCE:

June 1981 - Present — Consumers Power Company as Director of Radiological Services, responsible for all aspects of radiological control at Consumers Power Company nuclear facilities. These responsibilities include radiation protection for employees, environmental surveillance, radioactive waste, effluent control, and emergency planning.

1974 - 1981 — Consumers Power Company as Corporate Health Physicist, responsible for all aspects of radiological control at Consumers Power Company nuclear facilities. These responsibilities include radiation protection for employees, environmental surveillance, and effluent control.

1973 - 1974 (6-month period) — Consumers Power Company as Palisades Plant Health Physicist, responsible for radiation protection, effluent and environmental control at the Palisades Plant.

1971 - 1973 — Consumers Power Company as Environmental Health Physicist responsible for environmental radiological surveillance and control of radiological effluents from the Company's nuclear facilities.

1968 - 1971 — Consumers Power Company as Health Physicist, responsible for Big Rock Point Plant radiological control and Palisades Plant construction activities related to radiation protection.

1966 - 1968 — Consumers Power Company as Associate Engineer, General Engineer, and Chemical and Radiation Protection Supervisor at the Big Rock Point Plant, responsible for Plant radiation protection activities and various engineering tasks.

SOCIETIES:

Health Physics Society
American Public Health Association