April 20, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of HOUSTON LIGHTING & POWER COMPANY (Allens Creek Nuclear Generating Station, Unit 1)	Docket No. 50-466

INTERVENOR DOHERTY TESTIMONY

OF JAMES M. SCOTT

ON DOHERTY CONTENTION #3

INADEQUACY OF THE DESIGN SAFETY LIMIT

OF APPLICANT'S FUEL RODS

Q: What is the purpose of your testimony?

A: The purpose of my testimony is to show that, as alleged in Doherty Contention #3, the design proposed for the fuel system for the ACNGs will prsent an undue risk to the health and safety of the public and Mr. Doherty. Doherty Contention #3 states:

> The design safety limit of thermal energy for each fuel rod is too high for fuel rods which will be in a cluster such as that proposed for ACNGS. Tests on two General Electric 9/16 in. outside diameter zircaloy rods, that had been irradiated approximately one third (1/3) of the time a fuel rod typically is irradiated indicate that the cladding will rupture at between 147 calories/ gram of uranium oxide fuel to 175 calories/ gram of uranium oxide fuel if a power excursion (Reactivity Initiated Accident) occurs. The rupture of the rods means there is danger of:
>
> a. Fuel fragments escaping into the coolant from the rods due to the pressure of gases escaping through the rupture,
>
> b. Pressure pulses from fuel in contact with the water after it escapes from the fuel rod,

c. Serious weakening of the cladding strength of the rods after rupture. Such rupturing means the rods would not have sufficient resistance to withstand normal pressure actions of the circulating coolant or disturbances due to the power excursion (RIA) itself. The weakened rods would be bent out of allignment resulting in excess fissioning leading to further excess reactivity at the location where the rods are distorted and ruptured cladding, with results such as (a) and (b).

d. Fuel from damaged rods may prevent function of the control rods by jamming interstices between the control rods and the reactor bottom.

Because ACNGs has more compact rods in each fuel bundle and a higher power core density than any operating BWR in the United States, petitioner requests the design safety limit of the fuel rods be lowered, and various parameters in the reactivity control system be altered in accordance with this change.

Q: Do you agree with this contention?

A: Yes, except for the statement in the first part that the fuel rods had been irradiated approximately one third (1/3) of the time a fuel rod typically is irradiated. My testimony will show the irradiation approximately one tenth (1/10) to one fifteenth (1/15) of the <u>amount</u> a fuel rod is typically irradiated.

Q: Have you attached your personal qualifications?

A: Yes. These are Attachment F to this filing.

Q: Is there any source in the literature of nuclear science that poses the contention?

A: Not exactly. However, the diagram called Attachment A here essentially contains the contention. The diagram was taken from the publication ORNL-NSIC-23, "Potential Metal-Water Reactions in Light-Water-Cooled Power Reactors", August, 1968, by H. McLain. That publication took the Figure from an article, "A Review of Generalized Reactivity Accident for Water-Cooled

and -Moderated UO_2-Fueled Power Reactors, Nuclear Safety, 8 (2): 116-127 (Winter 1966-1967), by G. O. Bright.

Q: What is the "design safety limit" for fuel rods for the ACNGS?

A: The "design safety limit" for fuel rods for ACNGS and all BWR and PWR reactors for production of electricity is set at 280 cal/gm. of UO_2 fuel as set in Regulatory Guide 1.77 of the NRC. This limit was arrived at largely through review of the Special Power Excursion Reactor Test (SPERT) reactor tests of 1968 through 1970. The requirement is that in the worst case, "Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/gm. UO_2 at any axial location in any fuel rod." (Regulatory Guide 1.77) Recently it has been recommended that a radial average peak fuel enthalpy below approximately 240 cal/gm. would assure, "[N]either severe fuel-rod damage nor loss of geometry would occur", (MacDonald, et al, "Assessment of Light-Water-Reactor Fuel Damage During a Reactivity-Initiated Accident", Nuclear Safety, 21(5) 582-602, Oct. 1980, p. 602) and that this is the appropriate design safety limit. That "The SPERT test rods, which were subjected to a total energy deposition of 280 cal/gm. UO_2 actually reached a peak radial average fuel enthalpy of approximately 230 cal/gm. UO_2.",(Ibid. p. 583), explains this change in thinking about this limit for the researchers who authored the Nuclear Safety article. The 280 cal/gm. limit is currently in use as the limit to the control rod drop accident under any conditions in the operation of Quad Cities Station, Unit 1, an 800 MWe BWR, (Amendment 61 to DPR-29, December 5, 1980, Operating License for Quad Cities Unit 1) and Vermont Yankee, a 514 MWe BWR, (Amendment 55 to DPR-28, Oct. 26, 1979, Operating License for Vermont Yankee Nuclear Power Station), however.

NUREG 75/087, the Standard Review Plan, ("Fuel System Design", Sec. 4.2, p. 4.2-4) states that, "... [o]bserving the

limit... prevents widespread fragmentation and dispersal of the fuel and avoids generating pulses in the primary system during an RIA." The limit thus defines the maximum heat energy that can be safely inserted in fuel rods when the power level is changed accidently. This is to assure the fuel rods do not rupture and disperse fuel.

Reactivity has been defined as the "... percentage excess of neutrons per fission that will cause fissions in the next generation" (Webb, R. E. Accident Hazards of Nuclear Power Plants, 1976, p. 13) or, "A measure of the departure of a nuclear reactor from critical (i.e. from a state where a self-sustaining chain reaction can take place), such that positive values correspond to reactors above critical." (Sarbacher, R. A Dictionary of Electronics and Nuclear Eng. 1957

Q: What are the tests described in the contention?

A: These were tests performed at the National Reactor Test Center in 1969-1970 in the SPERT series, and reported in a limited distribution publication, (IN-ITR-113, "The Effects of Burnup on Fuel Failure, Part 1 - Power Burst Tests on Low Burnup UO_2 Fuel Rods," 1970, p. 20). In test 568, a fuel rod 5/16 in. outside diameter and 7% U_{235} enrichment (called a GEX rod) ruptured at 147 cal/gm. causing a pressure pulse of 165 p.s.i. Whether this 147 cal/gm was a measure of radial average peak fuel enthalpy or radial average total energy deposition is not reported. The rod had been irradiated to 3,480 mega-watt days/metric tonne of UO_2 fuel (MWd/t).

Burnup of fuel reaching 45,000 MWd/t is now being attempted at the Monticello plant, a 536 MWe BWR (Amendment-42 to DPR-22, Monticello Nuclear Generating Plant, Table 3.11.1, Dec. 28, 1979) having passed a calculated 40,000 MWd/t. Burnup greater than or equal to 45,000 MWd/t is planned for the ACNGS. (See Staff Reply to Question 12-20-10, Doherty Interrogatory Set #12, Feb. 19, 1980).

The contention is incorrect in its representation of the outside diameter of the rod. However, in all other respects

but this and enrichment, the rod in test 568 was a typical General Electric BWR fuel rod. In the final analyis of this test, the internal pressure in the fuel rod, and not melt of fuel was considered the cause of the rod rupture. (IN-ITR-113, p. 49)

Q: What were the results of other tests conducted in the SPERT series?

A: These were tests similar to the SPERT tests in the contention, but they used fuel rods with burnups of 13,000 and 32,000 MWd/t. At 7% enrichment, there were four tests performed, although the original plan was for 12 test excursions. (IN-ITR-118, "The Effects of Burnup on Fuel Failure, II. Power Burst Tests on Fuel Rods with 13,000 and 32,000 MWd/MTU Burnup", R. W. Miller, Dec. 1970, p. 1). One of the two rods irradiated to 32,000 MWd/t,"had three fractures which together extended over most of the active length of the rod, and one ruptured blister."(Ibid. p. 16) This happened at 85 cal/gm. The second rod failed at 176 cal/gm, which disagrees with the result for lower burnup fuel given in IN-ITR-113. These too were "GEX" rods, differing only in outside diamter and enrichment from typical General Electric BWR fuel rods. In the 85 cal/g test there was no evidence the rod was waterlogged, because the measured pressure pulse from the transient was small compared to what would be expected from a waterlogged fuel rod. (Ibid. p. 25) The energy deposition data was reported later to have an absolute uncertainty of ± 12% and relative precision between measurements was stated then to be about ± 2%. (Report RE-S-76-187 (RAM 585-76) "Light-Water Reactor Fuel Behavior Program Description: RIA Fuel Behavior Experiment Requirements", L. B. Thompson, D. L. Hagerman, P. E. MacDonald, Oct. 1976.

Other tests in the series attempted to determine the effect

of excursioning several rods together in an assembly such as in an actual reactor. The failure threshold for unirradiated fuel rods as measured at SPERT was 225 cal/gm. But, groups of five unirradiated rods in a cannister (which is similar to a BWR fuel channel box) showed a 5% to 10% decrease in threshold for fuel destruction under these conditions, that is 240 cal/gm to 225 cal/gm. (IN-ITR-116, "The Response of Fuel Rod Clusters to Power Bursts", L. J. Siefken, May, 1970, p. 42). The rods used were "SPXM" rods four sixteenth of inch outside diameter, with enrichment of 7% to 10.5%. The standard General Electric fuel assembly consists of an enshrouded 62 fuel rod arrangement. The use of more rods per assembly may result in rod destruction at lower reactivity insertion levels. The 5 rod assemblies used in these SPERT tests had the same ratio of rod center to center distance and rod outside diameter, 1.3, (Ibid. p. 1) as those described in the GESSAR (Nureg-0152, Table 4-1, p. 4-3, March 1977) fuel assemblies.

Because there are errors in two publications, NUREG/CR-0269 where the fact of fuel rod failure (the publication's term) for the 85 cal/g test is omitted from Table C-IV, at p. 144, and in Nuclear Safety, 21(5) Oct. 1980, p. 582-602, "Assessment of Light-Water-Reactor Fuel Damage During a Reactivity-Initiated Accident" P. E. MacDonald, et al, at p. 585, Fig. 2, four pages of IN-ITR-118 are presented as attachment B.

Q: You mentioned the enrichment as higher than operating BWR fuel enrichment. What difference does enrichment cause in the results of these tests that might make their applicantion to ACNGS erroneous?

A: It is reported that initial fuel rod failure threshold decreased with increased enrichment. Thresholds were determined to be: 265 to 277 cal/g UO_2 for 5% enriched fuel, 254 to 264 cal/g UO_2 for 10% enriched fuel, and 232 to 246 cal/g UO_2 for 20% enriched fuel. (NUREG/CR-0269, "Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments", Fujishiro, T., Johnson, R. L., MacDonald, P. E. and McCardell, R. K., Aug.1978, p. 27.

Q: How might reactivity be inserted in an amount exceeding the thresholds discovered in these earlier tests, in the ACNGS?

A: The reactivity excursion test results, as applied to Regulatory Guide 1.77, have to viewed critically. The design safety limit was 280 cal/gm, but this gives full credit for heat energy transfer to the fuel and does not include energy from the delayed neutrons. (Nuclear Safety 21(5) 583-603, Oct. 1980, p. 583 (Hereafter: Nuclear Safety Article). The design safety limit for BWR's under zero and low power conditions is 170 cal/gm "radial average energy density" (NUREG 75/087, "Standard Review Plan," Sec. 4.2, p. 4.2-4) and this document does not indicate the two factors which make the 280 cal/gm limit too great (heat transfer to coolant, and failure to include energy from the delayed neutrons) are factored into the term "radial average energy density".

In Table 6-4 of NEDO-20,626, "Studies of BWR designs for mitigation of ATWS," Oct. 1974", the peak enthalpy for the main steam isolation valve closure ATWS is 150 cal/gm, which when considered a low power event and the failure to account for delayed neutrons means a rather small margin if not actually exceeding the low power criteria of the Standard Review Plan.

An NRC sponsored research group considers the most severe Reactivity Insertion Accident to be a rod drop accident during start up. ("Physics Analysis of Power Burst Facility Test RIA 1-3 (PNL Hardware)", by E. H. White, Interim Report, December, 1978.

In the Nuclear Safety Article, p. 601, MacDonald et al. suggest that a limit of 230 cal/gm should have been chosen instead of 280 cal/gm. It follows then the 170 cal/gm limit should be proportionately reduced as well. On page 15.1-166a of the Supp. No. 3 to the Montague PSAR, a beginning of life core enduring a rod drop of 5 ft/sec, with a worth of 0.01435 (the most severe conditions) yields a calculated peak enthalpy of 221.96 cal/ gm. This squares with results in Chapter 4,

p. 25, of NEDO 10,527 "Rod Drop Accident for Large BWRs", where a 0.0123 worth rod dropped at a rate of 5 ft/sec inserts 245 cal/gm. In this same report, at 3.5 gigawatt days/tonne core exposure a 0.0129 worth rod dropped at 5 ft/sec inserted 239 cal/gm. (Ibid.)

NEDO 21,231 "Banked Position Rod Withdrawal Sequence", Jan. 1977, p.2-1/2-2, lists a worst case for the highest rod worth (0.012 rod worth) would produce a "peak fuel enthalpy of 232 cal/gm in a design based control rod drop accident." The Staff in its review of ACNGS (Safety Evaluation Report Supp No 2, p. 15-4) applied data from NEDO 10,527 Supp No. 1 which showed, "...[a] value for the inserted reactivity worth which would produce a resultant peak fuel enthalpy greater than 280 calories per gram" which is "...greater than 0.013 $\Delta k/k$." Hence it may be inferred that reactivity insertions such as of 0.012 rod worth may well exceed the point at which fuel damage other than loss of hermeticity occurs because the reactivity insertion will be close to the 280 cal/gm, which MacDonald et al. (Nuclear Safety Article suggest should be lowered.

And, according to the Staff (Reply to Doherty's 14th Set of Interrogatories, question 9, June 9, 1980) if the most adverse combination of by-passed rods is assumed, 232 cal/gm is the energy yield of a rod drop accident.

Q: What are the experimental results of the recent tests at the Power Burst Facility and other recent tests?

A: Most recently, NRC sponsored research has returned to the Reactivity Insertion Accident at the Power Burst Facility in Idaho. Unfortunately there have been no excurions of fuel rods irradiated more than 5,000 MWd/t, so the SPERT tests at 13,000 MWd/t and 32,000 MWD/t, few as they are remain neither confirmed or denied. According to Odekirk (ANCR-1095 "Detailed Test Plan Report for Power Burst Facility: The Behavior of Unirradiated PWR Fuel Rods under PCMA Conditions", 4/74) the Power Burst Facility (PBF) tests will be using fuel rods that differ from ACNGS rods in fuel enrichment and neutron spectrum during irradiation prior to excursion testing. Fuel enrichment causes differences as

shown above, which are not great.

Neutron spectrum is a different matter. In the SPERT tests, "...[a]typical of BWRs was the slightly lower fast-to-thermal flux ratio (between 0.15 and 0.20) which resulted in a slightly less than typical fast neutron dose to cladding" during pre-irradiation prior to excursion testing.(NUREG/CR-0269, "Light Water Reactor Fuel Response During Reactivity Initiated Experiments", Fujishiro, T., MacDonald, P. E., Johnson, R. L., and McCardell, R., August, 1978, p. 35) Taking this fact, one critic in arguing the current NRC design safety limit inadequate stated, "Fast neutrons damage the cladding by making it brittle, which suggest even more strongly the possibility of crumbling. Hence the cladding was not as damaged from this standpoint as it would be in an actual commercial reactor." (Accident Hazards of Nuclear Power Plants, Webb, R. E., 1976, p. 54)

The recent test reports I have seen do not indicate if irradiation in the Saxton reactor irradiates with a neutron spectrum more like one fuel rods in the ACNGS will obtain. However, the quote above by R. E. Webb has possibly become prediction, because data from the PBF, indicate, "The consequences of failure at Boiling Water Reactor hot-startup system conditions appear to be more severe than previously observed in either the stagnant capsule SPERT or NSRR tests. Metallographic examination of both previously unirradiated and irradiated PBF fuel rod cross sections revealed extensive variation in clad wall thickness (involving considerable plastic flow) and fuel shattering along grain boundaries in both restructured and unrestructured fuel regions. Oxidation of the cladding resulted in fracture at the location of cladding thinning and disintegration of the rods during quench." (Light Water Reactor Fuel Response During Reactivity Initiated Experiments", MacDonald, P. E., McCardell, R. K., Martinson, Z. R. and Seiffert, S.L., from the International Colloquium on Irradiation Tests for Reactor Safety Programs, June 25-28, 1979, Petten, The Netherlands, CONF-790646-08)

These same authors state that "The results of these test (RIA tests at PBF) indicate that, whereas the failure threshold for unirradiated and irradiated fuel rods of ~225 and ~140 cal/g UO_2 (radial average peak fuel enthalpy) are generally consistent with previous SPERT and NSRR results, the consequences of fuel rod failure at BWR hot startup conditions are somewhat more severe than those observed in either SPERT or NSRR. (Nuclear Safety Article, p. 591)

In test RIA 1-2, four fuel rods were subjected to 5,000 MWd/t burnup in a cluster arrangement and then were subjected to a power burst of 180 cal/gm. A single rod that had not been opened prior to excursioning failed at 135 cal/g UO_2 fuel and had what appeared stress-corrosion cracks. (Overview of Recent Power Burst Facility (PBF) Test Results", Paper Presented at the Seventh Water Reactor Safety Research Information Meeting, Nov. 5-9. 1979, By J. J. Zeile, Idaho National Engineering Laboratory)

Test RIA 1-1, showed that the current 230 cal/gm design safety limit is inadequate. In that test, radial average peak fuel enthalpy of 285 cal/gm. produced complete flow blockage in an encapsulated core within four seconds of the power burst. (Nuclear Safety Article, p. 594) A 250% expansion of fuel volume was required to block coolant in the PBF capsule. (Ibid.) See Attachment C for summary of the PBF tests.*

Q: Have the results of any of these tests been interpreted to require that the design safety limit be reduced?

A: In the Nuclear Safety Article mentioned previously, MacDonald, et al., finished by saying, "... [n]either severe fuel rod damage nor loss of normal geometry is expected at radial average peak fuel enthalpy below 240 cal/g. Therefore, we conclude that an RIA in an LWR would pose no real safety concern, but the NRC licensing criteria should be re-evaluated. (Ibid. p. 602) This is backed further by a memorandum from D. A. Hoatson of

* Taken from the Nuclear Safety Article, p. 592.

the Fuel Behavior Research Branch (NRC) to R. O. Meyer of the Reactor Fuels Section (NRC) titled "RIA Energy Deposition", dated Sept. 17, 1979. It states, "A reassessment of available data is needed to define a threshold in terms of 'fuel enthalpy'" because, "If a vendor calculated energy depositions near the threshold using the 'fuel enthalpy' definition the 'total energy' deposition could exceed 300 cal/gm." Mr. Hoatson further states, "It is our understanding CPB (Core Performance Branch) plans to take the necessary actions to resolve the inconsistency between the Regulatory Guide 1.77 position and the SPERT data on which it is based."

Q: Were there any tests between these two periods, and if not were any called for?

A: The answer is no and yes.

First, the various reviews of work on reactivity initiated accidents such as the Nuclear Safety Article, and NUREG/CR-0269 "Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments", Aug. 1978, do not mention any tests between the SPERT tests and those in the Power Burst Facility with the exception of the work at the Nuclear Safety Research Reactor (NSRR) in Japan. That experimental work showed the significance of the pellet-clad gap variable in dtermining the threshold energy deposition for the onset of departure of nucleate boiling. (JAERI-M-8087 "Effects of Initial Gap Width on the Fuel Failure Behavior under RIA conditions", Saito, S., et al. Jan. 1979) But none of these experiments attempted to describe fuel rupture.

By 1970, General Electric had plans to have reactors which produced more than 1,000 MWe, but no experimental basis for determining the physical correctness of its computer codes for calculating the effects of reactivity insertion in large cores. These were called point kinetics programs, but in WASH-1146("Water Reactor Safety Program Plan", prepared by the National Reactor Testing Station, Feb. 1970) the Atomic Energy Commission stated (p. III-93), "In general, application of the point kinetics approach cannot be expected to be re-

liable for large, loosely-coupled reactors." A proposed (Ibid. III-94) SPERT Large-Core Dynamics Program of research was never carried out. Today, with many large core reactors in use nationwide, the Power Burst Facility is doing small core experiments. PTR-738("Review of Generalized Reactivity Accident for Water Cooled and Moderated Uranium Dioxide Reactors", G. O. Bright, 1965, p. 102) stated that a problem area was excursion behavior as it related to core size, and spoke of the need for kinetics experiments, "..[a]dequate to determine the importance of the effects of space dependence under potentially hazardous conditions." So, the need was expressed, but many reactors were built before the research work was begun at the Power Burst Facility.

Q: What evidence indicates fuel will be extruded from the cladding in the event of a Reactivity Insertion Accident?

A: The sequence figure of a nuclear excursion accident (Attachment A) shows "Fuel Release" following "Thermal Stress Rupture" as an outcome when excess reactivity is available. (ORNL-NSIC-23 "Potential Metal-Water Reactions in Light Water Cooled Power Reactors, Aug. 1968, but originally from Nuclear Safety 8(2): 116-127, in an article titled, "A Review of Generalized Accidents for Water-Cooled and Moderated UO_2 Fueled Power Reactors", by G. O. Bright, Winter 1966-67.)

Attachment C, (Nuclear Safety Article, p. 592) indicates rods receiving 250 cal/g UO_2 and 260 cal/g UO_2 lost 10% and 15% respectively of their fuel. (At p. 598 of the Nuclear Safety Article, it states this loss was 13%.) These were single rod tests. If 10% to 15% of many fuel rods were to extrude this percentage of fuel, then a considerable mass of fuel would be released.

Q: What evidence indicates there will be pressure pulses from fuel contacting coolant under reactivity insertion conditions?

A: If molten fuel leaves the cladding perimeter, it is thought a steam explosion will take place. M. L. Corradini, of the Light Water Reactor Safety Studies Division of Sandia National Laboratories (SAND 80-1535c "A One Dimensional Transient Model for Analyzing Large-Scale Steam Explosion Experiments,"1980.) has divided this phenomena into three stages. "(a) mixing of the molten fuel and water; (b) triggering and spatial propogation of rapid fuel fragmentation through the fuel-coolant mixture; and (c) expansion of the steam against the surroundings." Estimates of the quantity required to produce the consequences to the surroundings include a July 1971 evaluation that but 1% to 3% of core mass melting would cause a steam explosion sufficient to destroy the (reactor) vessel. (BNI-1910 "Core Meltdown Evaluation", Morrison, Appendix C, 1971).

However, it has never been confirmed that a steam explosion has occured in a nuclear reactor. However, the explosion that accompanied the final Boiling Reactor Experiment (BORAX) at the National Reactor Testing Station has never been explained as anything else but a steam explosion. (ANL 5323 "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor," J. R. Dietrich, 1954, p. 39) A rough estimate of 6,000 p.s.i. to 10,000 p.s.i. was made (Ibid. p. 29) of the force of this explosion.

Pressure pulses from the rupture of fuel rods have been shown to occur. In IN-1370 ("Annual Report - SPERT Project, 10/68 - 9/69" R. W. Miller, p. 27, Attachment D) a 2,350 p.s.i.g. pulse was generated on failure at 300 cal/gm of UO_2 recention following depostion of 340 cal/gm of UO_2 to a rod of 3,000 MWd/ton burnup.

The force of pressure pulses from groups of rods in one assembly will bend the channel box outward where it will block movement of the control rod. The BWR/6 design uses

a 0.260 inch thick control rod blade in a 0.482 inch water gap, leaving 0.111 inch tollerance between the control rod blade and any fuel rod channel, so only a small amount of deflection will interfere with blade progress. (Measurements from NEDO 10,566 "Testing of Cruciform Control Rods for BWR/6", 1970, p. 1)

The ACNGS fuel channel wall thickness is 0.120 inches. (ACNGS PSAR, Table 4.3-1, p. 4.3-17, Amendment 56, March 1981) Pressure pulses in addition to damage caused indirectly by channel wall bulging, will propogate to other fuel assemblies through holes between the lower tie plate and channel, between the fuel support and lower tie plate (back pressure), and in the lower tie plate through which it will act on the next adjacent fuel channel. (See Attachment E) Note: As shown on the attachment, the pressure pulse will travel downward from the fuel rods as indicated by the arrow drawn in the figure.

Q: What evidence indicates fuel rod clad will be weakened by rupture, in particular that the affected cladding will not stand up to disturbances due to the reactivity insertion such as pressure pulses from the coolant contact as the contention describes?

A: The test RIA-ST-1, "Burst 2" at the Power Burst Facility (See Attachment C) produced 250 cal/g UO_2 deposition into a fuel rod. In two publications, P. E. MacDonald, R. K. McCardell, Z. R. Martinson, and S. L. Seiffert of the NRC's contractor, EG&G Idaho, Inc., showed that this deposition substantially weakened the fuel rods. I will come to the weakening after giving their description of how the weakening occured. They suggested, "... [d]uring the power burst the fuel expands out against the cladding; film boiling heat transfer is initiated and the cladding temperatures reach values near the melting point; rapid heat transfer to the coolant results in vaporization and a

modest pressure pulse ($\leq$ 2 MPa) which acts on the ductile cladding to deform it into thin and thick regions; the fuel rod cladding remains in film boiling for about 20 seconds during which the cladding reacts with both fuel and coolant (steam) and becomes heavily oxidized; and, finally, rod fracture occurs upon quench." (CONF 790441-3, "LWR Fuel Response During Reactivity Initiated Experiments, P. E. MacDonald, et al., EG&G Idaho, Inc., 1979, p. 4)

The weakening was extensive, such that 35% of the two unirradiated fuel rods, and 65% of the two irradiated fuel rods subjected to 285 cal/g UO_2 during experiment RIA 1-1, "... [c]rumbled into fine powder and larger chunks of fuel and cladding." (Nuclear Safety Article, p. 598) The irradiated rods were of 4,600 MWd/t burnup, well below the projected 30,000 - 45,000 MWd/t for fuel at ACNGS. Cracking was observed in the single, unopened, irradiated to 5,000 MWd/t burnup, fuel rod in test RIA 1-2, which had absorbed but 185 cal/g UO_2 on excursion testing.

The same authors from EG&G Idaho, Inc. state in greater detail (Ibid.) the weakening is characterised by oxidation, a zircaloy and water reaction found as zirconium oxide on the outside surface of the fuel rod, and a uranium oxide-zirconium reaction whose result was found on the inside surface of the cladding. In thin regions of the cladding the reaction fronts met, meaning the cladding was oxidized through. They state further that these oxide compounds are brittle compared to normal zircaloy tubing.

In line with this discussion of departure from nucleate boiling and energy deposition, the NSRR studies in Japan concluded (JAERI-M-8087) that the gap width had significant influence on the onset of departure from nucleate boiling as a result of reactivity insertion. The threshold energy depositions for the onset of departure from nucleate boiling are about 180 cal/gm UO_2 for a fuel rod with initial pellet-clad gap width of 0.195 mm, about 140 cal/g UO_2 for a fuel rod with initial pellet-clad gap width of 0.095 mm, and

about 110 cal/g UO_2 for a fuel rod with initial pellet-clad gap width of 0.050 mm. (JAERI-M-8087 "Effects of Initial Gap Width on the Fuel Failure Behavior under RIA Conditions", Saito, S., et al., Jan. 1979, p. 1)

Q: What evidence indicates there would be an increase in fissioning if fuel rods were bent out of line and that this would result in more ruptured cladding, fuel escaping into the coolant and pressure pulses from fuel-coolant contact?

A: Applicant's fuel rods will be 3.7 mm apart, according to NUREG-0152, Table 4-1, p. 4-3 (Safety Evaluation Report to GESSAR-238", Nuclear Steam Supply Standard Design, 1977). According to R. E. Webb, inward fuel rod bowing increases reactivity. (Accident Hazards of Nuclear Power Plants, 1976, p. 18) In the contention, the increased fissioning would lead to increased reactivity. The increased fissioning would be due to decreased neutron leakage, because this same author (Ibid. p. 17) states, "Moving fuel apart in a disassembly generally decreases the reactivity by enhancing neutron leakage." In NEDO 10,173, "Current State of Knowledge of High Performance BWR Zircaloy-Clad UO_2 Fuel", May, 1970, bowed fuel rods in the early operation of the Dresden-I plant had five rod failures from fuel rod-channel box contact.

Where pressure pulses distort the channel box, and force fuel rods inward (not outward as at Dresden-I) increased fissioning can be expected between the rods which have been pushed more closely together.

When fissioning fuel rods are closer together, the departure from nucleate boiling ratio decreases. Although recent research shows clearance can be reduced to 0.76 mm ("Reduced Clearance Effect on Critical Power in BWR Fuel Bundles", Ikeda et al. Transactions of the American Nuclear Society, 33 (898-9), 1980)if rods are closer than this fission will increase as will departure from nucleate boiling. Rods which had undergone the transient would already be weakened as described in the results of test RIA-1, and the effect of departure from nucleate boiling on them would be more serious. This scenario is part of the "further core damage"

which results from "conversion to mechanical energy" and leads to "Geometry Δk" and "power rise" as shown in Attachment A.

Q: Does that conclude your testimony on Doherty Contention No. 3?

A: I would point out that the hazard of fuel material clogging or obstructing control rod operation exists for many of the events where reactivity is inserted and the control rods do not insert, or are delayed in insertion, as in the Browns Ferry-III event of June 28, 1980, where rods were inserted after a fifteen minute delay. Reactivity insertion following an event is almost instantaneous, but the effects as described in the contention, such as further collapse of fuel rods, and pressure pulses conceivably would not be instantaneous. I have mentioned bulging of the fuel channels as one mechanism of control rod movement inhibition here, but the second mechanism, clogging with fuel bits might be important in an event where the consequences of the reactivity insertion are not all instantaneous.

Q: Thank you for your testimony, Mr. Scott.

ATTACHMENT A

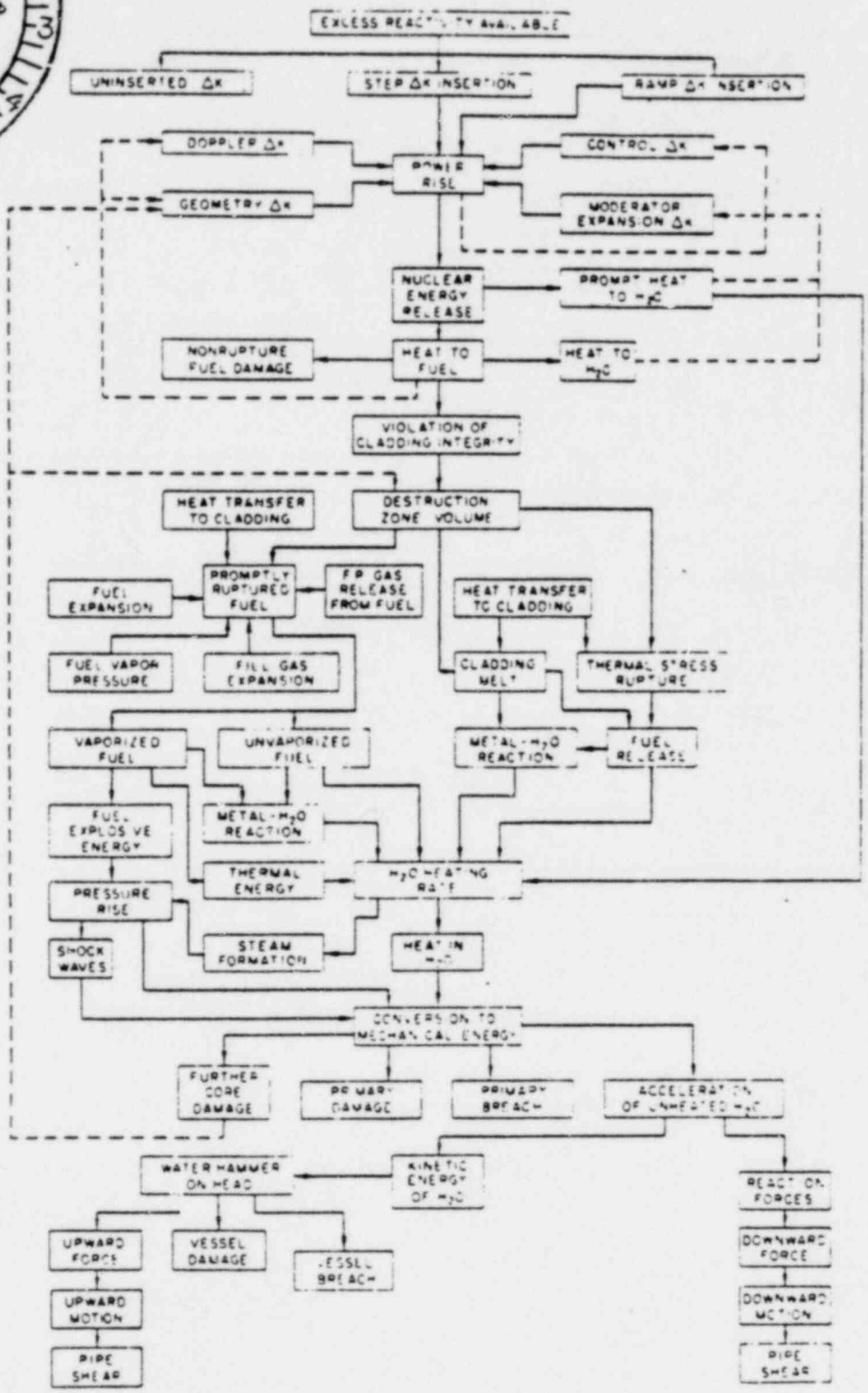

Fig. 3.1. Sequence Diagram of Nuclear-Excursion Accident. (From Ref. 24)

Return to → K. W. HOLTECLAW

ATTACHMENT B

IN-ITR-118
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THE EFFECTS OF BURNUP ON FUEL FAILURE

II. Power Burst Tests on Fuel Rods with 13,000 and 32,000 MWd/MTU Burnup

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FIGURES

In summary, the rod initiated a fracture as a result of this test, but the failure was minor and fission products did not escape.

2.4 Test 859; GEX Rod; 31,800 MWd/MTU; 190 cal/g

2.41 The Fuel Rod. Fuel rod SE-33 received an average burnup of 31,800 MWd/MTU. The rod was under irradiation for 216 days during which its average linear power was 13.9 kW/ft and its maximum linear power was 18.1 kW/ft. A hot-spot factor of 1.04 raised the linear power to 18.8 kW/ft over a short section of the rod. All power and burnup figures are $\pm$ 10%. Fast neutron (> 1 MeV) exposure to the cladding was about 2.7×10^{20} nvt.

A neutrograph of the rod taken after the preirradiation indicated slight centerholing over the entire active length and several visible fractures (typically 3 to 4) per pellet.

Appearance of the rod was identical to the preceding rod (SE-30) and indicated corrosion buildup over the active length.

As in all cases, this rod was carefully inspected before preirradiation and again after preirradiation by magnifying periscope in a hot cell prior to testing. No evidence in either inspection was found to indicate a sub-standard condition of the rod.

2.42 The CDC Test. The fuel rod was placed in a capsule containing room-temperature water under stagnant, atmospheric pressure conditions and subjected to a 3.94 millisecond period transient in the CDC. The maximum energy deposition during the test was 190 cal/g-UO_2. Figures 13, 14, and 15, show the power burst and other recorded data taken during the test.

2.43 Test Results. The rod failed early in the transient when only about 85 cal/g-UO_2 had been deposited. The time of failure is indicated clearly in Figure 13 by the sudden rise of capsule pressure as well as by disturbance of the signals from the cladding and fuel growth transducers.

The rod is shown in Figures 16, 17, 18 and 19. Four independent failures were found: three fractures, which together extend over most of the active length of the rod, and one ruptured blister.

Failure was accompanied by a multi-peak pressure pulse which reached about 70 psig maximum. Sufficient energy was released to cause the water column above the fuel rod to reach a maximum velocity of 5 ft/sec. It was determined that a nuclear-to-mechanical energy conversion of about 0.014% took place. Less than 1% of the active cladding was consumed in a metal-water reaction. Very little fuel was lost through the cracks into the capsule.

Metallurgical examination of the rod was not performed due to termination of the program.

ATTACHMENT C

Table 2 RIA Test Results at the PBF

RIA test	Fuel-rod type	Fuel enrichment, % ^{235}U	Burnup, MWd/t	Radial power peaking factor	Radial average total energy deposition,[a] cal/g UO_2	Radial average peak fuel enthalpy,[b] cal/g UO_2	Peak local fuel enthalpy,[c] cal/g UO_2	Comments
RIA-ST-1 Burst 1	17 x 17 PWR	5.8	0	1.076	250	185	205	Did not fail; first test on ST-1 rod
RIA-ST-1 Burst 2	17 x 17 PWR	5.8	0	1.076	330	250	275	10% of fuel washed out; second test on ST-1 rod
RIA-ST-2	17 x 17 PWR	5.8	0	1.076	345	260	290	15% of fuel washed out
RIA-ST-3	17 x 17 PWR	5.8	0	1.076	300	225	250	Did not fail
RIA-ST-4	15 x 15 PWR	20	0	1.48	695	350[d]	530[d]	Completely destroyed; pressure pulse of 35 MPa measured
RIA 1-1	Two Saxton	5.7	4600	1.13	365	285	330	Complete shroud flow blockage
	Two Saxton	5.8	0	1.077	365	285	315	Severe failure; partial flow blockage
RIA 1-2	Four Saxton	5.7	5000	1.13	240	185	215	One rod failed; three rods did not fail

[a]Five methods were used to measure the test-rod radial average fission energy deposited during each transient.[9] A detailed independent review of the five measurement methods confirmed that none were unreliable. The five measurement methods had estimated uncertainties ranging from ±11 to ±14%. These are conservative estimates of the uncertainties, and, based on previous PBF results (where the average burnup measurement is within 3% of the average thermal–hydraulic power measurements), these results are considered to be accurate to within about ±6%.

[b]The FRAP-T5 computer code[4] was used to determine the axial peak radial average fuel enthalpy from the measured total energy depositions. The fraction of energy generated by delayed neutrons after control-rod scram was calculated using the TWIGL computer code (Configuration Control Number H009971B). TWIGL solves the coupled time and space-dependent neutron diffusion and thermal–hydraulic equations for a reactor in two dimensions.

[c]This value will vary somewhat depending on the node sizes in the analytical models used to convert total energy deposition to peak local fuel enthalpy.

[d]The fuel enthalpy at the time of failure was ~3 ms after the time of peak power.

ATTACHMENT D

TABLE VII

SUMMARY OF TESTS ON GEX FUEL RODS WITH AND WITHOUT BURNUP

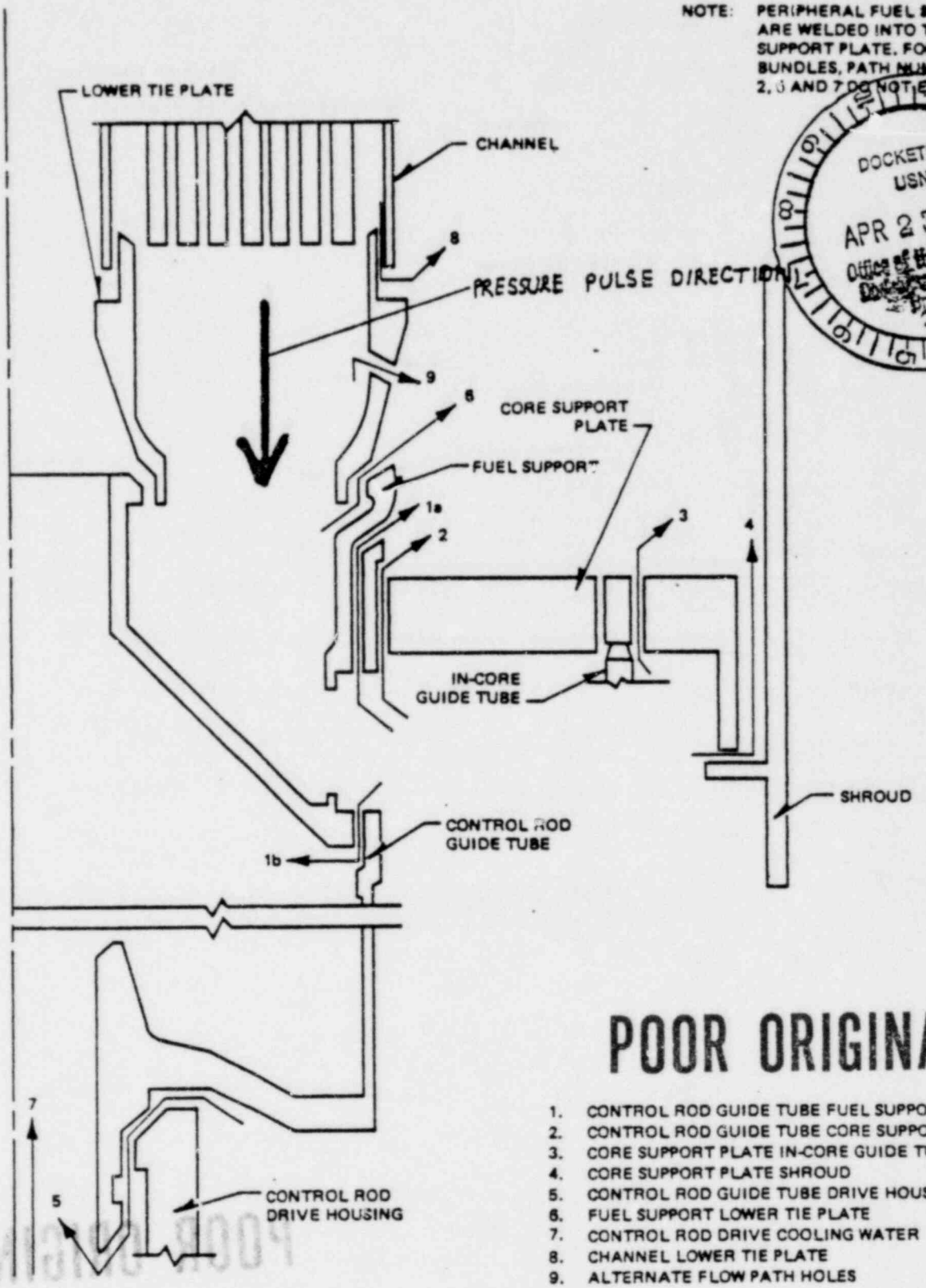
Total Test Rod Energy Deposition (cal/g UO_2)	Burnup (MWd/MTM)	Test No.	Energy at Failure (cal/g UO_2)	Maximum Capsule Pressure (psig)	Nuclear-to-Mechanical Energy Conversion (%)	Metal-Water Reaction (%)
150	0	(No test -- all quantities expected near zero -- no failure expected)				
	3300	571	(no failure)	0	0	0.06
200	0	471	(no failure)	0	0	1.3
	3000	568	147	165	0.03	1.3
260	0	476	260	0	0	8.8
	3300	567	225	350	0.05	4.9
340	0	478	300	375	0.36	19.0
	3000	569	300	2350	0.21	8.5

(1) The relatively low burnup of about 3000 MWd/MTM reduced the failure threshold from greater than 225 cal/g of UO_2 for unirradiated fuel to less than 200 cal/g of UO_2 for the irradiated fuel.

(2) Measurable capsule pressure and generation of mechanical energy occurred at significantly lower energy depositions for the irradiated fuel rods than for comparable unirradiated fuel rods (200 cal/g compared with over 300 cal/g).

(3) The failure mode was apparently affected by irradiation. Little or no cladding melting was observed for any of the irradiated rods, whereas all unirradiated fuel rods showed cladding melting for energy depositions above about 250 cal/g of UO_2. Near the failure threshold, unirradiated rods failed by cladding melting, whereas irradiated rods failed by sudden cladding rupture.

NOTE: PERIPHERAL FUEL SUPPORTS ARE WELDED INTO THE CORE SUPPORT PLATE. FOR THESE BUNDLES, PATH NUMBERS 1 2, 3 AND 7 DO NOT EXIST



1. CONTROL ROD GUIDE TUBE FUEL SUPPORT
2. CONTROL ROD GUIDE TUBE CORE SUPPORT PLATE
3. CORE SUPPORT PLATE IN-CORE GUIDE TUBE
4. CORE SUPPORT PLATE SHROUD
5. CONTROL ROD GUIDE TUBE DRIVE HOUSING
6. FUEL SUPPORT LOWER TIE PLATE
7. CONTROL ROD DRIVE COOLING WATER
8. CHANNEL LOWER TIE PLATE
9. ALTERNATE FLOW PATH HOLES

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

Schematic of Reactor Assembly Showing the Bypass Flow Paths

Figure 4.4-1.

ATTACHMENT F

PERSONAL QUALIFICATIONS OF JAMES MORGAN SCOTT, ESQ.

I received a Bachelor of Science Degree from Arkansas Polytechnical University, Russelville, Ark., in 1963. My major subjects were Physics, Chemistry, and Mathematics, meeting major requirements in the three areas. The school was 13 miles from my home.

In the summer of 1963, I was a full time laboratory assistant in the high energy physics laboratory at Iowa State University, Ames, Iowa, prior to a year as a teaching assistant in Physics. I chose to enter the program at the University of Texas, in the following fall, and worked full-time the prior summer in the nuclear physics accelerator laboratory. I was an Undergraduate Laboratory Teaching Assistant, part time and attended classes part time. In my second and third summers at the University of Texas, I did research in plasma physics, which was the subject of my master's thesis, which was prepared instead of the Ph D. I took the M. S., and a position with Westinghouse, because I was informed I no longer had a student deferrment from military service. Relevant to the Contention are the following undergraduate course: nuclear physics laboratory; and graduate courses: nuclear physics, plasma physics, quantum mechanics, electro-magnetic theory, classical mechanics, statistical mechanics, heat and thermodynamics, and solid state physics.

My post-master's degree work experience includes two and a half years working as an assistant and later associate engineer with Westinghouse doing design, manufacture, and testing of integrated circuits for various

solid state electronic components for several government sponsored projects. This was in Baltimore, Maryland.

From 1969 through 1971, I was employed by General Electic Company in Syracuse, N. Y. designing integrated circuits for G. E. products, such as computer components and radios.

From 1971 to 1974, I was emplyed by Texas Instruments,in Stafford, Texas at their Semi-Conductor Division. There I designed integrated circuits for military communications equiptment. I was later a Product Engineer doing failure analysis of integrated circuits.

From 1975 to 1977, I was a full time student at the University of Houston, Bates College of Law. The majority of my time in law practice has been on hydrology and air emission suits, in addition to the Allens Creek proceedings.

CERTIFICATE OF SERVICE

Copies of "INTERVENOR DOHERTY TESTIMONY OF JAMES M. SCOTT ON DOHERTY CONTENTION #3: INADEQUACY OF THE DESIGN SAFETY LIMIT OF APPLICANT'S FUEL RODS" were submitted to the Parties below via First Class U.S. Postal Service this 20th day of April, 1981 from Houston, Texas.

Sheldon Wolfe, Administrative Judge
Gustave A. Linenberger, Administrative Judge
Dr. E. Leonard Cheatum, Administrative Judge
Richard A. Black, Esq., Staff
Atomic Safety Licensing & Appeal Board
J. Gregory Copeland, Esq.
Jack R. Newman, Esq.
The Several Intervening Parties
Docketing and Service

Respectfully,

John F. Doherty

John Doherty