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 GEORGE KLOPP

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.

George Klopp
George Klopp

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

Maureen Sauer
Notary Public

TESTIMONY OF GEORGE KLOPP

ON DAARE/SAFE CONTENTION 4

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

A. My name is George Klopp. I'm a General Design Engineer in Commonwealth Edison's Station Nuclear Engineering Department.

Q. Briefly state your educational and professional qualifications.

A. In 1964 I received a B.S.M.E. and in 1965 I received an M.S.M.E. (nuclear option) from the University of Kentucky. I've been with Commonwealth Edison since 1965, except for a 2 year military leave of absence, and have been involved in engineering, operation, engineering management, and safety analysis relating to nuclear plants for my entire career.

Q. Describe your current duties and responsibilities with Commonwealth Edison.

A. At this time, I have a number of responsibilities. They include: 1) Ongoing work relative to the Zion Probabilistic Safety Study (PSS) including licensing activities, training Edison personnel with respect to the study, and evaluations of the need for power plant design modifications and changes to operating procedures related to the Zion PSS. 2) Lead engineer for Edison's technical participation in the Industry Degraded Core

Program. 3) Acting as Edison's representative to the technical writing group of the industry/NRC Probabilistic Risk Assessment Procedures Guide Program. 4) Acting as technical adviser on the Clinch River Breeder reactor plant probabilisitic risk assessment program. 5) Acting as Edison's representative to the Department of Energy Working Group on Probabilistic Risk Assessment. 6) Acting as pressurized water reactor technical adviser to Edison's Generating Stations Emergency Plan program. 7) Acting as a technical adviser to other groups or departments within Edison on matters related to nuclear safety, risk assessment and degraded core phenomenology.

Q. What other assignments have you held at Commonwealth Edison Company?

A. My earlier assignments with the Company included: 1) Acting as lead Nuclear Engineer, in Edison's Engineering Department, for the Zion Station project during design, construction, and initial start-up. 2) Acting as the Project Engineer for the first two years of the Byron/Braidwood Project Development. 3) Serving as an Operating Engineer in the radioactive waste disposal area at the Dresden Station. 4) Serving as a Project Engineer for the Station Nuclear Engineering Departments Reliability and Design Engineering Group. 5) Serving as Edison's technical director on the Zion Probabilistic Safety Study.

Q. To which contention is this testimony addressed?

A. Contention 4. In general this Contention asserts that the Byron FSAR "does not analyze the risks to the public health and safety from the potential of accidents resulting from multiple, mutually independent failures as opposed to a 'single failure' as defined in 10 CFR part 50, Appendix A". The Contention also identifies, and by implication requests analysis of, fifteen examples of "multiple failure accidents".

Q. Have you been provided with any additional information bearing on the matters which DAARE/SAFE are attempting to raise in Contention 4?

A. Yes. In the course of discovery, DAARE/SAFE identified Dr. Michio Kaku as its expert witness on Contention 4. I attended Dr. Kaku's deposition at which he was interrogated by lawyers for Edison and the NRC Staff. Since then, I have been provided with a copy of the transcript of Dr. Kaku's deposition which I reviewed in the course of preparing this testimony.

Q. Did Dr. Kaku further clarify and specify the scope and focus of Contention 4?

A. He did. Dr. Kaku set forth his understanding of single failure criteria on which the Intervenors base Contention 4. He also further specified and clarified the fifteen accident scenarios identified in the Contention, identified three additional accident scenarios, and provided

his rationale for the selection of these accident scenarios and a specification, in general terms, for their evaluation by Edison. (A restatement of the accident scenarios as explained and expanded by Dr. Kaku during his deposition is attached to this affidavit.) Finally, Dr. Kaku explained that if Edison evaluated these accident scenarios using the tools described by Dr. Kaku and the acceptance criteria selected by Dr. Kaku, DAARE/SAFE's concerns relating to Contention 4 would be satisfied.

Q. How does Dr. Kaku define single failure as used in the NRC's regulations governing assumptions for accident analyses?

A. Dr. Kaku appears to believe that the single failure which must be assumed for accident analyses and plant design is the event which initiates the accident.

Q. Is this definition consistent with the definition of single failure as used in 10 CFR Part 50, Appendix A?

A. No. My understanding of the regulation and the way in which it has been consistently applied by Commonwealth Edison Company and the NRC Staff in analyzing postulated accidents is very different. Appendix A actually requires that individual safety systems be designed such that no single failure in a given system will disable the required safety function. These "single failures" are independent of the event which initiates the accident.

Q. In your opinion, how does DAARE/SAFE's interpretation of a "single failure" affect its assertion that the Byron FSAR does not address multiple failures?

A. It demonstrates that DAARE/SAFE and Dr. Kaku misunderstood the accident analysis required by the NRC regulations and hence the analysis performed for Byron. The Byron design complies with the single failure criteria as defined in 10 CFR Part 50, and also considers what DAARE/SAFE refers to as "multiple failures". For example, in Chapter 15 of the Byron FSAR, Edison postulates and analyzes the following large loss of coolant accident. A double ended, cold leg break in the reactor cooling system is analyzed coincident with the following events: a design basis earthquake; loss of off-site power; failure of one of the two residual heat removal trains, a failure of one of the two containment spray trains, and a failure of two of four containment fan cooler units. Also, the accumulator on the broken reactor coolant loop is assumed to spill all its water on the containment floor with no flow to the reactor core. Clearly, these postulated accidents are multiple failure events.

Q. Have the accident scenarios identified in Contention 4, as these were explained by Dr. Kaku during the course of his deposition, and the three additional accident scenarios identified for the first time by Dr. Kaku during his deposition been included as design-basis

accidents for the Byron Station or otherwise evaluated in the FSAR?

A. With two exceptions (examples 6 and 13), they have not. In addition, however, by reference to WCAP - 8330, Edison has included a generic evaluation of a variety of ATWS events for four loop Westinghouse plants, such as Byron. Some of these events are included in Dr. Kaku's listing of accident scenarios.

Q. Do the NRC Regulations, the NRC Staff's Standard Review Plan, or any other NRC Staff guidance documents require that the accident scenarios identified by Dr. Kaku be included as design-basis accidents for the Byron Station?

A. No, with the exception of ATWS and examples 6 and 13, Edison's analysis of which is discussed in Chapter 15 of the FSAR, the scenarios identified by Dr. Kaku are not required to be analyzed.

Q. Please explain the selection process for the accidents analyzed in the Byron FSAR.

A. The accidents analyzed in the FSAR were selected based on the NRC Staff Regulatory Guide 1.70, Revision 2, and the requirements imposed by NUREG 0737, "Clarifications of TMI Action Plan Requirements."

In essence, these requirements prescribe that an applicant consider certain accident scenarios and conservatively evaluate whether the equipment and systems included as part of the design of the facility

are capable of effectively withstanding and mitigating the results of the accident. The accident scenarios selected for analysis are considered to be bounding accidents. In other words, it is expected that credible events which are not specifically analyzed would not result in conditions which are more severe than those resulting from the accidents analyzed and designed for.

Q. Do the accidents analyzed bound the entire spectrum of conceivable nuclear power plant accidents?

A. No. Certain accidents, which can be hypothesized and which would result in conditions which are more severe than those which are predicted to result from design basis accidents, are not considered. These accidents, which are beyond the design basis accident and are commonly referred to as Class 9 events, are judged to be so unlikely in terms of probability that an applicant is not specifically required to design for such events.

Q. If an accident scenario, is beyond the design basis accident, is an applicant by definition relieved of having to consider the accident scenario in developing the design of its nuclear facility?

A. No, under existing Commission policy, if the NRC Staff or the Commission determines that a particular facility has unusual or unique design or siting characteristics, an applicant may be required to consider accident scenarios which go beyond design basis event.

Q. In your opinion, does the Byron facility have any such unique or unusual characteristics?

A. No, it does not. Byron Units 1 and 2 are typical Westinghouse PWR reactors and the site of the facility is in a relatively low population area. In addition, there are no unusual hazards near the Byron site from which one could conclude that the Byron site is unique or unusual.

Q. Have you considered the accident scenarios identified by Dr. Kaku to determine whether they represent Class 9 events?

A. Yes. For the most part, if one employs the conservative evaluation techniques and acceptance criteria typical of FSAR-type accident analyses and specified by Dr. Kaku, the accident scenarios would likely demonstrate some degree of core degradation. In other words, these scenarios would represent Class 9 events.

Q. Have you evaluated the specific scenarios identified by DAARE and SAFE.

A. Yes, I have. I have examined each of these scenarios to determine whether they had been evaluated as part of the FSAR process, whether they are in fact physically possible, and whether under Dr. Kaku's relative risk criterion there is any basis for conducting the requested analyses.

Q. Could you please define what you have just described as "Dr. Kaku's relative risk criterion"?

A. During the course of Dr. Kaku's deposition, he admitted that as a practical matter there had to be some upper bound on the accident selection process which he advocates. Dr. Kaku stated that he would not require that Edison analyze an accident scenario which results in a PWR-1 type release.

A PWR-1 release is one of a series of release groupings discussed in the Reactor Safety Study otherwise known as WASH-1400. These groupings specifically address the amount of radioactive material released from the containment building given groups of accident scenarios which damage the core and the containment in various ways. They also address the energy with which such releases occur which affects the dispersion, in the environment, of radioactive material. In WASH-1400 the PWR-1 release is the most severe and others, in descending order of severity, range down to a PWR-9. The PWR-1 releases come directly from a postulated steam explosion inside the reactor vessel. It assumes that molten core material drops into the lower vessel head, contacts the water therein and results in an in-vessel steam explosion which forces a solid slug of water against the upper head of the vessel. This upper head, followed by a large amount of radioactive material, is postulated to impact on the upper dome of the containment building with the resulting failure of that dome

and energetic release of large quantities of fission products. Dr. Kaku admitted that this scenario appears to strain credibility for individual licensing proceedings and stated that he did not believe that Edison should be required to analyze such an accident in connection with the Byron operating license proceeding.

Accordingly, I have evaluated the 18 scenarios postulated by Dr. Kaku and compared the risks associated with these scenarios in relation to the risks associated with a PWR-1. In other words, I have attempted to determine whether the risks, that is the consequences and probabilities of occurrence, of a PWR-1 are greater than the risks associated with the 18 accident scenarios. If this is the case, I assume that Dr. Kaku would not require the evaluation of his 18 accident scenarios as a conditon to licensing of the Byron facility.

Q. What is the probability in WASH-1400 of the PWR-1 type accident discussed by Dr. Kaku?

A. In WASH-1400, the most likely single sequence leading to a PWR-1 type release was determined to have a probability of 3×10^{-8} per year. I have used this probability figure as the yardstick against which I have compared, where possible, the probability of the accident scenarios discussed by Dr. Kaku.

Q. Is the evaluation that follows based strictly on probability estimates?

A. No. Since the contention alleges that Edison did not adequately consider the risk associated with possible accidents at Byron, and since it is impossible to consider the degree of risk without also considering the likely consequences associated with an event, in addition to comparing probabilities, I have also compared the consequences of the 18 scenarios identified by Dr. Kaku with the consequences associated with a PWR-1. Thus, if the probability and consequences of these 18 scenarios can reasonably be judged to be lower than those of a PWR-1, the risk associated with these accidents is lower than the risk associated with the PWR-1 and we have satisfied Dr. Kaku's implicit acceptance criteria relating to accidents which Dr. Kaku believes ought to be considered in the design of the Byron Station.

Q. Do your estimates represent absolute values in terms of probabilities of the examples considered?

A. No. They are intended to represent relative values which can be compared to the probability of PWR-1 type events. These estimates are approximate, first order estimates. However, I believe they are adequate for establishing the relative safety importance of the examples identified by Dr. Kaku, particularly when used in connection with estimates of the relative consequences of these examples.

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Q. How did you compare the severity of the consequences of PWR-1 with the severity of the consequences of the examples identified by Dr. Kaku?

A. First, it is important to keep in mind that to date PWR-1 constitutes the most severe type of accident in terms of postulated radioactive releases. Thus, as a comparison tool, I classified the PWR-1 release as "very severe." At the other end of the spectrum, WASH-1400 identifies certain events that do not involve a failure of the containment building. These types of events are identified as PWR-8 or PWR-9 events. The releases associated with such events are very much lower than those associated with a PWR-1, and for the purposes of the relative risk comparison, I classified these types of events as "very low" severity events. The other release groupings in WASH-1400, PWR-2 through PWR-7, are classified as "severe" to "low" relative to the PWR-1 "very severe" classification. For the purposes of my consequence comparison, I have classified PWR-2 and PWR-3 events as "severe," PWR-4 and PWR-5 events as "moderate," and PWR-6 and PWR-7 as "low."

Q. Please describe your analysis of each of the 18 examples identified by Dr. Kaku, including your risk comparison assessment of these scenarios to a PWR-1 type event.

A. For the sake of clarity, I will discuss each example in the order presented in the contention. I will discuss Example 1 in substantial detail, explaining the entire thought process that went into the evaluation. For the remaining examples, the discussion is somewhat abbreviated, but the analytical process by which I came to a conclusion for each succeeding example is substantially similar to that described for Example 1.

Example 1 postulates a rupture of defective CRDM housings which cause ruptures of adjacent, similarly defective CRDMs with resulting control rod cluster ejections coupled with a failure to scram immediately and with a postulated, variable time delay on the scram.

Two separate techniques may be employed to judge the probability of a multiple CRDM housing rupture. To simplify the process, I considered only the probability associated with two such ruptures. The probability of additional ruptures would, of course, be lower.

The first technique consists of viewing the housing as a pipe section and using WASH-1400 data for the frequency of pipe section failure. From WASH-1400, that frequency is 8.6×10^{-10} failures per hour. Given roughly 7.2×10^{3} hours of operation per year, the annual probability of failing a single housing is 6.2×10^{-6}. The probability of failing two housings is the square of this failure probability or 3.8×10^{-11} per year. This assumes that the failures are independent.

Such an assumption is warranted since the design precludes one such failure from inducing another. (FSAR chap. 15.)

The second technique consists of recognizing that operating experience shows that there have been over 20,000 CRDM operating years with no failure such as that postulated in Example 1. In fact, in that time period no defects have been noted which would lead to this type of failure. If a defect were to occur, experience indicates that any defect growth would most likely lead to a slow leak which would be readily detected. It would most likely not lead to a sudden rupture. Thus, we can conservatively say that 1/10 of CRDM defects might lead to a rupture. Then, one can conservatively develop the likelihood of a single rupture as 1/20,000 x 1/10 for a total probability of 5×10^{-7} failures per year. If we conservatively assess the likelihood that one rupture would cause another, we can multiply the single rupture probability (5×10^{-7}) times a second CRDM defect probability (5×10^{-6}) for a total of 2.5×10^{-12} double failures per year.

These two approaches provide a reasonable range of the probabilities for the failure of two CRDM housings (between 3.8×10^{-11} and 2.5×10^{-12} failures/year). Dr. Kaku augmented this pair of failures by postulating

a delayed scram. The design of the Westinghouse system is such that a scram delay comes only from an initial failure to trip and subsequent operator action to effect a scram. The initial failure can result from gross core distortion or trip logic and breaker failures. The probability of such a failure would, when coupled with the CRDM housing failure, lower the overall event probability by a few more orders of magnitude below the values noted above. At this point, it is clear that the probability of the event is far below the probability of a PWR-1. Indeed, the probability of the two CRDM housing failures is itself below the probability of a PWR-1.

With respect to the consequence comparison, since the failure of two CRDM housings depressurizes the primary system through ultimate loss of cooling inventory, the reactor would shut down on voids i.e., the reactor core would become subcritical. Even if a delayed scram was postulated, the reactor coolant system and reactor protection/ESF actuation logic will respond to the event as if it were a more conventional loss of coolant accident (LOCA). Operation of the emergency core cooling system (ECCS) which uses borated water, would reflood the core. Containment fan cooler and containment spray operation would help to maintain containment pressure within limits and remove fission products. Even if the scram system were delayed long

enough for significant core degradation to occur, containment integrity would not be threatened. In such a circumstance, although core degradation would result, it is unlikely that full core melting would occur since once significant core distortion occurs, continued criticality is not realizable. Also, there could be a significant amount of hydrogen generated, and a hydrogen burn could result. However, given the low steam back pressure and the robust nature of the Byron containment, a burn of even up to 100% of the clad-water reaction generated hydrogen would not threaten containment integrity.

Since the containment would remain intact and the core, though degraded, would not fully melt, the event postulated would result in a PWR-8 or PWR-9 type of release; i.e. a "very low" consequence release.

Thus, not only is the probability of the scenario identified in Example 1 lower than for a PWR-1 type event, the consequences of the accident would also be much lower than for the PWR-1 release. Therefore the level of risk associated with the accident postulated in example 1 is considerably lower than the risk associated with an accident leading to a PWR-1 release.

Example 2 is a consolidation of five projected initiating events coupled with the failure to scram. Using an analysis similar to the one discussed above,

the core melt frequency from the most likely event (loss of feed water combined with ATWS) is approximately 2×10^{-8} per year. Again, using the consequence analysis discussed above with respect to Example 1, the consequences of experiencing an event such as those postulated in Example 2 would also be classified in the "very low" category. Therefore, the relative risk associated with Example 2 is much lower than a PWR-1 type event.

Example 3 involves a continuous withdrawal of control rods coupled with a delayed scram. The estimated frequency of all core power excursion events coupled with a delayed scram is on the order of 1×10^{-16} per year and the resulting consequences would be in the "very low" category. Again, the relative risk would be much less than that associated with a PWR-1.

Example 4 involves a large LOCA with a failure of an accumulator (two out of four available) and the failure of the high pressure ECCS systems. The estimated annual frequence of occurrence of this example is about 1×10^{-13}. Since a high pressure injection system is not required for the large break accident, only the failure of the accumulator acts to degrade ECCS performance. The consequence of this action would also be in the "very low" category. Thus, example 4 identifies an accident which presents lesser risks than a PWR-1 type release.

Example 5 involves a spontaneous reactor vessel failure involving a defective closure bolt and either an ATWS event or a rod ejection accident with failure of the relief valves to open. The most likely scenario which would lead to the accident identified in this example has a probability of about 6×10^{-11} per year. Also, it is likely that even if a defective vessel bolt were to fail no additional bolt failures would occur. At worst, such an event might be postulated to have some likelihood of inducing a LOCA or steam generator tube rupture due to pressurization. Given ECCS operation, the event would be successfully terminated and the consequences would be "very low." Again, the probability of the accident postulated is lower than that of a PWR-1 event, the consequences are less severe and thus the relative risk associated with this accident is lower than that of an accident leading to a PWR-1 release.

Example 6 has been considered and analyzed as a design basis accident in Chapter 15 of the FSAR. Therefore, there is no basis for DAARE/SAFE's assertion that Edison should evaluate this accident.

Example 7 involves two concurrent LOCA events, a large cold leg break and a small hot leg break. The frequency of this event is estimated to be approximately 9×10^{-12} per year. Given the break size postulated for the large break (one square foot), it is likely

that the combination of the breaks would not exceed the capability of the ECCS to prevent severe core degradation. However, even in the event of a full core melt, the operable containment fan coolers and sprays would prevent containment failure. Again, the probability and consequences and thus the risk associated with this event are lower than the PWR-1 type event.

Example 8 involves a postulated reactor coolant system pipe break caused by one of three mechanisms. The first mechanism is a primary system water hammer. Given the configuration and the design of the primary system, a water hammer is not physically realistic. No such event ha been noted on a pressurized water reactor. Therefore, I did not give this mechanism detailed consideration. The second failure involves the control rod drive mechanism failure. This type of CRDM failure does not lead to pressures which cause stresses beyond the faulted limit as specified in the ASME Code as discussed in Chapter 15 of the FSAR. The annual frequency of pipe break resulting from this event is estimated to be about 1×10^{-15} and the consequences would be very low. The third mechanism postulated was an in-vessel steam explosion resulting from a core melt. No probabilistic evaluation was performed for this postulated scenario since the event starts out by presuming a core melt. Given a 50% core melt and sudden drop of the molten material into the lower

vessel plenum, the reactor vessel lower head would fail at one or more in-core guide penetrations within one or two minutes. The concern over a pipe break is obviously misplaced. However, even given such an event, the operation of the containment fan coolers and sprays would insure flooding and continued cooling of the ex-vessel debris, thereby protecting against containment failure.

Example 9 involves secondary to primary flow in a steam generator tube rupture. In explaining this scenario, Dr. Kaku stated that his concern was with contamination of the primary coolant by secondary side water. The use of all volatile chemistry on the secondary side would cause the introduction of chemicals such as morphylene and hydrazene into the primary side water. However, since these chemicals are nitrogen, hydrogen and oxygen based compounds, they would have no deleterious effect on the primary side system.

Example 10 involves a steam generator tube rupture (similar to that at Ginna) coupled with an operator error which would depressurize the primary system. The annual frequency of such an event, leading to a core melt, is about 6×10^{-9}. The consequences of such an event would range from very low to severe. In the severe case, one must postulate a failure to isolate the affected steam generator and further postulate

rather gross, open ruptures in a dry secondary side. These postulates would lower the likelihood by approximately three orders of magnitude to approximately 6×10^{-12} Again, the probability and consequences are lower for a PWR-1 release.

Example 11 essentially postulates a TMI type event with subsequent high pressure injection failure due to operator error. The frequency of such an event, given the emphasis on a TMI event, is estimated to be about 4×10^{-11} per year. The frequency of a core melt is significantly lower. Since the consequences of the event are very low, it is of little safety significance.

Example 12 postulates a degraded core and a pressurized system followed by operator error that depressurizes the system to atmospheric pressure. Dr. Kaku essentially requested that Edison demonstrate that the core would not uncover. First, by definition, to obtain the degraded core state described in the contention one must postulate that the core had already been uncovered. Secondly, even if one postulates the core had been recovered due to ECCS flow, it is obvious that to depressurize to atmospheric pressure the core would have to be not only uncovered but totally void of water. The example, thus, is absurd and should not be analyzed.

Example 13 involves a loss of main feedwater and a loss of the safety injection system. Loss of main feedwater is considered in Chapter 15 of the FSAR.

Moreover, operation of the safety injection system has no bearing upon the ability of the plan to prevent or mitigate adverse and unacceptable consequences resulting from loss of main feedwater. Therefore, the combined failure postulated has no safety significance.

Example 14 involves a large LOCA and a delayed scram. The large LOCA would scram the reactor on voids very quickly. The injection of borated ECCS water would cool the core. At worst, depending on the scram delay, we would expect some core damage, but it is not likely that the full core melt would occur. The overall likelihood of a core melt is thus judged to be below 1×10^{-10} per year. In addition, the consequences of the accident would be low.

Example 15 involves a total blackout of the AC power to one unit. The basic likelihood of such an event is about 3×10^{-5} per year. The event does not lead to significant core damage. Moreover, it is estimated that the likelihood of the event occurring and persisting for one-half hour is less than about 1.4×10^{-8} per year. Given the lack of significant consequences, the example is trivial.

Q. Can you recapitulate your conclusions relative to the examples, ignoring those already in the FSAR or classified as absurd?

A. They are trivial and insignificant relative to Dr. Kaku's own criteria.

Q. How might they be judged in a more global sense of risk or safety?

A. To answer that, one has to be familiar with a number of individual power plant probabilistic risk assessments, the proposed Policy Statement on Safety Goals for Nuclear Power Plants and Byron Station. Having that familiarity, I judge that the examples in contention 4 are truly trivial and insignificant in terms of real public safety for Byron Station. In other words, I judge they are very unimportant in terms of their contribution to risk at Byron.

Q. What do you conclude as a result of these assessments?

A. Clearly, I must conclude that the requested analyses of the examples in contention 4 would be nonproductive in terms of learning anything important about safety or in terms of contributing to safety at Byron. I must also conclude that any such examination of class 9 events in these proceedings is contrary to NRC policy. Therefore, the requested analyses should not be performed.

DAARE/SAFE CONTENTION 6

CONTENTION 6

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 hybride 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 temperatures 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, 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.

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

1. The amount of embrittlement and/or weakening of zirconium due to the dissolving of hydrogen and/or oxygen in the zirconium under normal operating conditions does not cause unacceptable embrittlement or weakening of the zirconium cladding material. (Affidavit of Dr. Harry Ferrari, p. 8.)

2. Required monitoring of the activity level of iodine-131 during operation will permit detection of any unacceptable cladding defects. (Affidavit of Dr. Harry Ferrari, p. 9.)

3. The peak cladding temperature of the Byron zirconium alloy cladding fuel rods following a postulated worst case large-break-loss-of-coolant accident is 1,982° F., and the total zirconium metal-water reaction is less than 0.3%. (Affidavit of Dr. Harry Ferrari, p. 10-11.)

DISCUSSION

In Contention 6 DAARE/SAFE make two separate arguments. The first is that zirconium alloys are an unacceptable cladding material due to weakening caused by dissolving hydrogen or oxygen or both in the material at normal operating temperatures. Second, DAARE/SAFE argues that if the temperature of the zirconium cladding reaches 1100° C. (2,012° F.) there is a significant risk of rapid metal-water reaction and failure of the cladding. This second argument is clearly a challenge to the fundamental basis for the final acceptance criterial for emergency core cooling systems in 10 CFR §50.46(b)(1). The final acceptance criteria clearly expresses the Commission's determination that if the maximum fuel embrittlement cladding temperature does not exceed 2,200° F., the emergency core cooling system provides adequate protection to the public health and safety. In any event, calculations of the performance of the emergency core cooling system of the Byron Station demonstrate that the peak cladding temperature will be no more than 1,982° F., less than the temperature about which Dr. Gulbransen has expressed concern.

As to the first issue, embrittlement at normal operating temperatures, Dr. Gulbransen candidly admitted in his deposition in this proceeding (see Transcript, p. 124 and 130) that, although some embrittlement of the zirconium would occur at normal operating temperatures, he did not know nor was he qualified to express an opinion on whether the

embrittlement would lead to fuel failure. In the attached testimony of Dr. Harry Ferrari, a qualified expert, he states that, although there is some embrittlement of the zirconium under normal operation conditions, the zirconium retains an acceptable level of ductility. Dr. Gulbransen's disclaimer of his ability to ascertain whether the embrittlement caused by dissolved hydrogen or dissolved oxygen is unacceptable does not raise a genuine issue of material fact. Virginia Electric Power Co., (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC 451 (1980). The above-listed and adequately supported facts are not disputed and demonstrate that Applicant is entitled as a matter of law to a favorable decision on this Contention.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In the matter of:

COMMONWEALTH EDISON COMPANY :Docket Nos. 50-454
: 50-455 (OL

(Byron Nuclear Power Station,
Units 1 and 2)

- -x

Conference Room 102
Harley House Motel
Monroeville, Pennsylvania

Friday, May 28, 1982

Deposition of DR. EARL GULBRANSEN, called for examination by counsel for the Applicant, taken before Ann Riley, a Notary Public in and for the State of Maryland, beginning at 10:35 a.m., pursuant to agreement of counsel.

1 understand it, and not on the details, -- what I am saying
2 in this letter was that this -- in dissolving the metal
3 and precipitating -- and certainly you are going to
4 precipitate at 500 parts per million, which will develop
5 embrittlement. It develops stresses, which is bad for
6 the tubing. Whether it's going to fail or not, that is
7 something else. All this happens.
8 Q All right. So you don't really know, is the
9 answer to my question; is that right?
10 A I know it embrittles, but I can't tell it's
11 going to lead to a failure, because I am not that involved
12 in this failure analysis. That statement stands as it is
13 written. It embrittles and it weakens metal tubing. I
14 didn't say it was going to lead to a failure. I just said
15 it embrittles the metal and weakens the tubing.
16 Q So when you wrote your letter and included
17 that sentence, you weren't suggesting that a failure
18 would occur. You just -- what were you suggesting?
19 A I am suggesting that this happens, and this is
20 not a good thing to happen in cladding tubing.
21 Q Whoever wrote Contention 6 seemed to think it
22 was a problem, because they used your sentence as a

1 cladding; isn't that right?

2 A I would say that I know that it is -- I would

3 not like to have hydride --

4 Q No, no, answer the question. You don't know

5 whether or not the cladding is going to fail under those

6 conditions from embrittlement, do you?

7 A I have no evidence it's going to fail or not

8 fail.

9 Q So you just don't know?

10 A I don't know.

11 Q All right.

12 A But it's very bad to have hydride precipitating

13 in the lattice, and we know it's going to precipitate. It's

14 going to be there and there are going to be stresses, there

15 is going to be embrittlement. And you say okay, or some

16 tests show it's okay. I say you'd better be careful, if

17 you have got that kind of situation, because it's a very

18 localized stressed area.

19 Q Well, let me ask you this:

20 Can you come up with zero hydriding? Is it

21 possible to come up with zero hydriding in the technology

22 we are talking about?