UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-295
	)	50-304
(Zion Station, Units 1 and 2)	)	

NRC Staff Testimony on

Contention 2(g)

BY

Richard M. Lobel
Jack N. Donohew
Edward Lantz

Contention 2(g) is as follows:

> The Applicant's discussion of spent fuel boiling is inadequate in that (1) there is no consideration given to the possibility that the pool might boil and (2) there is no discussion of possible damage to fuel cladding or of the consequent release of radionuclides under such conditions; therefore, there is no assurance that public health and safety will not be endangered.
>
> In addition, the heat removal capacity of the Spent Fuel Pool Cooling Systems has not been shown to be adequate to support the expanded pool capacity.

Boiling, as used in this contention, is not clearly defined. There are several modes of boiling. At the conditions of the spent fuel pool with loss of cooling, the coolant surrounding the fuel rods would be in the nucleate boiling mode. Nucleate boiling is a highly efficient mode of heat transfer. In the pressurized water reactor core, at full power operation, a small number of the fuel rods normally operate in nucleate boiling. In a boiling water reactor core most of the fuel rods operate in nucleate boiling.

If the density of water bubbles were to increase at the surface of the fuel rod so that the mode of boiling changed from nucleate boiling to film boiling, the cladding temperature would increase significantly over that in nucleate boiling. However, because of the low heat flux of a fuel rod in a spent fuel pool (with all its power coming only from decay heat), such a fuel rod would not undergo film boiling. Therefore, no damage is expected to fuel rods due to boiling of the spent fuel pool.

Cumulative spent fuel pool experience as recent as June 1978 has shown that "no commercial water reactor fuel has yet been observed to develop defects while stored in spent fuel pools" at normal spent fuel pool conditions. (Reference 1). Also, available evidence cited in Reference 1 shows that a fuel rod which was already defected from operation in the reactor would not undergo further degradation. At the temperatures of fuel rods in a boiling spent fuel pool there should be no dissolving of the UO_2 pellets if exposed to fuel pool water through a cladding defect. Observations at Karlsruhe, West Germany showed no detectable dissolving of fuel pellets at normal spent fuel pool temperatures (Reference 1). We would expect that the increased temperatures due to loss of spent fuel pool cooling would not change this result.

Oxidation of Zircaloy cladding at boiling conditions can be assumed to be negligible based on data from Zircaloy 2 tubes exposed to treated Columbia River water (less pure than spent fuel pool water) at approximately 90°C*

Reference 1: A.B. Johndon, Jr., "Behavior of Spent Nuclear Fuel in Water Pool Storage", Battelle Pacific Northwest Laboratories, BNNL 2256 Sept. 1977.

481 050

(Reference 1). Extrapolation of this data to 100 years yielded a conversion of Zircaloy to oxide of less than 0.1% (clad wall thickness).

An indication of the ability of Zircaloy clad UO_2 fuel bundles to withstand boiling after reactor irradiation and subsequent spent fuel pool storage is the reirradiation of three fuel bundles of CANDU reactor fuel (Canadian reactor design). CANDU fuel rods are shorter than those used in the Zion reactor and have a thinner cladding and a slightly larger diameter. However, the cladding material is Zircaloy 4 which is the same as the Zion fuel rods. After a first irradiation to a low burnup, these bundles were placed in pool storage for 10, 9 and 5 years, respectively. The bundles were then reirradiated for approximately one month at conditions approximately those in a CANDU reactor. The CANDU reactor operates at nominal inlet and outlet temperatures of 510° and $594^{\circ}F$, respectively. The reactor pressure drops from 1630 psia the 1449 psia across the core.

These conditions are much more severe than those which could occur in a boiling spent fuel pool. The reactor conditions also would include a high energy neutron and gamma ray flux which would not be present in the spent fuel pool.

The peak power in these fuel rods was higher than that expected in the Zion peak power fuel rods and would therefore be at least an order of magnitude higher than the power of a fuel rod in the spent fuel pool. One would therefore expect the stresses in the cladding to also be higher than those which could be imposed on the cladding of the Zion fuel rods in a boiling spent fuel pool.

Nevertheless, it is reported (Reference 1) that following this period of reirradiation, no defects were found in any of the CANDU fuel rods.

In conclusion, the spent fuel pool boiling mode would be nucleate boiling. PWR fuel rods are designed to operate in the reactor core in nucleate boiling at heat fluxes which are orders of magnitude higher than those which could occur in the spent fuel pool. Therefore, failures of fuel rods in the spent fuel pool due to boiling would not be expected. Data also exists to show that a fuel rod defect would not be further degraded if boiling were to occur.

In regard to the possible release of radionuclides to the atmosphere it should be noted that the radioactivity of the additional spent fuel in the pool because of the pool modification would have decayed for several years. The volatile radioactive nuclides in the defective failed fuel would have, therefore, either decayed or diffused into the pool water. The remaining radioactivity in the spent fuel would then be non-volatile. For this activity, the leakage of activity from the fuel pin during pool boiling would not be significantly different from that at normal pool operations. Under normal conditions, experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months.

The predominance of radionuclides in the spent fuel pool water appears to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the spent fuel pool. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating conditions. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that fuel clad temperature is relatively cool. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the fuel rod, thereby tending to retain the fission products within the fuel rod.

In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the Licensees or discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or a Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

The conditions in the spent fuel during pool boiling which will affect the leakage of radioactivity from this additional spent fuel are not significantly different from the conditions in the pellet-cladding gap during normal pool operations. Based on the experience discussed above for normal pool conditions, we would not expect boiling of the pool to result in a significant increase, if any, in the leakage of activity from the additional spent fuel in the pool. Under normal pool conditions, any non-volatile radioactivity leaking from spent fuel into the pool water should remain in the pool water to be removed by the pool purification system. Under conditions of the pool boiling, this radioactivity may be entrained in water droplets in the air above the pool. These droplets will condense out on surfaces in the fuel building and a fraction of these droplets could be entrained in the building ventilation air flow. In the ventilation system, the droplets will condense out on the ducts or be filtered by the filtration system. The filtration system has prefilters, HEPA filters and charcoal filters. These filters will remove the water droplets and the radioactivity from the air until the pool cooling system and purification system is repaired or the hot spent fuel is returned to the reactor vessel.

Thus, it is our conclusion that fuel pool boiling, and its resultant effects on spent fuel stored therein, does not constitute a credible threat to public health and safety.

The following three sections of the NRC's March 29, 1979 Safety Evaluation for the proposed modification address the second part of this contention, which is on the adequacy of the heat removal capacity of the heat removal capacity of the spent fuel pool cooling systems:

2.2 <u>Spent Fuel Cooling</u>

> The licensed thermal power for each unit of the Zion Station is 3,250 MWt. The licensee plans to refuel both units 1 and 2 annually. This will require the replacement of about 64 of the 193 fuel assemblies in each core every year. Thus normal refuelings will take place at 6-month intervals. To calculate the maximum heat load, the licensee assumed that it would take 10 days after the reactor was shut down to complete the transfer of both 1/3 of a core in a normal refueling and full core in a full core offload. With these delay times, the licensee used the method given in American National Standard 5.1 to calculate 20.4×10^6 Btu per hour as the maximum heat load for an annual refueling and 35.0×10^6 Btu per hour as the maximum heat load for a full core offload.
>
> The spent fuel pool cooling system as described in Chapter 9 of the Final Safety Analysis Report consists of two pumps and two heat exchangers. Each pump is designed to pump 2,330 gpm (1.15×10^6 pounds per hour), and each heat exchanger is designed to transfer 14.9×10^6 Btu per hour from 120°F fuel pool water to 95°F component cooling water, which is flowing through the heat exchanger at a rate of 1.9×10^6 pounds per hour.
>
> As shown in Chapters 6 and 9 of the Zion Final Safety Analysis Report, there are seismic Category I sources of makeup water for the spent fuel pool. These are the refueling water storage tanks. There is one of these stainless steel lined, reinforced concrete, Class I structures for each units, and each one holds 389,000 gallons of water.

2.2.1 Evaluation

Using the method given on pages 9.2.5-8 through 9.2.5-14 of the NRC Standard Review Plan dated November 24, 1975, with the uncertainty factor K equal to 0.1 for decay times longer than 10^7 seconds, we calculated that the maximum peak heat load during the 33rd refueling (the one that fills the pool) could be 22.2×10^6 Btu per hour and that the maximum peak heat load for a full core offload that essentially fills the pool could be 41.4×10^6 Btu per hour. This full core offload was conservatively assumed to take place 6 months after the 30th refueling. We also determined that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pool from 868 to 2,112 is 5.4×10^6 Btu per hour. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools.

We calculated that with one pump operating with one heat exchanger, the spent fuel pool cooling system can maintain the fuel pool outlet water temperature below 102°F for the normal refueling. In the highly unlikely event that both spent fuel pool cooling systems were to fail at the time when there was a peak heat load from a full core in the pool and the water was at its maximum temperature, we calculate that boiling could commence in about 7 hours. We also calculate that after boiling commences, the required water makeup rate will be less than 85 gallons per minute. We find that 7 hours will be sufficient time to establish an 85 gallon per minute makeup rate.

2.2.2 Conclusion

We find that the present cooling capacity for the spent fuel pools at the Zion Nuclear Power Plant will be sufficient to handle the incremental heat load that will be added by the proposed modification. We also find that this incremenal heat load will not alter the safety considerations of spent fuel cooling from that which we previously reviewed and found to be acceptable.

These sections were prepared by Edward Lantz, and to the best of his knowledge they are true and correct. In order to calculate the maximum heat load for this evaluation he assumed that a full core offload would take place six months after the thirtieth refueling at the plant. The reason for this is after fifteen years of operation it is unlikely that a full core will be discharged to the fuel pool right after a refueling. Also, in this regard, it should be noted that a core that has operated only six months after its last refueling will have fewer fission products and actinides in it than one that has operated a full year. However, if we assume for extra conservatism that the core which is being offloaded ten days after a normal refueling does have a full inventory of fission products, the maximum possible heat load would be 51 x 10^6 Btu/hr. With both spent fuel pool cooling loops operating with this heat load in the pool the outlet water temperature from the spent fuel pool will be about 125°F. If one of the cooling loops were to be stopped, i.e., if a single failure occurred, the outlet water temperature would go up to about 170°F. If both of these cooling loops were to fail at the time of this peak heat load, the rate of increase of the average fuel pool water temperature would be about 11°F/hr. Thus, there would be about eight hours before boiling would commence. After this the maximum possible boil-off rate would be 105 gpm. Since these numbers are not significantly different from those given in the Safety Evaluation, the conclusion remains that the present cooling capacity is adequate for the proposed modification.

PROFESSIONAL QUALIFICATIONS

OF

RICHARD M. LOBEL

I am employed as a Reactor Engineer with the Division of Operating Reactors, USNRC.

I graduated from California State University at San Jose with a B.S. in Mechanical Engineering in 1966. I then began work as a Mechanical Engineer at Lawrence Livermore Laboratory, Livermore, California. At the same time I began work towards an M. S. degree in Mechanical Engineering at California State University at San Jose which I received in 1970. Since my masters degree I have taken an additional number of university courses in nuclear and mechanical engineering.

In my present work at NRC I am responsible for reviewing reactor fuel reload applications and other safety matters concerning operating reactors. My prime responsibility is in the areas of nuclear fuel thermal behavior and thermal hydraulic aspects of reactor behavior during steady state, anticipated transients and accidents.

Prior to my current assignment, I worked for three years in the Core Performance Branch where I was responsible for fuel rod thermal performance including reviews of computer programs used by fuel vendors for predicting fuel conditions during steady state and transient conditions, fuel densification and analysis of fuel rods during a Loss-of-Coolant Accident.

During the period of 1966 to 1973 while I was employed by Lawrence Livermore Laboratory I was responsible for the mechanical design of nuclear physics experiments.

I have been a lecturer on nuclear fuel behavior at two University short courses titled "Nuclear Power, Safety and the Public" and "Nuclear Power Reactor Safety Analysis."

PROFESSIONAL QUALIFICATIONS

OF

EDWARD LANTZ

An an Engineering Systems Analyst in the Plant Systems Branch, I am responsible for technical reviews and evaluations of component and system designs and operating characteristics of licensed nuclear power reactors.

I have a Bachelor of Science degree in Engineering Physics from the Case Institute of Technology and a Masters of Science degree in Physics from Union College and a total of 28 years of professional experience, with over 20 years in the nuclear field. My experience includes work on reactor transients and safeguards analysis, nuclear reactor analysis and design, research and development on nuclear reactor and reactor control concepts and investigations of their operational and safety aspects.

I have held my present position with the Commission since December 1975. My previous position, which I held for about two and one half years, was Project Manager in the Gas Cooled Reactors Branch, Division of Reactor Licensing, U.S. Nuclear Regulatory Commission, where I was responsible for the technical review, analysis, and the evaluation of the nuclear safety aspects of applications for construction and operation of nuclear power plants. For about ten years prior to that I was Head of the Nuclear Reactor Section in NASA. My section was responsible for the development and verification of nuclear reactor analysis computer programs, conceptual design engineering, and development engineering contracting. Prior to my employment with NASA, I was a nuclear engineer at the Knolls Atomic Power Laboratory for about six years, where I worked on the safeguards and nuclear design of the S3G reactors and the initial development of the nuclear design of the S5G reactors. Previous experience includes system engineering and electrical engineering with the General Electric Company and electronic development engineering with the Victoreen Instrument Company.

PROFESSIONAL QUALIFICATIONS

OF

JACK N. DONOHEW, JR.

My name is Jack N. Donohew, Jr. I am a Senior Nuclear Engineer in the Environmental Evaluation Branch in the Division of Operating Reactors, U.S. Nuclear Regulatory Commission (NRC). My duties include the review of rad-waste treatment systems and engineered safety feature ventilation systems for operating reactors.

I received a Bachelor of Engineering Physics Degree from Cornell University in 1965, a Masters of Science Degree in Nuclear Engineering from Massachusetts Institute of Technology in 1968, and a Doctor of Science Degree in Nuclear Engineering from Massachusetts Institute of Technology in 1970. I received my Professional Engineers License in Nuclear Engineering from the Commonwealth of Pennsylvania in 1974.

After graduation, I worked for Stone and Webster Engineering Corporation as an engineer in the Radiation Protection Group. I was responsible for estimating source terms, release rates and resulting doses for the Safety Analysis Report, Environmental Report and response to NRC questions for boiling water nuclear reactors. I was also responsible for shielding design for the reactor water cleanup system.

In February, 1973, I became a Power Engineer in the Process Engineering Group, Stone and Webster Engineering Corporation. I was lead engineer for the Shoreham Project and the equipment specialist for all nuclear plants for the containment iodine spray removal system, ventilation filter assemblies, and Boiling Water Reactor and Pressurized Water Reactor gaseous wate treatment system.

In June 1975, I jointed the Nuclear Regulatory Commission as a senior nuclear engineer in the Effluent Treatment Systems Branch, Directorate of Licensing. I was involved in rad-waste system licensing reviews of nuclear power plants. I have conducted generic studies of the degradation of charcoal absorbers in ventilation filter assemblies.

In December 1975, I jointed the Environmental Evaluation Branch in the Division of Operating Reactors.