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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE POWER COMPANY, et al.	)	Docket Nos. 50-413
	)	50-414
(Catawba Nuclear Station,	)	
Units 1 and 2)	)	

TESTIMONY OF THOMAS R. MAGER AND
THEODORE A. MEYER REGARDING CESG
AND PALMETTO ALLIANCE CONTENTION 18/44

Q. Please state your names.

A. Thomas R. Mager, Theodore A. Meyer.

Q. Mr. Mager, by whom are you employed?

A. Westinghouse Electric Corporation, Water Reactor Division, Post Office Box 355, Pittsburgh, Pennsylvania, 15230.

Q. Please describe the nature of your employment.

A. I am Manager, Metallurgical and NDE Analysis Group, Nuclear Technology Division. In this position, I am responsible for the non-destructive examination and materials support technology relating to design, fabrication, construction, licensing and operation of pressurized water reactor plants, exclusive of fuel. A statement of my professional qualifications is attached to this testimony as Attachment A.

Q. Mr. Meyer, by whom are you employed?

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A. Westinghouse Electric Corporation, Water Reactor Division, Post Office Box 355, Pittsburgh, Pennsylvania, 15230.

Q. Please describe the nature of your employment.

A. I am Manager, Reactor Vessel Integrity Programs Group, Nuclear Technology Division. In this position, I am responsible for identifying and performing structural analyses required by utilities in the evaluation of concerns relating to reactor vessel integrity. A statement of my professional qualifications is attached to this testimony as Attachment B.

Q. Are you familiar with CESG/Palmetto Alliance Contention 18/44?

A. Yes.

Q. The contention states that because of radiation effects on the reactor vessel, the reference temperature, RT_{NDT}, will exhibit a much more rapid increase than has been calculated. Do you agree?

A. No. The phenomena of radiation effects on reactor vessel materials has been extensively studied and is well understood. Correlation of these effects on the material properties of reactor vessel materials has been clearly established. With regard to the Catawba reactor vessels, based on these studies, which include extensive use of and verification by substantial

experimental data, in our professional opinion, the reference temperature (RT_{NDT}) will not increase more rapidly than calculated.

Q. What do you mean by radiation effects?

A. By radiation effects, we mean the change in material properties caused by radiation. It should be noted that neutron radiation is the only component of the total radiation spectrum that has a significant effect on pertinent material properties of the reactor vessels at issue here. While the entire reactor vessel is subject to neutron radiation, the beltline region of the reactor vessel is subjected to the highest amount of neutron radiation or neutron fluence, and thus, is the primary region of concern as it relates to radiation effects on the reactor vessel. The effect of this neutron fluence is a predictable change in the reference temperature, RT_{NDT}, of the reactor vessel material.

Q. Please explain the terms neutron fluence, beltline region, and reference temperature.

A. Neutron fluence is the number of neutrons in a given area over a given period of time. Mathematically, it is the quotient of dN divided by da, where dN is the incremental number of neutrons that enter an incremental cross-sectional area da in a given time period.

The beltline region refers to the vessel shell material including welds and heat affected zone material that directly surrounds the effective height of the core and adjacent regions and that is predicted to experience sufficient neutron fluence to be considered in selecting the limiting reactor vessel material. The heat affected zone refers to the interface of the shell material and the weld metal.

The reference temperature, RT_{NDT}, is the reference nil-ductility transition temperature used to index the reference stress intensity factor K_{IR} to the temperature scale in Appendix G to Section III of the ASME Code. This information is used in developing heat up and cooldown pressure-temperature curves to address the normal, upset and test operating conditions. Technically, the reference temperature, RT_{NDT}, is defined as the greater of the drop weight nil-ductility transition temperature or the temperature $60^{\circ}F$ less than the 50 ft-lb and 35 mils lateral expansion temperature as determined from Charpy specimens.

In this regard, all reactor vessels have an initial RT_{NDT} at the start of reactor vessel life. As the vessel is exposed to a neutron fluence over the years of reactor life, this initial RT_{NDT} value increases or shifts upward.

Q. In its discussion of this Contention, CESG appears to use the terms nil-ductility temperature and nil-ductility transition temperature interchangeably with RT_{NDT}. What do these two terms mean?

A. The nil-ductility temperature (NDT) and the nil-ductility transition temperature (NDTT) are one and the same. In essence, they are the temperature at which a given material will exhibit a marked change in fracture behavior. Above the nil-ductility transition temperature a specimen or structure will sustain a specific amount of deformation without cracking or instability.

Q. Is there a relationship between RT_{NDT} and NDT or NDTT?

A. Yes. They are one and the same when the RT_{NDT} is governed by the drop weight nil-ductility transition temperature. However, RT_{NDT} can be higher than the nil-ductility transition temperature when governed by the Charpy impact tests at NDT plus 60°F. RT_{NDT} can never be less than the nil-ductility transition temperature.

Q. In order to determine the change in the reactor vessel material reference temperature due to neutron fluence, what information must be known?

A. In order to accurately estimate the change in RT_{NDT}, the reactor vessel material reference temperature, two pieces of information must be known. These are (1) the calculated neutron fluence at the location of interest, and (2) the reactor vessel material composition. With this information, a trend curve can be used which plots for different reactor vessel material compositions the change in reference temperature as a function of the calculated neutron fluence to which the reactor vessel will be exposed. To determine the total reference temperature, this change in reference temperature is added to the initial reference temperature of the vessel material.

Q. What trend curves were used for your analysis of the Catawba reactor vessels?

A. Westinghouse trend curves were used in the analysis of the Catawba reactor vessels. These curves are attached to this testimony as Attachment C.

Q. How did you develop these trend curves?

A. We plotted the results of literally hundreds of tests involving surveillance capsule specimens from other Westinghouse reactors relative to the weight percent of copper in the vessel material, neutron fluence, and the resulting shift in the reference temperature. We bounded the results of these tests with curves for different copper levels in the material and derived

equations giving their mathematical description. From these curves and equations, a full set of trend curves were developed.

Q. Why did you focus on copper?

A. It is a well known fact that with regard to material composition, the presence of copper is the dominant factor regarding radiation effects on reactor vessel materials, the greater the copper content, the greater the effect. Experimental data from Westinghouse, as well as other NSSS vendors, national laboratories, and universities demonstrate this point. It should be noted that phosphorous and nickel also have an effect on the radiation sensitivity of reactor vessel materials. However, the effect of phosphorous is generally masked by the presence of copper, and nickel does not become important until copper content exceeds approximately 0.20 weight percent. For the material in the Catawba vessels, where the limiting reactor vessel material of units 1 and 2 contain 0.08 weight per cent copper and 0.09 weight per cent copper, respectively, the effects of phosphorous and nickel are insignificant.

Q. What are surveillance capsule specimens?

A. Surveillance capsule specimens are specimens that have been placed in capsules and inserted into the reactor for a period of time (thus being exposed to measured neutron fluence), have been removed from the reactor, and tested.

Q. Do the Catawba reactor vessels have surveillance programs?

A. Yes, the programs consist of, among other things, the periodic withdrawal and testing of specimens in surveillance capsules. There are six surveillance capsules for each unit. Each surveillance capsule contains sixty Charpy V-notch specimens, nine tensile specimens and twelve 1/2T-CT specimens. With specific regard to heat-affected zone material, 15 of the 60 Charpy V-notch specimens are made up of this material, i.e., base material of the same heat (melt) as that used in the pertinent Catawba vessel, welded together by weld material of the exact same heat (melt) as that used in the pertinent Catawba vessel. The surveillance capsule also contains dosimeters (neutron fluence monitors) and thermal monitors.

Q. Why are there six capsules per plant?

A. It is standard Westinghouse practice to use six surveillance capsules per reactor vessel. However, ASTM E-185, endorsed by Appendix H to 10 C.F.R. Part 50, would only require Catawba units 1 and 2 to each have three capsules.

Q. You stated that RT_{NDT} was affected by neutron fluence. What is the estimated maximum fluence at end-of-life for the Catawba reactor vessels beltline regions?

A. The neutron fluence values for the Catawba reactor vessels, as well as for any reactor vessel, vary due to attentuation through the vessel walls. Based on our calculations, the end-of-life maximum neutron fluence values for the Catawba reactor vessels are $2.1X10^{19}$ n/cm^2 at the inside surfaces, $1.2X10^{19}$ n/cm^2 at the 1/4 wall thicknesses, and $2.7X10^{18}$ n/cm^2 at the 3/4 wall thicknesses. The end-of-life neutron fluence is determined by considering core physics and the calculated neutron spectrum at and in the vessel wall. Neutron fluence calculations are benchmarked against dosimetry measurements from reactor vessel surveillance capsules.

Q. Using the neutron fluence values just described and the appropriate material compositions of the Catawba vessels, how did you determine the end-of-life RT_{NDT} values for the Catawba vessels?

A. Using the end-of-life neutron fluence values, we entered the Westinghouse trend curve for reactor vessel material of 0.1 weight percent copper (Attachment C) and determined the appropriate shift or change in RT_{NDT} over the lives of the reactor vessels. (It should be noted that in 1978 when these initial calculations were made, Westinghouse conservatively assumed that any vessel materials less than 0.1 weight percent copper would be treated as having 0.1 weight percent. Since the Catawba vessels contained less than 0.1 weight percent copper, this resulted in higher and more conservative RT_{NDT} values.) These values of shift or change in RT_{NDT} over the lives of the vessels were reported in the Catawba FSAR as 58°F and 94°F for units 1 and 2, respectively. However, the unit 1 value for shift in RT_{NDT} reported in the FSAR is incorrect. The correct value is 94°F.

To determine the estimated end-of-life RT_{NDT} values, the initial RT_{NDT} values (-8°F and +15°F for units 1 and 2, respectively) were added to these 94°F shifts or changes in RT_{NDT}. Thus, it was determined that the final end-of-life RT_{NDT} values were 86°F and 109°F for units 1 and 2, respectively. The difference in the units 1 and 2 values can be attributed to the difference in initial RT_{NDT} values for the two units,

which were effected by, among other things, product form (i.e., forged versus plate material) and effective quenching time.

Q. When did you initially calculate the RT_{NDT} values for Catawba?

A. The actual calculations were made in approximately 1978. However, the trend curves used were developed in 1976.

Q. Since the time the trend curves were developed have you obtained additional data points from other surveillance capsule programs?

A. Yes. I would estimate that we have tripled the size of our data base from 1976 through present. I might add that much of this new data relates to vessels with lower copper content.

Q. Have you examined this additional information in an effort to determine the accuracy of the initial trend curves and associated estimates of the end-of-life RT_{NDT} values for the Catawba vessels?

A. Yes. The new data confirmed the accuracy and conservative nature of our earlier predictions. Significantly, from the new data we were able to predict with greater confidence the shift in RT_{NDT} for reactor vessels with copper content below 0.1 weight percent, such as those at Catawba. These new predictions were made using the following equation:

$$RT_{NDT} = [420(Cu - 0.05) + 21] \left(\frac{F}{10^{18}}\right)^{.2615}$$

Where Cu is the weight percent copper in the reactor vessel material and F is the neutron fluence experienced by the material.

From this new data, taking into consideration the fact that the limiting copper content of both vessels are 0.08 and 0.09 weight percent, respectively (i.e., both are below 0.1 weight percent), the final estimated end-of-life RT_{NDT} values for units 1 and 2 were calculated to be 66°F and 98.9°F, respectively. In short, the new data showed the conservative nature of the initial calculations which, as previously noted, were 86°F and 109°F for units 1 and 2, respectively.

Q. In your calculations of the estimated RT_{NDT} values for the Catawba vessels you used specific Westinghouse trend curves. Are there other trend curves?

A. Yes, not only did Westinghouse develop a set of trend curves, but also the NRC has provided a set in Regulatory Guide 1.99, and the Guthrie Formula represents a set of trend curves. It should be noted that both the Westinghouse and NRC trend curves represent a bounding of data for the various vessel material compositions which include different copper levels. However, the Guthrie trend curves represent the mean data values and are not bounding, and, more

importantly, in calculating standard deviation, the Guthrie Formula does not take into consideration copper levels in the material. In that copper levels are a dominant factor regarding radiation effects in vessel material, the Guthrie Formula will therefore overestimate RT_{NDT} for vessels with low copper content, such as that contained in the Catawba vessels.

It should be noted that since the Guthrie standard deviation was derived from consideration of the entire set of available data without regard to the significance of copper, there is uncertainty as to the appropriateness of applying the same standard deviation to all copper ranges.

Q. In your initial calculation of RT_{NDT} for Catawba, did you rely on either the NRC or Guthrie Formula trend curves.

A. No. The Guthrie Formula was not available when we initially calculated the RT_{NDT} values for Catawba and, in any event, as previously stated, the Guthrie Formula would have produced results for low copper vessels, such as Catawba, which did not accurately reflect the true end-of-life RT_{NDT}. With regard to the NRC trend curves, since their data base was limited and consisted largely of test reactor samples, rather than

surveillance capsule data, Westinghouse did not consider the basis for these curves to be the most appropriate for Catawba.

Q. Have you had an opportunity to compare the value you calculated with that which would be derived if you used Regulatory Guide 1.99 or the Guthrie Formula?

A. With regard to the Guthrie Formula trend curves, in that the curves are not bounding and do not accurately reflect RT_{NDT} for reactor vessels with low copper content, such as the Catawba vessels, there is no valid basis for comparison. With regard to the NRC trend curves, we did perform a comparison and found the values to be essentially equivalent.

Q. Do the NRC and Guthrie Formula trend curves use current data?

A. The NRC curves set forth in Regulatory Guide 1.99 use a data base composed of tests performed primarily in the mid 1970's. The Guthrie Formula trend curves use available data up to October, 1981.

Q. Would additional, more recent data significantly affect predictions made using the NRC trend curves relative to Catawba?

A. No. If anything, updating the NRC trend curves with more recent data would show that the trend curves are even more conservative than originally thought.

Q. On another matter, does fatigue or defects/flaws in the vessel material or surveillance material have a bearing on RT_{NDT}?

A. No. The determinations of RT_{NDT} is not a function of fatigue or defects/flaws in the reactor vessel material or surveillance capsule material.

Q. How is reactor vessel integrity (as related to RT_{NDT}) addressed for accident conditions?

A. RT_{NDT} is used to evaluate the acceptability of reactor vessel integrity for emergency and faulted (accident) conditions by comparison of plant specific RT_{NDT} values (calculated using a methodology specified by the NRC) against the NRC pressurized thermal shock RT_{NDT} screening criteria. To explain, some transients may lead to a severe cooldown of the reactor vessel coincident with a high pressure in the primary coolant system. This condition is called pressurized thermal shock (PTS), and, assuming other conditions, if RT_{NDT} is not below a prescribed value, could hypothetically lead to a non-ductile condition of the reactor vessel.

To assure that this condition does not exist, the NRC Staff has specified a methodology for conservatively calculating RT_{NDT} values and comparing them to PTS screening criteria. The methodology and screening criteria are reported in the NRC Staff's position paper on pressurized thermal shock (SECY-82-

465). If the calculated RT_{NDT} values are below the screening criteria in SECY-82-465, the Staff states that the reactor vessel is acceptable as it relates to PTS. Significantly, the methodology used to calculate RT_{NDT} for screening criteria comparison purposes is that set forth in the Guthrie Formula, which, as previously stated, overestimates RT_{NDT} for vessels with low copper content, such as at Catawba.

Q. What are the pressurized thermal shock RT_{NDT} screening criteria?

A. The values for the PTS RT_{NDT} screening criteria, as set forth in SECY-82-465, are (1) the maximum acceptable RT_{NDT} value for longitudinally oriented welds and base plates and forgings is $270^{\circ}F$ and (2) the maximum acceptable RT_{NDT} value for circumferentially oriented welds is $300^{\circ}F$.

Q. Have you performed an analysis of the validity of these criteria?

A. At Westinghouse such an analysis was performed.

Q. What are the results of the analysis?

A. The analysis reflected that if the screening criteria are not exceeded, the risk of reactor vessel fracture due to PTS is 6×10^{-6} occurrences per reactor year of operation. Further, upon extrapolation, the analysis reflected that if the RT_{NDT} values conservatively calculated for Catawba using the Guthrie Formula are

not exceeded, the risk of reactor vessel fracture is less than 10^{-8} occurrences per year of reactor operation. Both figures are well below the Commission's safety goal regarding core melt of 10^{-4} occurrences per year of reactor operation. The Westinghouse analysis is in line with the risk analysis figures set forth in and extrapolated from SECY-82-465, and reflects the very conservative nature of these screening criteria as they apply to Catawba.

Q. Using the conservative methodology set forth in SECY-82-465, what RT_{NDT} values were calculated for the Catawba vessels?

A. Based on the conservative methodology set forth in SECY-82-465, RT_{NDT} for the reactor vessels were calculated to be 102.5°F for Catawba unit 1 and 126°F for Catawba unit 2. Thus, the RT_{NDT} values for Catawba units 1 and 2 are predicted to be at least 140°F below the PTS RT_{NDT} screening criteria at end-of-plant life. In view of this large margin of safety, coupled with the conservative calculational methodology required by the Staff to determine the pertinent PTS RT_{NDT} values, the likelihood of a transient resulting in a non-ductile condition in either Catawba reactor vessel is so remote that it is essentially non-existent.

Professional Qualifications

Thomas R. Mager

Structural and Equipment Engineering Department
Nuclear Technology Division
Westinghouse Electric Corporation

Manager, Metallurgical and NDE Analysis Group, Structural and Equipment Engineering Department, NTD. B.S., 1959 and M.S. 1962 in Metallurgical Engineering. In 1966, I attended the Fracture Mechanics Workshop at the University of Denver. In 1967, I attended the Advance Fracture Mechanics course at the University of Denver.

From April 7, 1959, to August 31, 1966, I was employed by the Westinghouse Electric Corporation in Pittsburgh, Pennsylvania, at the Corporation's Research and Development Laboratory. During this time, I was assigned to the Magnetics Department. In my work, I was responsible for developing new and improving existing alloy systems of soft magnetic materials by controlling the chemical composition and/or the metallurgical processing of the material.

From May 1967 to January 1975, I was assigned to the Material Engineering Group of Westinghouse Electric Corporation's Pressurized Water Reactor Systems Division. I was designated as Lead Engineer--Fracture Prevention. I was one of the individuals responsible for reviewing the safety design and operation,

relative to fracture prevention, of the nuclear steam supply systems. I was responsible for developing and applying the "fracture mechanics" or "crack toughness" approach to fast fracture in nuclear steam supply systems. I was the principal investigator and coordinator of the work performed under the Westinghouse-AEC-Euratom program on the effect of irradiation on reactor pressure vessel materials; the Westinghouse-Empire State Atomic Development Associates (ESADA) program on fracture mechanics; and the Westinghouse-AEC Heavy Section Steel Technology (HSST) program on light water reactor vessel integrity.

From January 1975 to September 1976, I was Manager, Materials Engineering, Westinghouse Nuclear Europe. I was responsible for specifying, approving and reviewing materials, fabricational procedures, manufacturing controls and inspection processes for nuclear plant and engineering equipment. I was responsible for planning and directing the Materials Engineering Group.

From September 1976 to November 1977, I was an Advisory Engineer in the Mechanics and Materials Technology Department PWRSD. My responsibilities included providing consultation to assure that W PWR components are designed, fabricated, and operated so as to preclude fracture, providing consultation to customers (utilities) in the area of brittle fracture, radiation effects, fracture mechanics technology, inservice inspection, and local and Federal

regulatory rules; providing a focal point for brittle fracture and fracture mechanics of reactor plant components and for coordinating outside funded technical programs.

Since November 1977, I have been Manager, Metallurgical and NDE Analysis Group. I supervise an engineering group responsible for the nondestructive examination, materials and process support technology relating to design, fabrication, construction, licensing, and operation of PWR plants, exclusive of fuel elements.

I am currently Principal Investigator for the following Electric Power Research Institute/Westinghouse Research Programs:

Feasibility and Methodology for Thermal Annealing of an Embrittled Reactor Vessel.

Steady-State Radiation Embrittlement of Reactor Vessels.

Corrosion Fatigue Characterization of Irradiated RPV Steels.

Development of a Crack Arrest Toughness Data Bank for Irradiated RPV Materials.

Prediction of Environmental Crack Growth in Nuclear Power Plant Components.

Professional Qualifications

Theodore A. Meyer

Manager, Reactor Vessel Integrity Programs Group, Structural and Equipment Engineering Department, NTD. B.M.E., 1972 and advance degree work in Mechanical Engineering, University of Detroit. MSIE in Engineering Management, 1979, University of Pittsburgh.

From 1969 to 1972, I was employed as a 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.

From 1972 to 1975, I was employed by Westinghouse Electric Corporation as an 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. I managed and directed structural integrity engineering analysis efforts performed by members of RVIP and coordination of these efforts with other disciplines and customer/NRC needs.

From 1975 to 1981, I was employed by Westinghouse Electric Corporation as a 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.

From 1981 to the present time, I have been employed by the Westinghouse Electric Corporation as Manager of the Reactor Vessel Integrity Programs Group. In this position, I am responsible for identifying and performing structural analyses 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 Westinghouse organizations.

ATTACHMENT C

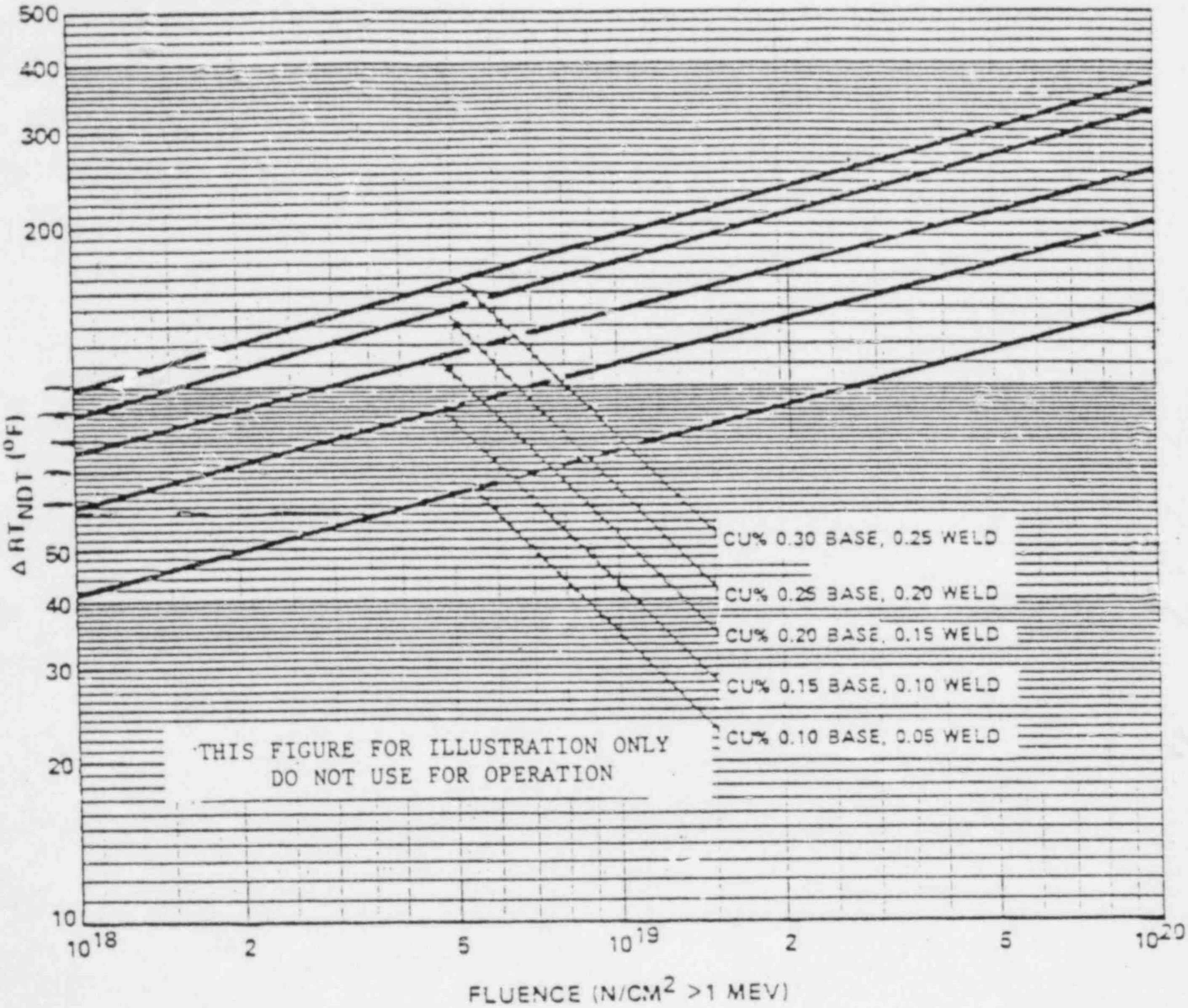

Figure B 3/4.4-2

Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessels Exposed to 550° F Temperature

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(Catawba Nuclear Station,	)	
Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of Applicants' "Testimony of Thomas R. Mager and Theodore A. Meyer Regarding CESG and Palmetto Alliance Contention 18/44" in the above captioned matter have been served upon the following by deposit in the United States mail this 4th day of October, 1983.

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* Designates those hand delivered.