UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD
Before Administrative Judges:
James P. Gleason, Chairman
Frederick J. Shon
Dr. Oscar H. Paris

In the Matter of	
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2)	Docket Nos. 50-247 SP 50-286 SP
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	April 1, 1983

POWER AUTHORITY'S TESTIMONY OF THEODORE A. MEYER
ON BOARD QUESTION 1.4

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My name is Theodore A. Meyer. I am Manager of the Division of Reactor Vessel Integrity Group of the Nuclear Technology Division of Westinghouse Electric Corporation. A statement of my professional qualifications is attached.

The purpose of this supplemental testimony is to address the current status of Indian Point Unit 3's fuel management program as it relates to Board Question 1.4.

In November, 1982, the Nuclear Regulatory Commission (NRC) issued a report entitled, "NRC Staff Evaluation of Pressurized Thermal Shock." This report discussed the NRC screening criteria of 270°F reference temperature, nil-ductility transition (RT_{NDT}) for longitudinal flaw orientations and 300°F RT_{NDT} for circumferential flaw orientations in the reactor pressure vessel. The screening criteria and the issue of pressurized thermal shock assumes that there are flaws either detected or undetected in the reactor pressure vessel.

During the service life of the reactor vessel, the RT_{NDT} increases above the initial value of RT_{NDT} because of neutron irradiation by an amount delta RT_{NDT} which depends on fluence and materials properties. The initial RT_{NDT} is determined from materials tests made at the time the vessel is fabricated. The change, delta RT_{NDT}, is determined from fluence measurements, calculations, and from trend curves, based on tests of irradiated specimens that measure the effects of neutron irradiation. Analysis of a surveillance capsule which was removed from the Indian Point Unit 3 pressure vessel during this refueling outage supports the calculated fluence levels used in calculating delta RT_{NDT}. There are, however, a number of uncertainties in the estimation of both initial RT_{NDT} and delta RT_{NDT}. Therefore, the NRC Staff has established a prescribed, conservative method for calculating RT_{NDT} which would be compared to the screening criteria. The current total RT_{NDT} values calculated using these conservative methods for the longitudinal and circumferential flaw orientations for Unit 3 are both 218°F because a lower vessel shell plate is curently the most limiting location. Both circumferential and longitudinal flaws are postulated in vessel plates.

One important aspect of this issue of which the Board should be aware is the actual risk posed by pressurized thermal shock. The NRC Staff has calculated and stated in their testimony on this Board question that when the screening criteria are reached a reactor pressure vessel "would have a frequency of crack extension without arrest between 10^{-5} and 10^{-6} per reactor-year." The NRC Staff has also testified "that not all through wall cracks will result in core melt since some crack sizes and crack shapes and crack locations will not preclude ability of the emergency systems to keep the core cooled." Testimony of Dr. Hugh W. Woods and Raymond W. Klecker on Board Question 1.4 at 8.

The Licensees have testified that "[t]he Indian Point Probabilistic Safety Study (IPPSS) evaluated the frequency of a reactor vessel rupture large enough to exceed the capability of the emergency core cooling systems. That evaluation used the same methodology and assumptions as the Reactor Safety Study . . . and yields a mean frequency of 3 x 10^{-7} per reactor year for all types of vessel failure, which included those failures induced by transients (pressurized thermal shock (PTS) chain of events) and spurious events." Licensees' Testimony of Dennis C. Richardson and Dennis C. Bley on Board Question 1.4 at 2. Given this mean frequency, the frequency of core melt resulting from PTS is less than 3 x 10^{-7} per reactor year.

Calculations performed by the Westinghouse Owners Group on Reactor Vessel Integrity using generic fluence values have shown that the Indian Point Unit 3 reactor pressure vessel, before modification to its present reload fuel core, would not reach the NRC screening criteria for approximately 16.5 effective full power years for the most limiting case.

The Power Authority is currently taking measures to reduce the neutron irradiation of the reactor pressure vessel. The current reload fuel core, cycle 4, is a modified low leakage core. This is accomplished by placing spent fuel assemblies at select locations around the periphery of the core. This modified core loading pattern will reduce the peak neutron flux on the limiting vessel shell plate by a factor of 1.4. This modification alone will extend the time by which the pressure vessel will reach the NRC screening criteria to approximately 2005. The expiration of the plant license is 2009.

The Power Authority will be evaluating in the near future other fuel loading patterns which would preclude the Indian Point Unit 3 pressure vessel from ever reaching the NRC screening criteria.

EXPERIENCE: 11 years with Westinghouse and 3 years at Atomic Power Development Associates

WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR TECHNOLOGY DIVISION, STRATEGIC OPERATIONS DIVISION AND PWRSD

1981 - Present: Manager of Reactor Vessel Integrity Programs Group - Responsible for identifying and implementing strucutural analysis required by utilities in the evaluation and resolution of reactor vessel integrity concerns relative to Pressurized Thermal Shock (PTS) and other structural integrity concerns. These responsibilities include the development of methods and the identification and utilization of appropriate technology to evaluate reactor vessel integrity including the identification and evaluation of benefits derived from modifications aimed at improving reactor vessel integrity. These activities include interfacing with the NRC, utilities and numerous other impacted W organizations.

Manage and direct structural integrity engineering analysis efforts performed by members of RVIP and coordination of these efforts with other disciplines and customer/NRC needs.

1975 - 1981: Senior and Principal Engineer responsible for identifying, developing and implementing structural analyses programs and their associated thermal/hydraulic inputs relative to addressing reactor vessel integrity concerns. These programs included evaluations of Large LOCA, Large Steam Line Break and Small LOCA to determine their impact on vessel integrity as well as test programs to develop appropriate boundary conditions (e.g. heat transfer coefficients). Additional major responsibilities included the design, fabrication, testing and operation of capsules for the purpose of irradiating vessel material specimens in test reactors.

1972 - 1975: Engineer responsible for thermal/hydraulic evaluation of reactor internals including evaluation of the reactor vessel for emergency and faulted conditions. Responsibilities included the development of analysis methods, development of required computer programs, as well as evaluation and testing of various reactor internals components. The test program responsibilities included the development of the test program and objectives, design and fabrication of required hardware and test facilities, performance of the required tests and the obtaining of data and reduction of that data into useful engineering evaluations.

ATOMIC POWER DEVELOPMENT ASSOCIATES

1969 - 1972: Co-operative education student engineer and Engineer at Atomic Power Development Associates which was responsible for the design of the Enrico Fermi Breeder Reactor. Responsibilities covered a wide range of thermal/hydraulic and structural analyses, hardware test programs, methods and computer program development activities as well as on-site operational testing associated with the recovery from a major plant accident testing and operation of the plant.

MILITARY:

1967 - 1969: ROTC U.S. Air Force

EDUCATION:

1967 - 1972: B.M.E., University of Detroit in Mechanical Engineering

1971 - 1972: Advanced Degree work in Mechanical Engineering at University of Detroit

1975 - 1979: Masters Degree in Engineering Management (MSIE) at University of Pittsburgh

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CERTIFICATE OF SERVICE

I hereby certify that on the 1st day of April, 1983, I caused a copy of Power Authority's Testimony of Theodore A. Meyer on Board Question 1.4, to be served by United States Express Mail Service to Jeffrey Blum, Esquire, and by first class mail, postage prepaid on all others:

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