UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In The Matter of	)	
	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454 OL
	)	50-455 OL
	)	
(Byron Nuclear Power Station,	)	
Units 1 & 2)	)	

AFFIDAVIT OF HARRY M. FERRARI

The attached questions and answers constitute my testimony in the above-captioned proceeding. The testimony is true and accurate to the best of my knowledge, information and belief.

Harry M. Ferrari

Harry M. Ferrari

Subscribed and sworn to
before me this 4th day
of June, 1982.

Paulette Slonska

Notary Public

PAULETTE SLONSKA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES MARCH 10, 1986
Member, Pennsylvania Association of Notaries

TESTIMONY OF HARRY M. FERRARI

CONCERNING DAARE/SAFE CONTENTION 6

Q. Please state your name, present occupation and present position.

A. My name is Harry M. Ferrari. I am employed by Westinghouse Electric Corporation as Consulting Engineer in the Engineering Department, Nuclear Fuel Division (NFD), Westinghouse Nuclear Energy Systems, Westinghouse Power Systems Company.

Q. Please describe your professional and educational background.

A. I am responsible for providing technical consulting to the Engineering Manager and staff on nuclear fuel technology. Prior to my appointment as a Consulting Engineer in 1973, I was Manager, Fuel Licensing, and I was one of the individuals responsible for reviewing the safety and licensability of the nuclear fuel for nuclear power generating plants. This included the design, construction, and operation of nuclear fuel for nuclear power reactors.

I graduated from Wayne State University in 1954 with a B.S. degree in Metallurgical Engineering. From 1954 to 1958, I attended the University of Michigan and received an M.S. and Ph.D. degree in Metallurgical

Engineering. From May 1958 to March 1960, I was employed by Westinghouse Electric Atomic Power Division as a Senior Engineer responsible for developing advanced UO_2 fabrication processes and cladding materials. During this period, I first became acquainted with UO_2 fuel technology and zircaloy and stainless steel clad technology.

From March 1960 until September 1961, I was assigned as a technical advisor to EURATOM in Brussels, Belgium, and FIAT in Turin, Italy. The purpose of this assignment, which was at the request of the U.S. Atomic Energy Commission, was to advise EURATOM on joint AEC/EURATOM development programs on UO_2 fuel technology.

From September 1961 to May 1966, I was employed by Westinghouse Electric Atomic Power Division, Pittsburgh, Pennsylvania, as Manager of Irradiation Technology. During this period, I was responsible for designing and conducting irradiation tests on water and fast reactor fuel and structural materials. From May 1966 until January 1968, I was an Advisory Engineer on the Engineering Manager's Staff in the Nuclear Fuel Division of Westinghouse Electric Corporation. My job function was to advise the Engineering Manager on a variety of technical subjects related to nuclear fuel technology.

It was during this period that I conceived and developed the Internally Pressurized Fuel Element Concept which is currently being used almost exclusively by all PWR fuel manufacturers.

From January 1968 to May 1969, I was employed as Manager, Fuel Analysis. My responsibility was to direct an analytical group on modeling the behavior of fuel elements under irradiation. During this period, we developed models to predict Zircaloy clad stresses and strains as a function of lifetime. My group was responsible for specifying the key element design parameters for Westinghouse nuclear power plants.

In May 1969, I assumed responsibility for Fuel Assembly Development. My function was to develop new fuel assembly materials and designs having higher reliability and lower cost. In August 1972, I became Manager of Fuel Licensing with prime responsibility for coordinating all licensing responsibilities within the Nuclear Fuel Division.

I have published over 20 articles in various technical periodicals related to nuclear fuel technology. In addition to these technical publications, I have authored or co-authored many Westinghouse reports. I have received eight patents and have several additional patents pending in the area of nuclear fuel technology.

Patents issued include: The Internally Pressurized Fuel Element Concept, which has resulted in the significant improvement in reliability in PWR fuel elements; Advanced Pressure Balancing Fuel Concepts Design to Minimize Stresses on the Cladding; advanced cladding material; and novel grid designs.

I was the first person selected by the AEC to represent the United States in the AEC/EURATOM exchange program. I was an active member and contributor from 1964 to 1968 to the Invited High Temperature Fuels Committee Meeting. I have lectured, presented talks and seminars at ANS meetings, universities, national laboratories, and at international conferences. I have been active in the American Nuclear Society and have served on the Executive Committee of the Materials Science and Technology Division. I have also been active on the ANS Program Committee and on Standards work. In June 1972, I was elected Fellow of the American Nuclear Society at the Eighteenth Annual Meeting in Law Vegas. This award was made for my work related to developing basic nuclear fuel technology data and reliable and economic nuclear fuel designs for commercial pressurized water reactors.

Q. To which contention is this testimony addressed?

A. DAARE/SAFE Contention 6. This contention reads as follows:

> The Intervenors contend that the FSAR provides insufficient assurance of containment of radioactive materials in light of, among other factors, the risks of use of zirconium cladding alloys resulting in a breach of the integrity of both internal and external systems. Our evidence for the unacceptability of zirconium cladding includes the matter contained in a letter to the Bulletin of Atomic Scientists by former Westinghouse nuclear engineer, Earl A. Gulbransen, published on page 5 of the June, 1975 issue of that journal. Quoting Dr. Gulbransen from that letter: 'At the operating temperature of nuclear power reactors zirconium cladding alloys react with oxygen in water to form an oxide layer which partially dissolves in the metal, embrittling and weakening the metal tubing. Part of the hydrogen formed in the zirconium-water reaction dissolves in the metal and may precipitate as a hydride phase also embrittling and weakening the metal tubing.' Further evidence of risk of using zirconium alloys occurs a bit later in the same letter: 'At temperature above 1100° Celsius zirconium reacts rapidly with steam with a large evolution of heat and the formation of free hydrogen, with most metals to form intermetallic compounds and with other metallic oxides to form its own oxide. Once zirconium is heated to 1100° Celsius, which could occur in loss of coolant accidents, it is difficult to prevent further reaction, failure of the tubing and of the reactor.
>
> Thus the conclusion is reached by Dr. Gulbransen that: 'The use of zirconium alloys as a cladding material for the hot uranium oxide fuel pellets is a very hazardous design concept since zirconium is one of our most reactive metals chemically.'
>
> Additionally, Applicant has not demonstrated the adequacy of its internal and external safety systems as impacted by a zirconium cladding failure. In the event of a loss of integrity of zirconium cladding, radiation levels exceeding those of the

design environment of the internal and external safety equipment and systems would occur. As the design basis for these systems and equipment does not include an integrity assurance in the event of a zirconium cladding failure by failing to consider such potential radiation levels in the design environment of the internal and external safety systems, the Intervenor contends that the proposed use of zirconium cladding, and the impact on the internal and external safety systems and equipment in the event of a zirconium cladding failure, require further examination.

Q. What materials are, in your opinion, suitable to use as a cladding material for fuel in light water reactors?

A. Suitable cladding materials for light water reactors are quite limited. A suitable material must have adequate mechanical properties, good corrosion resistance, and relatively low neutron absorption cross section, i.e., the material should absorb as few valuable neutrons as possible. Basically, only two types of materials - stainless steel and zirconium alloys - meet the above requirements.

Q. What is the experience gained from using either stainless steel or zirconium as a cladding material for light water reactor fuel?

A. Stainless steel was used only in a small number of early reactors. Operation with stainless steel cladding was quite successful with low defect levels. However, in order to conserve uranium resources, essentially all new reactors which began operation after 1967 used zirconium alloy cladding instead of stainless steel.

Zirconium alloys have the principal advantage over stainless steel that they have lower neutron absorption cross section and hence fewer neutrons are lost by wasteful absorption by the clad which means that more neutrons are available to sustain the nuclear reaction and hence less uranium is required to generate a given amount of energy. This results in lower fuel costs and hence savings to the utility and customer.

Experience with zirconium alloy cladding was initially not quite as good as stainless steel; however, after these initial problems were solved the overall experience with zirconium allow clad has been excellent. Zirconium alloys have been used as cladding material in several million fuel rods in hundreds of nuclear power plants throughout the world. During the past 25 years vast knowledge on the behavior of zirconium alloys as cladding material under both normal and accident conditions has been accumulated. This extensive data base provides the assurance that zirconium base alloys are indeed acceptable cladding materials for power reactors.

Q. In Contention 6, Dr. Gulbransen states that "at the operating temperature of nuclear power reactors zirconium cladding alloys react with oxygen to form an

oxide layer which partially dissolves in the metal, embrittling and weakening the metal tube. Part of the hydrogen formed in th zirconium-water reaction dissolves in the metal and may precipitate as a hydride phase also embrittling and weakening the metal tubing." Can you comment on this statement?

A. As stated by Dr. Gulbransen, zirconium alloys react or corrode slightly in the high temperature water under normal operating conditions to form an oxide layer. However, very little of the oxide layer dissolves in the metal so its effect on the mechanical properties of the metal is insignificant. Also, it is well known that the effect of oxygen is to strengthen, not weaken, the zirconium. On the other hand, some hydrogen does dissolve in the zirconium which tends to embrittle the metal; however, the quantity is appreciably less than 500 ppm (parts per million) and extensive testing has shown that zirconium alloys with less than 500 ppm of hydrogen retain sufficient ductibility so as to be acceptable. The excellent performance of zirconium-base alloy cladding in millions of fuel rods over 25 years confirm that there is no safety concern with normal operation.

In addition, there are Technical Specification limits for coolant activity which define the maximum

permissible equivalent Iodine-131 radioactivity in the coolant. The amount of radioactivity which is released by a single defect rod is small. Furthermore, during normal reactor operation, defects, if they occur, occur one or a few at a time. Thus, if defects were to occur in sufficient numbers to result in unacceptable levels of coolant activity there is time to shut down the plant and remove the defected fuel. Thus, it is never necessary to operate a reactor with unacceptable coolant activity levels. There is no safety concern with normal operation of zirconium-base alloy clad in a power reactor.

Q. During a worst case design basis accident (loss-of-coolant accident), Dr. Gulbransen states that at temperatures above 1100°C (2012°F) zirconium reacts rapidly with steam to form zirconium oxide, releasing hydrogen and heat. Is this statement correct?

A. Yes, this so-called zirconium metal-water reaction was a primary consideration which led the NRC to establish a post-LOCA upper limit on zirconium cladding temperature of 2200°F and maximum cladding oxidation (metal-water reaction) of 1% for the core (see 10 CFR 50.46). These criteria are established by law as a result of extensive hearings in 1973 and are generally considered to be conservative.

Q. Have there been other research programs conducted which evaluate the rate of zircaloy oxidation at high temperatures?

A. Subsequent to the ECCS hearings in 1973, a worldwide program evaluating the rate of Zircaloy oxidation at high temperatures has been carried out. Specific examples are programs by Oak Ridge National Laboratory, Battelle Northwest Laboratory, Worcester Polytechnical Institute, Japan Atomic Energy Research Institute and the Nuclear Safety Research Center of the Federal Republic of Germany.

These tests have investigated oxidation rates up to 1500°C, well above the 1100°C cited in the contention. Results of the tests, which again are available in the open literature, show that the oxidation rates at high temperatures are substantially lower than those calculated by the formulation specified by 10 CFR 50 Appendix K and used in our licensing calculations.

Q. What analyses have been performed regarding the fuel to be used at Byron?

A. Westinghouse has calculated the peak clad temperature of the Byron zirconium alloy clad fuel rods following the worst case large-break LOCA using an NRC approved evaluation model that meets the requirements of 10 CFR 50 Appendix K. These calculations, which appear in Section 15.6 of the Byron/Braidwood FSAR, show that the

peak clad temperature reaches 1982°F (1083°C) and the total zirconium metal-water reaction is less than 0.3%. These values are appreciably below the permitted values of 2200°F (1204°C) clad temperature and 1% metal-water reaction.

Although a loss-of-coolant accident results in cladding failures as stated by Dr. Gulbransen, the worst case design basis loss-of-coolant accident has been analyzed and the FSAR shows that the boundary dose limits are well within the 10 CFR Part 100 allowable limits and thus the health and safety of the public is adequately protected.

Q. Could you please summarize your conclusions regarding the safety significance of the matters raised in Contention 6.

A. All the concerns raised in Contention 6 have been carefully considered and reviewed in the FSAR, and by the NRC and experts in many countries and it is clear that sufficient assurance exists to assure the containment of radioactive materials under normal operation and even under the worst postulated design basis accident.

DAARE/SAFE CONTENTION 7

THE CONTENTION

The Intervenor contends that the FSAR and Applicant offer insufficient safeguards against hydrogen explosions, such as are alleged to have occurred at Three Mile Island Reactor 2. There is no evidence that the recombiners for taking up hydrogen would be adequate if circumstances similar to those at TMI-2 should occur at Byron.

MATERIAL FACTS AS TO WHICH THERE IS NO GENUINE ISSUE TO BE HEARD

1. Byron Station, Units 1 and 2, are Westinghouse PWR reactors housed in large dry containments. (Affidavit of T. R. Tramm, p. 3.)
2. The Nuclear Regulatory Commission is considering hydrogen control issues in pending rulemaking proceedings. 46 Fed. Reg. 62,281.
3. Commonwealth Edison Company intends to comply with any new or additional hydrogen control requirements applicable to the Byron Station developed as a result of the pending NRC rulemaking on hydrogen control. (Affidavit of T. R. Tramm, p. 3.)

DISCUSSION

The accident at Three Mile Island resulted in the generation of hydrogen from the reaction of water with fuel cladding in the reactor core. It is generally accepted that the level of hydrogen generated was in excess of the amount

required to be assumed by the NRC regulations. 10 C.F.R. § 50.46(b)(3) and § 50.44(c)(3). In response to this problem, the NRC initiated rulemaking proceedings.

On December 23, 1981, the Commission published a proposed interim rule which specifically contemplates "the ability of all light water reactors . . . to withstand an accident with the concomitant generation of large amounts of hydrogen such as the type which occurred at . . . TMI-2." 46 Fed. Reg. 62,281. The proposed Interim Rule of December 23, 1981 would require design analyses to be performed for new hydrogen control measures. For purposes of these analyses, the amount of hydrogen to be assumed in the design of the hydrogen control system is that amount generated by assuming that 75% of the fuel cladding reacts with water. As the Commission stated: "This 75% value . . . appears to be reasonable because it is sufficiently greater than the fuel cladding - water reaction analyzed to have occurred at TMI-2 to provide a conservative estimate for the cladding reaction that may occur during a TMI-type degraded core accident." 46 Fed. Reg. at 62,282. The proposed Interim Rule of December 23, 1981 specifies different analysis requirements for different containment designs. For this purpose, containments are divided into several classes: Mark I & II BWR containments, Mark III BWR and ice condenser PWR containments, and "other" containments. The proposed

rule requires that for every light-water reactor for which analyses are required, other than the Mark I, II, and III BWR's and ice condenser PWR's, the analysis shall be performed "by [two years after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 per cent of full power, whichever is later." 46 Fed. Reg. at 62,284. (Emphasis added.)

The Byron Station is a PWR with a large dry containment. (Affidavit of T. R. Tramm, p. 3.) Contention 7 explicitly concerns an accident scenario leading to hydrogen generation similar to the accident which occurred at TMI-2. The Commission has under consideration the consequences of the generation of large amounts of hydrogen within a containment, including a containment such as the Byron containments, following a TMI-2 type event, in the pending rulemaking discussed above. Quite obviously, the matters raised in Contention 7 are within the scope of rulemaking proceedings pending before the Commission.

Under the general rule stated by the Appeal Board in Potomac Electric Power Co. (Douglas Point Station, Units 1 and 2), ALAB-218, 8 AEC 79 (1974), and the specific Appeal Board holding in Sacramento Municipality Utility District (Rancho Seco Nuclear Generating Station), ALAB-655, 14 NRC

799, at 816-17 (1981), it is clear that Edison is entitled to summary disposition of Contention 7 as a matter of law. In Rancho Seco, the Appeal Board considered the appropriateness of a licensing board's initial decision on a contention which concerned the ability of the plant "to cope with the generation of hydrogen within the containment following an accident like that at TMI-2." 14 NRC 799 at 816. The Appeal Board refused to review or explicitly comment on that portion of the licensing board's initial decision pertaining to the hydrogen control contention based on the fact that the Commission is considering the consequences of hydrogen generation following a TMI-2 event in a general rulemaking proceeding. In so ruling, the Appeal Board expressly relied on its holding in Douglas Point that "licensing boards should not accept in individual licensing proceedings contentions which are (or are about to become) the subject of general rulemaking by the Commission." Id.

Thus, in view of the pending hydrogen control rulemaking and in view of Edison's commitment to comply with any applicable hydrogen control requirements developed as a result of this proceeding (T. R. Tramm Affidavit, p. 3), the Licensing Board has ample grounds to conclude that there is reasonable assurance that any reasonably postulated generation of hydrogen at the Byron Station will not pose a threat to the public health and safety. Therefore, Edison is entitled to summary disposition of Contention 7 as a matter of law.