4/25/79

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	:	Docket No. 50-272
PUBLIC SERVICE ELECTRIC & GAS CO. (Salem Nuclear Generating Station, Unit #1)	:	Proposed Issuance of Amendment to Facility Operating License No. DPR-70
	:	

RESPONSE TO THE ATOMIC SAFETY AND LICENSING BOARD ORDER DATED APRIL 18, 1979

On April 18, 1979, the Atomic Safety and Licensing Board issued an Order presumably pursuant to Rule 2.752(c) or other applicable rule which provides for the submission of evidence addressed to the following questions:

"1. To what extent did the accident at Three Mile Island affect the spent fule pool at that site?

2. If there had been an explosion or 'meltdown' at Three Mile Island, what affect would that have had upon the spent fuel pool? To what extent would it have mattered how much spent fuel was present at the pool?

3. If an accident such as the one at Three Mile Island occurred at Salem, to what extent would the accident affect the spent fuel pool? If an explosion or 'meltdown' occurred at Salem, to what extent would it have mattered how much spent fuel was present at the pool at Salem? "

The Intervenor, Township of Lower Alloways Creek is in receipt of a technical report dated February 27, 1979 which deals with

the accident hazards of spent fuel storage at the Salem Nuclear Power Plant. This technical report is relevant to Question #3.

The proponent of this testimony is Dr. Richard E. Webb, and Dr. Webb's qualifications are attached to his testimony.

Respectfully submitted,

CARL VALORE, JR., Special Nuclear
Counsel for the Intervenor, Township
of Lower Alloways Creek

TESTIMONY OF RICHARD E. WEBB, Ph.D.
IN RESPECT TO BOARD QUESTION #3 OF ORDER
DATED APRIL 18, 1979

THE ACCIDENT HAZARDS OF SPENT FUEL STORAGE AT THE SALEM NUCLEAR POWER PLANT SALEM, NEW JERSEY

BY
RICHARD E. WEBB, Ph.D.
February 27, 1979

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1. INTRODUCTORY CONTENTION: THE LOSS-OF-WATER ACCIDENT

The utility operating the Salem Nuclear Power Station at Salem, New Jersey--Public Service Electric and Gas Company--(PSE&G)--is requesting a license from the United States Nuclear Regulatory Commission to store indefinitely up to 1170 highly radioactive, spent nuclear fuel rod assemblies in each of two spent fuel storage pools located at the reactor site. The Station consists of one operating nuclear power reactor and one under construction. Each spent fuel pool is housed in a separate fuel handling building which is located next to its respective reactor containment building. Originally, it was intended only to have in storage about 64 spent fuel assemblies at any one time in each pool, as the plan was to ship spent fuel away from the site for disposal after a brief, 150 day cooling-off period that allows the radioactivity and associated heat in the spent fuel to decay substantially. Now, however, PSE&G proposes to increase the storage capacity of each storage pool, by replacing the original design of the storage racks with a rack design which allows the spent fuel assemblies to be packed in the pool at a high density (compaction). The proposed increase in storage capacity would increase the amount of long-lived radioactivity to be stored in the pool eighteen-fold. Approval to increase the storage capacity is requested by PSE&G because there presently exists no nuclear waste disposal system for disposing of the spent fuel.

With respect to the hazards of the proposed spent fuel storage increase, it is contended that:

(a) The proposed design changes to the spent fuel storage pools would <u>greatly</u> increase the nuclear accident hazards of the

Salem Station with respect to the health and safety of the public.

(b) The proposed design changes would create many severe accident possibilities which would have the potential for extremely disastrous consequences. Such accidents would involve the loss-of-pool-water, hereafter denominated the loss-of-water accident.

(c) Both the PSE&G's Safety Analysis Report and the Nuclear Regulatory Commission's Safety Evaluation Report for the proposed design changes fail to analyze the loss-of-water accident.

(d) The potential consequences of loss-of-water accidents are so serious that the utility (PSE&G) and the Nuclear Regulatory Commission's staff must analyze them, and the Atomic Safety and Licensing Board (AS&LB) and the Commission itself must investigate and consider them for both their likelihood and potential harmful consequences, in order to enable the Nuclear Regulatory Commission, that is, the Commission, itself, to responsibly form an opinion as to whether the proposed spent fuel storage would be "inimical to the health and safety of the public" (referring to Section 103 of the Atomic Energy Act) and to responsibly inform the public of the full risks to health and safety.

(e) The likelihood of a loss-of-water accident occurring is <u>not</u> remote or extremely low; but rather, the probability of occurrence is indeterminable. More specifically, it cannot be

proven mathematically or statistically that the probability of such an accident occurring in the time period of the life of the plant or even a decade is less than 100% or significantly less than 100%. There exists an indeterminable but extremely large number of possibilities for potentially or conceivably causing a loss-of-water accident in a storage pool. Furthermore, many incidents associated with nuclear power reactors of near-accidents, equipment malfunction accidents, and human error have occurred. These facts indicate that the probability of a loss-of--water accident is high, not low. Because of these facts, plus the fact that the probability of a loss-of-water accident is indeterminable and the fact of the extreme potential for harmful consequences of such an accident,make the proposed storage facility unsafe.

(f) The Nuclear Regulatory Commission's current practice of evaluating the risks of the worst or severe nuclear accident possibilities by considering only the likelihood of such accidents, and not evaluating and considering the potential harmful consequences, is not consistent with the well-established method of assessing accident risks, which is to consider both the likelihood and the consequences of accidents.

2. PHYSICAL CONSEQUENCES OF A LOSS-OF-WATER ACCIDENT

(a) The radioactivity in spent fuel generates heat which must be dissipated in order to prevent the spent fuel assemblies

from overheating. For this reason and for radiation shielding purposes, the spent fuel assemblies are stored under water. The pool water serves to remove the heat of the radioactivity. The pool water in turn is cooled by water circulating cooling systems to prevent the pool from overheating and boiling dry. In a loss-of-water accident the spent fuel assemblies will heat up to a high temperature, because natural air convection and thermal radiation heat dissipation processes are insufficient to cool the spent fuel. The full potential for spent fuel heatup has yet to be predicted by a thermal/ hydraulics analysis.

(b) Upper bound calculations exist which indicate that the potential may exist for the uranium dioxide in the spent fuel to heat up beyond its melting temperature of about 2800°C, even if all of the spent fuel were stored for ten years.

(c) Calculations exist which tend to set a mathematical lower bound of the spent fuel heatup potential; and these calculations indicate that as a minimum the zirconium (zircaloy) fuel rod cladding material will heat up to 900°C and catch on fire for spent fuel that has decayed (aged) for three years. These calculations were performed by Sandia Laboratory and are presented in a report titled "Spent Fuel Heatup Following Loss of Water During Storage" (SAND77-1371, Sept. 1978, draft), by A. S. Benjamin, et al.; hereafter called the Sandia Report. The Sandia Report does not calculate the fuel temperature rise beyond the point when the temperature is calculated to reach the zirconium fire ignition temperature, and subsequent zirconium clad melting (1857°C).

(d) A zirconium fire would generate substantial additional heat with the potential for melting away the cladding of the fuel rod and also melting the uranium oxide fuel or raising the fuel to its melting temperature of 2850°C (about).

(e) A zirconium fire which starts in relatively new spent fuel (say, three year storage or less), which would include 16% of the total planned storage or less, could conceivably spread to old spent fuel and thus engulf the whole load of spent fuel in the pool.

(f) Severe zirconium explosions are conceivable, due to zirconium-water reactions in conjunction with zirconium-air reactions.

(g) Hydrogen explosions are conceivable due to the hydrogen released in a zirconium-water reaction and its reacting with air.

(h) Since zirconium fuel clad melting is possible, it is conceivable that the air flow passages inside the spent fuel rod assemblies could become plugged due to the zirconium dioxide reaction product and due to molten zirconium running down toward cooler portions of the spent fuel and freezing there. Plugged air flow passages would greatly worsen the spent fuel heatup. Also, explosive zirconium-water reactions and hydrogen explosions could conceivably damage adjacent spent fuel so as to constrict air flows and thus worsen the spent fuel heatup in these assemblies as well.

(i) Strontium-90, Cesium-137 and Plutonium are the dominant radioactive substances in spent fuel from a public health risk standpoint. It is conceivable--meaning that it has not been ruled out scientifically--that a near 100% release of Strontium-90 and Cesium-137 radioactivity from the spent fuel into the atmosphere would occur in a spent fuel heatup excursion in a loss-of-water accident. For such a near-100% release to occur, the spent fuel need not necessarily reach melting temperature, but need only attain a level of only about 1900°C and maintain that temperature for a day or so. The Strontium-90 and Cesium-137 could then diffuse out of solid UO_2 fuel at such temperatures. This assumes that the fuel rods have lost their zirconium cladding upon meltdown of the zirconium but that the rods would maintain their rod shape because the UO_2 fuel pellets inside the fuel rods would have sintered together during reactor operation to form a long UO_2 rod capable of maintaining its shape. If the UO_2 rods should crumble, air cooling would be further impeded and lead to higher UO_2 temperatures and consequently a greater thermal potential for strontium and cesium diffusion out of the UO_2 fuel.

(j) Calculations exist which indicate that the air inside spent fuel storage building would heat up and pressurize due to the heat of the spent fuel (the building would become like an oven). The air pressurization would burst open the building and thus allow the radioactive vapor and smoke to escape into the atmosphere. If the building vents were opened, the radioactive

vapor and smoke could conceivably escape through these vents. Zirconium and hydrogen explosions could conceivably rupture the building as well, to allow the escape of radioactivity.

(k) No experimental data or theoretical analyses exist on which to establish the potential for release of plutonium in the spent fuel into the atmosphere in a loss-of-water accident. Steam explosions, hydrogen explosions, and zirconium explosions are conceivable mechanisms which could pulverize large quantities of spent fuel bearing plutonium and blow it into the outside environment, where the plutonium would then spread through the environment.

(l) Calculations exist which indicate that the Salem spent fuel storage building could not be modified to eliminate the possibility of a zirconium fire occurring in a loss-of-water accident. The Sandia Report suggests the possibility of modifying the building to provide for an open chimney effect: a large hole in the ceiling and a large hole at the floor level of the building side wall, to allow perfect room air ventilation during a loss-of-water accident to expel the heated air exiting from the spent fuel assemblies. The holes or openings would be normally closed by large doors, which would be opened in a loss-of-water emergency to create the chimney effect. Such a chimney effect by expelling heated air, would tend to limit the spent fuel heatup temperatures, according to Sandia's analysis, but would not eliminate the possibility of a zirconium fire. Since such a chimney feature would not eliminate the possibility of a zirconi-

um fire, a chimney could conceivably not have any mitigating effect at all; for the building openings would provide unlimited air (oxygen) to promote the spreading of the fire and would provide ready access of radioactive vapors and smoke to the outside atmosphere. Nor would the activation of the chimney (automatic or manual opening of its doors) be reliable in the case of a severe reactor accident which causes a spent fuel loss-of-water. A severe reactor accident can potentially cause such a high level of radiation in and around the site that the whole site operating crew could flee in panic, leaving the spent fuel pool and related safety and cooling systems unattended. Under such a panic situation, it would not be expected that the chimney doors, if incorporated into the building, would be opened.

(m) A reduction in the number of spent fuel assemblies stored in the pool could not eliminate the possibility of a zirconium fire occurring in a loss-of-water accident, nor preclude the possibility of a loss-of-water accident.

(n) Emergency efforts to cool the spent fuel following a loss of pool water could conceivably worsen the accident or otherwise have no mitigating effect. Spraying the overheated spent fuel with water (which would have to be done remotely, due to the heavy radiation emanating from the spent fuel) would cause zirconium-water reaction that could promote the ignition or spreading of a zirconium fire, or cause explosions. Moreover, the ..eatup of the spent fuel could conceivably cause the boral

neutron absorbing material to meltdown, leaving a region of spent fuel without enough neutron absorption to prevent a criticality should the pool be reflooded. Furthermore, the heat of the spent fuel in a loss-of-water accident (and possible explosions) could conceivably damage the spent fuel to such a degree that the pool would continuously leak heavily, should the pool be reflooded, which would result in a heavy seepage of radioactivity into the ground and nearby waters.

(o) In order to evaluate the potential for radioactivity release in a spent fuel pool loss-of-water accident, a thermal analysis must be performed, of course. The only mathematical theory of spent fuel heatup which exists in a form for ready use (besides this author's theory) is the SFUEL computer code of the Sandia Laboratory, which is described in the above-mentioned Sandia Report. The Sandia Report analyzes the loss-of-water accident for a spent fuel storage pool which is close to the Salem design. However, the Sandia Report is not sufficient for evaluating the spent fuel heatup potential for Salem (nor any other spent fuel storage pool); and, furthermore, the SFUEL computer code is not sufficiently developed and verified to provide reliable heatup temperature prediction with reasonable accuracy. To elaborate:

(1) The Sandia Report does not investigate the spent fuel temperature excursion beyond the ignition of the zirconium or zirconium melting.

(2) The Sandia Report does not analyze the high-density storage rack design for the case of imperfect building ventilation, which is the case for all pressurized water reactor (PWR) storage pools, including Salem.

(3) Sandia's mathematical theory (SFUEL) contains serious theoretical deficiencies which, based on independent scoping calculations, may be causing the code to be drastically underpredicting spent fuel heatup temperatures. Foremost are the assumptions in the SFUEL theory that the temperatures of the fuel rods in a given spent fuel rod assembly and at a given elevation are the same (uniform temperature distribution horizontally), and that the temperature distribution inside a fuel rod at any given elevation is also uniform.

(4) Sandia's mathematical theory is not adequately described in the Sandia Report, and requires a systematic checking to verify the code theoretically and calculationally.

(5) A reliable mathematical theory of spent fuel heatup may not be practical, due to computer limitations.

(6) Sandia's SFUEL theory has not been experimentally verified, contrary to the claim made in the Sandia Report that adequate experimental data exists

to validate the SFUEL theory. There experiment relied on in the Sandia Report consisted of two heated plates held at a low, constant and uniform temperature cooled by natural air convection; whereas the situation in a spent fuel heatup accident is one of a highly variable temperature distribution and extreme air temperatures in a rod bundle configuaration. Moreover, thermal radiation heat transfer aided by thermal heat conduction, appear to be a crucial heat transfer processes in a spent fuel heatup, which were totally absent in the two-heated-plate experiment cited in the Sandia Report. To adequately account for thermal radiation interchange among, and heat dissipation from, spent fuel rods in a storage pool under a loss-of-water accident, it would be necessary to conduct an experiment which includes a large scale loading of simulated spent fuel (electrically heated) or actual spent fuel. Because the electrical resistance of electrical heater filaments is dependent on temperature, an adequate simulation of spent fuel heatup may not be possible with electrically heated rods; in which case it may not be possible to experimentally verify a mathematical theory of spent fuel heatup, because it would not be practical or safe to conduct such tests with spent nuclear fuel rods.

(7) In short, the Sandia Report must be critically evaluated.

(p) It would not be practical or safe to experimentally investigate the radioactivity release potential of a loss-of-water accident; particularly in the event of a zirconium fire, zirconium melting, explosion, or other severe process which causes significant changes in the fuel's physical condition, because the fuel temperature excursion and the interrelated radioactivity release would both depend on the physical condition of the fuel and on the size of the spent fuel mass undergoing a loss-of-water accident. Moreover, the behavior of the spent fuel may be a function of the prior aging of spent fuel in water and the physical history of the spent fuel when it was in the reactor, such as whether the fuel had undergone overheating in the reactor in an accident.

(q) It is not possible to accurately predict the course of a loss-of-water accident once the zirconium cladding becomes ignited. Instead, only mathematical upper bound estimates of the radioactivity release potential could be developed, which presently do not exist. A near-100% release of radioactive strontium and cesium is plausible and could not be disproven.

(r) The Salem Safety Analysis for the proposed spent fuel storage supplies inadequate information on which to perform heatup calculations; for example, the pool and building dimensions are not given.

3. POTENTIAL HARMFUL CONSEQUENCES OF THE RADIOACTIVITY RELEASE FROM A LOSS-OF-WATER ACCIDENT

(a) Each spent fuel storage pool at Salem would contain at capacity forty-five million curies of Strontium-90 radioactivity and about the same amount of curies of Cesium-137. For comparison the United States Atomic Energy Commission's report Theoretical Possibilities and Consequences of Major Accidents In Large[1] Nuclear Power Plants (WASH-740, March, 1957) calculates that the release of 0.15 million curies of Strontium-90 (150,000 curies) could cause agricultural restrictions over a land area equal to 150,000 square miles, which is the size of New Jersey, New York, Connecticut, Massachusetts, Rhode Island, Vermont, New Hampshire, Maine, and half of Pennsylvania, combined. A loss-of-water accident in one Salem spent fuel storage pool could conceivably release nearly all of the forty-five million curies of Strontium-90, or three hundred times the WASH-740 assumed release quantity of Strontium-90.

Assuming that land which is contaminated more than fifty times the WASH-740 contamination limit for Strontium-90 would be ruined agriculturally, which is a prudent assumption and one which is consistent with the view taken in the WASH-740 Report (the WASH-740 Report asserts that Strontium-90 land contamination at ten times the Report's contamination limit would require prohibiting dairying for a very long time), it can be calculated

[1] The size of the power reactor assumed in the WASH-740 Report is not "large" compared to present size plants. Specifically, WASH-740 assumes a 500 megawatt thermal (M Wt.) reactor whereas each Salem reactor has a rated pwer outputof about 3300 M Wt.

that a spent fuel pool loss-of-water accident which releases forty-five million curies of Strontium-90 (which cannot now be shown to be impossible) could result in ruining agriculturally a land area of the size of about one-third of the land East of the Mississippi River, or certainly the entire eastern seaborad of the United States and Canada, for a hundred years or more.

(b) The release of Cesium-137 radioactivity from the storage pool into the atmosphere could result in high levels of gamma radiation (intense x-ray-like radiation) emanating from the ground over an area equal to 150,000 square miles. The gamma radiation exposure to persons standing on the ground could potentially occur at a rate which exceeds by a factor of thirty-eight or more the health limit recommended by the United States Environmental Protection Agency of 25 millirems per year for total radiation exposure from emission of radioactivity due to nuclear power.

(c) No reliable estimates exist of the potential cancer and genetic harm that could result from a near full release of Strontium-90 and Cesium-137 (and other volatile radioactive materials) in a spent fuel loss-of-water accident. Such estimates are necessary and should be developed, in order that the spent fuel accident hazards can be fully evaluated.

(d) The contamination levels indicated in (a) and (b) above apply to the boundary of the fallout land area zones that are quantified in those sections. In the interior of the zones and closer to the plant the contamination levels would be much worse.

(e) One spent fuel pool at Salem would contain the equivalent of thirty-nine tons of Plutonium-239 alpha-radioactivity. If dispersed uniformly, this amount of plutonium would have the potential for causing abandonment of about five million square miles of land, which is 1.5 times the total United States land area, including Alaska. No analysis exists which proves that an area of the size of New Jersey, say, would not require permanent abandonment due to a plutonium release in the event of a loss-of-water accident in one spent fuel storage pool.

(f) It is possible that a reactor accident at the Salem Station could induce loss-of-water accidents in both spent fuel storage pools, which would then double the above estimates of potential harmful consequences.

(g) Even if the spent fuel pool held a minimum of spent fuel--sixty-five fuel assemblies, or one-third of a core, as was the original intent--the potential consequences of a loss-of-water accident would still be extreme: for example, a land area of the size of Ohio, or five times the size of New Jersey, could be ruined agriculturally for a hundred years or more, due to Strontium-90 release alone.

4. POSSIBLE LOSS-OF-WATER ACCIDENTS: SPECIFIC POSSIBILITIES

A loss-of-water accident is possible, which can happen if the pool water cooling system should break down. A boil-off of the pool water is possible in such an event, which would take

about four days to two weeks, based on the figure for the "maximum evaporation rate" (56 gallons per minute) given in the Nuclear Regulatory Commission's Safety Evaluation Report (p. 2-5). The most likely cause of a breakdown in the pool water cooling system is a severe reactor accident (see contention No. 7 below). A severe reactor accident could result in such heavy radiation levels at the reactor site that the storage pools would be abondoned. In that event the cooling system would have to be assumed to breakdown; and there would be no adequate assurance that makeup water could be supplied to the pool. Such a reactor accident must be assumed to be highly likely to occur (see contention No. 7).

There are other possibilities for cuasing a loss-of-pool-water accident through a breakdown in the pool water cooling system which must be given serious consideration. One such possibility is for the reactor plant to have to be permanently closed down due to a reactor accident, leaving only a very small crew to perpetually watch over the storage pool and maintain perpetual cooling. In this situation, a cooling breakdown could occur through negligence and not be corrected. Sabotage and acts of war are other possibilities.

5. CONCEIVABLE POSSIBILITIES FOR LOSS-OF-WATER ACCIDENTS

There are a number of conceivable possibilities of accidents and sabotage which could result in a loss-of-water accident and which, therefore, must be evaluated for their likelihood and

their potential for causing a loss-of-pool-water. They are:

(a) Spent Fuel Shipping Cask Drop.

It appears to be possible for the heavy shipping cask to fall from its crane into the storage pool. Such an incident should be evaluated for the potential for rupturing the pool and causing rapid drainage of the pool. A crane failure has already occurred over a spent fuel storage pool (Shippingport); and an incident of improper handling of a spent fuel pool cask has already occurred (Big Rock Point).

(b) Criticality.

Indications are that it is possible for a local criticality to occur in the storage pool (see contention No. 6 below). Such a criticality has yet to be evaluated for the course it could take; so no upper bound exists of its thermal and mechanical consequences. It may be possible that the fission heat generated by such a criticality could cause a rapid boil-off of the pool water, despite the pool water cooling system.

(c) Sabotage and Terrorism.

The possibilities for sabotage and acts of terrorism are very real. The use of explosives could destroy the cooling system, and the removal of a new spent fuel assembly out of the pool water would produce such high levels of radiation in the pool building that action to supply makeup water would be severely impeded. Also, explosives conceivably could be used to rupture the pool and thereby cause rapid drainage.

(d) Others.

Under this heading, earthquakes breaking open the pool and large airplane crashes should be considered.

6. CRITICALITY ACCIDENTS

A criticality accident in the spent fuel pool is a very real possibility. Possible causes are as follows:

(a) Missing boral plates in a local region of a storage rack, or boral plates with a deficient amount of Boron-10; and

(b) Underprediction of the effective neutron mulliptication factor (K_{eff}). Public Service Electric and Gas Company's Safety Analysis Report and the Nuclear Regulatory Commission's Safety Evaluation Report do not provide adequate information to assess the hazard of a criticality accident. For example, there is no indication that there would not occur any positive reactivity feedback effect during the fission power rise in a criticality situation. It is a valid concern that a criticality might lead to a rapid boil-off of the pool water. The radiation from such a high-power criticality could conceivably obstruct efforts to control the accident. In order to assess the criticality hazard, therefore, it is necessary that a full analysis be made of all possible courses a criticality may take.

The benchmark critical experiments used by Public Service Electric and Gas Company to verify its mathematical theory for calculating (K_{eff}) are not adequate to verify the accuracy of the predicted (K_{eff}) factor. Those experiments should only be considered as a means to develop the theory for design

purposes. In the final analysis, the loading of fuel assemblies into the racks will be the proof of the validity of the predictions of (K_{eff}). Therefore, it would be necessary to perform an experiment in which new fuel is placed in the storage racks under controlled insertion and neutron monitoring for criticality. This should be a practical confirmatory experiment. It is well-established that such an experiment is necessary. Also, consideration should be given to the question of whether local boiling in a number of spent fuel assemblies could cause an increase in (K_{eff}); that is, whether the fuel in the storage racks would be over-moderated. In this regard the above described experiment should investigate the effect of voids and water temperature.

(c) It is conceivable, too, that spent fuel--particularly, the uranium dioxide--could melt (at 5000°F) and thus form a liquid pool of molten fuel within a frozen shell or crust of uranium dioxide and steel and zirconium. Under this condition, it is conceivable that the plutonium in the molten uranium dioxide could separate and stratify in such a pool--or at least a mass of fuel material could form which is rich in plutonium--and create as a result a nuclear fuel mass capable of generating the same kind of atomic reaction which takes place in an atomic bomb--a runaway reaction which could produce a strong nuclear explosion that would increase the dispersal of the radioactivity, into the environment, especially the plutonium. Plutonium might not escape heated solid fuel rods as readily as Strontium-90 and Cesium-137; and so pulverization or vaporization of the fuel may be required,

as in a nuclear explosion, before a large amount of it (plutonium) could be released into the atmosphere. (This nuclear explosion possibility is similar to the mechanism which has been speculated to have caused the "nuclear disaster" in the Soviet Union, namely, a concentration of plutonium in a nuclear wate burial trench.)

7. REACTOR ACCIDENTS CAUSING A SPENT FUEL POOL LOSS-OF-WATER ACCIDENT

(a) Severe reactor accidents are the most likely cause of a loss-of-water incident in a spent fuel storage pool. A severe reactor accident could result in such heavy radioactive contamination in the area of the spent fuel storage pool and building that the entire operating crew would be forced to flee for their lives. Such high radiation levels would persist for months and thus would prevent emergency crews from returning to the spent fuel pool building to maintain the pool cooling system. In such an accident, it is likely that the cooling system would breakdown, due to a lack of maintenenace, which would lead to a rapid boil-off of the pool water. In addition, the spent fuel storage pool for Unit No. 2 would suffer the same consequences. (Indeed, the Unit No. 2 reactor would likely be abandoned as well, setting in motion a train of events leading to a core meltdown and possible explosion in that reactor as well).

(b) there exists a great number--essentially an infinite number of severe reactor accident possibilities that could result in a loss-of-water incident in the spent fuel storage pool.

(c) Severe reactor accident possibilities have never been investigated and analyzed by the Nuclear Regulatory Commission and its Atomic Safety and Licensing Board for the potential consequences or the likelihood of such accidents, except to a limited degree in the Nuclear Regulatory Commission's Reactor Safety Study (Rasmussen Report), which is not an adequate hazards' analysis to assess the reactor accident risks (see contention No. (f)(6) below). It is contended that it has not been ruled out by scientific concensus that the potential harmful consequences of a servere reactor accident causing radioactive contamination could be:

(1) 120,000 square miles of land requiring evacuation or living restrictions.

(2) A lethal range of seventy-five miles of a released radiation causing acute radiation disease.

(3) 500,000 square miles of land requiring agricultural restrictions due to the release and fallout over the land of Strontium-90 alone; and

(4) If the living and agricultural restrictions are relaxed substantially, about 100,000 to 500,000 additional cancer deaths could result. From the figures, it can be appreciated that there exists the potential for causing abandonment of the spent fuel storage pools in the event of a severe reactor accident.

(d) The proposed increase in the storage of spent fuel in each storage pool from about 65 spent fuel assemblies to 1170 spent fuel assemblies, amounts to an eighteen- fold increase in the quantity of spent fuel and hence Strontium-90 and Cesium-137 radioactivity to be stored. Since the core of one reactor would contain about 3.7 million curies of Strontium-90 which, if released in a reactor accident, would have the potential for causing agricultural restrictions over 500,000 square miles, and since one storage pool would, by the proposed storage increase, contain forty-five million curies of Strontium-90, or twelve times more Strontium-90 than in the core of the reactor, which could conceivably be released into the atmosphere in a loss-of-water incident, it is imperative that the most likely causes of a loss-of-water incident in a storage pool, namely, severe reactor accident possibilities, be investigated. Severe reactor accident possibilities cannot be considered independent of spent fuel storage loss-of-water accidents. From a radiological health standpoint, and in view of the fact that Strontium-90, Cesium-137 and Plutonium are among the most biologically hazardous radioactive substances, if not the most hazardous, the proposed storage increase would so greatly increase the potential consequences of reactor accidents that the issue of the likelihood of severe reactor accidents must be thoroughly and completely investigated. (The proposed storage increase is like proposing the construction of twenty-four large power reactors from a radiological hazards standpoint, particularly with respect to Strontium-90, Cesium-137, and Plutonium release potentials).

(e) The Nuclear Regulatory Commission has announced on January 18, 1979 that it supports the "use of probabilistic risk assessment in regulatory decision making," in other words, the making and considering estimates of the numerical probability of severe reactor accidents. However, it is contended that the probability of a severe reactor accident occurring within the next twenty years or so which results in a loss-of-water accident in a storage pool cannot be proven to be significantly less than 100%, and that, therefore, probabilistic risk assessment methods should not be used to assess the risks of the proposed storage increase.

It is contended that in order to safely judge the overall safety or hazards of the Salem reactors and associated storage pools, the applicants (utility) and their nuclear plant designers and supplier and/or the Nuclear Regulatory Commission must analyze and evaluate all known accident possibilities (such as multiple control rod ejection accidents, including chain reaction ruptures of control rod drive mechanism housings, loss-of-coolant accidents without SCRAM, ejection of a high reactivity worth control rod, and power excursions with excess boron concentration in the coolant) for both their likelihood of occurrence and their potential consequences, and publish the entire analysis and evaluation (that is, without reduction or simplification), as well as a reduced, simplified summary. Furthermore, the Nuclear Regulatory Commission should accept and hear testimony from all parties on the adequacy of such an analysis and evaluation, and

should accept general testimony as to the likelihood and consequences of all possible serious reactor accidents (that is, the testimony should not be limited to the scope of the applicant's present safety analyses or the analysis and evaluation called for above, but receive independent analysis as well), and should fully consider and fully weigh all of the testimony and analyses and evaluations as above described in forming its opinion on the application. The called for analysis and evaluation of all possible accident--their likelihood and consequences--should also include:

(1) A listing of all theoretical uncertainties with regard to the possibility for worse consequences than predicted and the combined effect of the uncertainties.

(2) An identification of all parts of the analyses which have not been experimentally verified.

(3) A detailed fault tree graph for each accident possibility and a graph of the chain of events and equipment failures and human errors for each accident possibility; and

(4) A compilation of all experiences of reactor equipment failures and human error related to each accident chain of events.

It is further contended that a severe reactor accident which would likely cause a loss-of-water accident in a spent fuel storage pool is likely to occur--that is, such an accident can

reasonably be expected--based on the fact that there is seemingly an infinite number of such accident possibilities, and based on the large potential for human error and carelessness and other human failings, and on the experience record of equipment malfunctions, past reactor accidents, and near-accident incidents.

(f) The following additional contentions regarding reactor accidents are offered:

(1) The theoretical predictions of the course of the reactor design basis accidents have not been adequately verified experimentally. The accidents of most concern are the loss-of-coolant accidents, the control rod ejection accident, coolant pump seizure, control rod withdrawal accident, and the anticipated transients without SCRAM (that is, without emergency fast shutdown of the fissioning). For examples of particulars, see The Accident Hazards Of Nuclear Power Plants by Richard E. Webb (University of Massachusetts, 1976), Chapter 4 and 9. The applicant's reactor safety analysis reports do not give adequate scientific reasons why full-scale reactor tests are not necessary, nor do the reports even address the question of the necessity of full-scale or even large-but-less-than-full-scale tests.

(2) The theoretical analyses of the design basis accidents have a number of theoretical and mathematical shortcomings. See examples in chapter four of The Accident Hazards of Nuclear Power Plants.

(3) The safety analysis reports submitted by the applicant do not justify the selection of the reactor design basis accidents relative to possible accidents which are more severe.

(4) The reactor design basis accidents are analyzed in the applicant's safety analysis report with the added assumption in some cases of a single additional failure of some component in the safety systems intended to control the accident. However, the applicant's and the Nuclear Regulatory Commission's analyses do not give adequate analysis and consideration of past reactor accidents and near-accident incidents, some or most of which occurred by and with multiple malfunctions and human error. This is further reason why the full analysis and evaluation of all accident possibilities--their likelihood and potential consequences--should be prepared and considered. The Nuclear Regulatory Commission's "single failure criterion" to judge accidents worse than the design basis accident as "incredible" is wholly inadequate to assure safety, and should not be a basis to deny the full investigation of all accident possibilities as called for above.

(5) The magnitude of the potential consequences discussed above requires that the Nuclear Regulatory Commission should require the analysis and evaluation

of the likelihood and potential consequences of all accident possibilities, as described by the above contentions, and should fully consider and fully weigh the said likelihood and consequences of all accident possibilities, and should fully consider and fully weigh the said likelihood and consequences in the light of the experience of past reactor malfunction (see Accident Hazards generally, and chapters 5 and 6, including the section on Probability of Accidents, pp. 96-98 and appendix 2, and the testimony by D. Bridenbaugh, et al., before the Joint Committee on Atomic Energy of the U. S. Congress, February 18, 1976, which suggest that the likelihood of such severe accidents is not remote and may be unacceptable). A sound, rational judgment of reactor safety is not possible without the full analysis and evaluation called for in the above contentions.

(6) The Nuclear Regulatory Commission relies on the before-mentioned Rasmussen Report and a review of that Report known as the Risk Assessment Review Group Report (Lewis Report) to judge that the risk to the public health and safety due to the accident possibilities which are more severe than the design basis accidents is acceptably low and that the more severe accidents need not be further considered. It is contended that the Rasmussen Report and the Lewis

Report have fundamental shortcomings which preclude their being used to establish the level of risk of the said severe accident possibility. See Accident Hazards, chapter six and appendix one, and the reviews of the Rasmussen Report by the United States Environmental Protection Agency, dated August, 1975 and June, 1976 (EPA-520/3-75-012 and EPA-520/3-76-009), for discussions of some of the shortcomings. For example, the most severe class of reactor accidents, namely nuclear runaway, are not analyzed for their likelihood and consequences in either the Rasmussen Report or the Lewis Report.

Other shortcomings of the Rasmussen Report are: The report does not present the analysis of the probability of the severe accidents which the report considered, such as transients-without-scram; rather the report merely gives the results of the analysis performed by the Rasmussen study group, by the use of simplified, "reduced" fault trees, for example. In one extremely important instance, at least, there is no fault tree given at all, specifically, for the accident involving the failure of the recirculation pump trip safety action during an "anticipated transient without scram" (though this is a boiling water reactor accident, there likely are instances for the pressurized water reactor in the report as well, for I recall no fault tree for coolant pump siezure and

control rod ejection accidents). The public is being asked, therefore, to accept the results of the Rasmussen Report and the Lewis Report on faith. This prevents others from being able to adequately scrutinize the probability evaluation of the Rasmussen Report for its accuracy, completeness, and validity of assumptions (explicit and implicit), which are mostly subjective. Moreover, the simplified analysis presented in the Rasmussen Report contains symbols which are not defined adequately for purposes of examining the safety systems for their potential for, and the likelihood of, malfunction.

Overall, it is contended that the applicant's and the Nuclear Regulatory Commission's safety analysis reports are not an adequate basis for assessing the safety of the proposed Salem pressureized water reactor and its storage pools, and that the Rasmussen Report and the Lewis Report are not an adequate supplement to answer the concerns of these contentions.

(7) The reliability of the SCRAM system to control accidents has not been adequately demonstrated. (SCRAM means the rapid insertion of the reactor control rods, which shuts down the atomic reaction). No backup SCRAM system exists. The applicant has not adequately demonstrated that a backup scram system is

unnecessary, inasmuch as the pressure surge of anticipated transients without scram may be too high.

(8) The integrity of the reactor containment system under a design basis accident (loss-of-coolant) has not been adequately confirmed experimentally. Full-scale tests appear to be necessary.

(9) The appplicant's design basis accident for the containment system and the emergency core cooling system (ECCS) has not been shown to be the most likely form of a loss-of-coolant accident. Specifically, the applicant has not demonstrated that a loss-of-coolant accident will not more likely occur as a result of a strong pressure surge transient. Stronger coolant pressures would produce stronger forces on the various components of the containment systems. As for the ECCS, a stronger coolant pressure may be the result of a transient that produces a hotter core at the time of the coolant system rupture. The ECCS is not designed to control the higher pressure and hotter core (higher temperatures) of such a loss of coolant accident. This point has been demonstrated by the recent Three Mile Island reactor accident.

(10) It is contended that there should be additional consideration of an earthquake producing a loss-of-coolant accident, inasmuch as a prototype reactor

plant will not be proof tested by simulated earthquakes (due to obvious impracticality).

(11) It is contended that the spontaneous reactor vessel rupture type of accident and a vessel rupture due to pressure surges of anticipated transients without scram have not been adequately demonstrated to be of negligible probability (to warrant their neglect in the reactor containment design). There is the question of no leak-before-break warning.

(12) The applicant's safety analysis has not given adequate consideration for the possibility, perhaps the likely possibility, that a severe reactor accident will occur as a result of unforseen causes or effects, as that seems to be the experience of accidents or near-accidents in nuclear power plants.

(13) The applicant's safety analyses have given inadequate consideration to the possibility of common-mode type failures in the coolant piping and the emergency core cooling system piping, especially the possibility for sequential failure of the latter due to the forces generated by the former.

(14) the applicant has given inadequate consideration to the possibility of sabotage, for example, consideration should be given to the lack of provision for separate rooms and blast shielding in between, to separate backup safety systems, instrumentation, and

cables from primary equipment in rooms normally unattended, to minimize the likelihood of a saboteur's bomb knocking out primary and backup safety equipment at once. Also, a multiple control rod ejection accident could easily be caused by a saboteur's bomb.

(15) Amplification of the preceding contention along with supporting arguments and information are given in the following documents, which have never been disputed by the Nuclear Regulatory Commission:

a. The Accident Hazards Of Nuclear Power Plants, R. E. Webb.

b. Memorandum in support of the contention of the Coalition for Safe Energy in the construction permit hearings for the proposed Erie pressurized water reactor (Docket No. STN-50-580 and 581), dated September 26, 1977, which treats issues concerning the emergency core cooling system; specific possibilities of "anticipated transients without scram" and their likelihood; the need for full scale testing of analyses of certain accidents; kinds and causes of multiple control rod ejection accidents; power excursions with excessive boron concentration in the coolant; loss-of-coolant accidents without scram; and common mode failures in coolant piping and emergency core coolant piping in loss-of-coolant accidents.

c. Remarks by R. E. Webb before the Nuclear Regulatory Commission's Atomic Safety and Licensing Board on the said Erie proceeding, July 28, 1977, Transcript pages 81-176, defending his contention.

d. Petition to Congress "Calling for a Full Review and Investigation of the Hazards of Nuclear Power Plants and Radioactive Waste Disposal," by R. E. Webb, May 20, 1978, including an appendix titled "Remarks on the Crucial Factor of the Surface Contamination Limits for Plutonium and Strontium-90."

8. PERMANENT SPENT FUEL REPOSITORY AT SALEM

(a) It is contended that it is likely that the spent fuel from the Salem reactors will be stored permanently in the on-site storage pools--that the Salem reactor site will become a permanent repository for the high-level radioactive, spent fuel generated at the plant.

(b) There presently exists no geologic nuclear waste repository for disposing of the spent nuclear fuel; and no such repository is likely to be developed and demonstrated to be safe, or permitted to be built and operated.

(c) Off-site spent fuel storage pools which store only aged spent fuel assemblies (older than six months or a year) have catastrophic loss-of-water accident possibilities as well as the reactor site storage pools. Such off-site pools have yet to be

evaluated fully for their spent fuel heatup and radioactivity release potential in loss-of-water accident. Furthermore, the theoretical deficiencies in Sandia's mathematical theory (SFUEL) for spent fuel heatup in a loss-of-water accident, which are discussed in contention no. 2.(o) above, may very well mean that the heatup predictions presented in the Sandia Report for off-site pools may be grossly in error in the unsafe direction. Therefore, off-site storage pools cannot be considered a safe alternative for storing Salem spent fuel; nor does it appear to be an economically viable alternative.

(d) Even if the spent fuel were not allowed to accumulate in the Salem pools, there will be at least sixty-five new spent fuel assemblies stored in each pool at any one time, which means that there would be about 2.5 million curies of Strontium-90 in each storage pool (and a like amount of Cesium-137). Information recently developed about the loss-of-water accident hazards of spent fuel storage pools reveals that it is conceivable that the Strontium-90 and Cesium-137 could be released from the fuel into the outside atmosphere in such an accident (even for open, low density storage racks). Thus, a loss-of-water accident in a single spent fuel pool could result in ruining agriculture over a land area equal to three times the size of New Jersey, among other disastrous consequences. The combined release of radioactivity from a reactor accident and two spent fuel storage pools (as a consequence of a reactor accident) would be about three times worse.

(e) The only way to avoid the risks of spent fuel storage, therefore, is to cease generating the radioactivity by closing down the station and terminating the construction of Unit II--that is, to revoke the reactor licenses.

9. IMPRACTICALITY OF THEORETICAL ANALYSIS AND EXPERIMENTAL VERIFICATION

The preceding contentions describe a broad scope of theoretical analyses and experiments that would be necessary in order to fully evaluate the hazards of spent fuel storage (and reactor accidents). However, it is contended that it is not practical (humanly possible) to prepare the needed analyses nor to conduct the needed experiments; and, therefore, the full hazards could never be scientifically established, except by assuming the worst conceivable consequences--that is, a near full release of radioactivity from the storage pool.

10. CONCLUSION

Spent fuel storage at Salem (and any other reactors) is unsafe because loss-of-water accidents are possible and because the potential harmful consequences are extreme. Closing down the reactor is the only responsible course of action. This would eliminate the risk of reactor accidents, which itself is extremely grave.

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1. Ph D in Physics. Nuclear Reaction Physics and Engineering, Ohio State University (March 1972).

2. B S in Engineering Physics. University of Toledo (1962).

3. Commission in U. S. Naval Officer Candidate's School. Newport, Rhode Island (May 1963). Achieved rank of Lieutenant.

4. Served 4 years in the Division of Naval Reactors Atomic Energy Commission as a Junior Reactor Engineer (1963 - 1967). My responsibility was for the nuclear reactor portion of the Shippingport pressurized water reactor.

5. Certificate of successful completion Bettis Nuclear Engineering School of the Atomic Energy Commission's Bettis Atlantic Power Laboratory, West Mifflin, Pennsylvania (1965).

6. Reactor Plant Training (one month) Navy's D1G Prototype Reactor Plant, Atomic Energy Commission's Knolls Atlantic Power Laboratory (1966).

7. Associate Engineer with Reactor Engineering duties at the Big Rock Plant Nuclear Power Station (boiling water reactor) for the Consumers Power Company. Charlevoix, Michigan. (one-half year 1967).

8. Offered position with the Liquid Metal Fast Breeder Reactor Program Planning Office of the Argon National Laboratory in January 1968; but declined in order to pursue Ph D.

9. Post-doctoral studies:
Nuclear Reactor Hazards Analysis and Constitutional Law, Department of Advanced Studies in Science Technology and Public Policy. Indiana University, Bloomington, Indiana. (1972 - 1974).

10. Research on nuclear reactor hazards and preparation of a treatise on the subject of nuclear reactor accident hazards and constitutional law, Department of Physics and Environmental Science. University of Massachusetts. Amherst, Massachusetts. (October 1974 - October 1976).

11. Presently self-employed Nuclear Hazards Analyst.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	:	
PUBLIC SERVICE ELECTRIC & GAS CO. (Salem Generating Station Unit #1)	:	DOCKET NO. 50-272
	:	
	:	

CERTIFICATE OF SERVICE

I hereby certify that copies of Testimony of Richard E. Webb, in respect to Board Question #3 of Order dated April 18, 1979

in the above captioned matter have been served upon the attached list by deposit in the United States mail at the post office in Northfield, N.J., with proper postage thereon, this 25th day of April, 1979.

CARL VALORE, JR., Special Nuclear Counsel for the Intervenor, the Township of Lower Alloways Creek

Dated: April 25, 1979

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