UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
UNITED STATES DEPARTMENT OF ENERGY	)	
PROJECT MANAGEMENT CORPORATION	)	Docket No. 50-537
TENNESSEE VALLEY AUTHORITY	)	
(Clinch River Breeder Reactor Plant)	)	

APPLICANTS' DIRECT TESTIMONY
CONCERNING NRDC
CONTENTION 8

Dated: November 1, 1982

Q.1. Please state your names and affiliations.

A.1. My name is Albert A. Weinstein, I am employed by the S. M. Stoller Corporation as a Manager of Engineering in the New York City office.

My name is Richard K. Disney, I am employed by Westinghouse Electric Corporation, Advanced Reactors Division, Madison, Pennsylvania as Manager for Shielding Analysis.

My name is James F. Murdock, I am employed by the U. S. Department of Energy at the CRBRP Project Office, Oak Ridge, Tennessee, as Reactor Engineer, Reactor and Plant Components Branch.

Q.2. Have you prepared a statements of professional qualifications?

A.2. Yes. Copies are attached to this testimony.

Q.3. What subject matter does this testimony address?

A.3. This testimony addresses NRDC Contention No. 8 which alleges:

The unavoidable adverse environmental effects associated with the decommissioning of the CRBRP have not been adequately analyzed, and the costs (both internalized economic costs and external social costs) associated with the decommissioned CRBR are not adequately assessed in the NEPA benefit-cost balancing of the CRBR.

(a) There is no analysis of decommissioning in the Applicants' Environmental Report;

(b) Environmental Impact Statements (EIS) related to LWRs prepared by NRC have been inadequate due in part to recently discovered omissions (see below), and the FES for the CRBR is no different;

(c) A recent report "Decommissioning Nuclear Reactors" by S. Harwood; K. May; M. Resnikoff; B. Schlenger; and P. Tames, (New York Public Interest Research Group (N.Y. PIRG), unpublished, January, 1976) indicates that (with the exception of the Elk River Reactor) the isolation period following decommissioning of power reactors has been based on the time required for Co-60 to decay to safe levels. Harwood, et al., (p.2) believe the previous analyses are in error because they have underestimated the significance of radionuclide, Ni-59. The time period for Ni-59 to decay to safe levels is estimated by Harwood, et al., (p. 2) for LWR to be at least 1.5 million years. The economic and societal implications of this 1.5 million year decay period are at present unknown.

(d) Petitioner believes that the NRC must systematically analyze all neutron activation products that may be produced in the proposed CRBR to determine the potential isolation period, following decommissioning, and then provide a comprehensive analysis of the costs (both economic and societal) of decommissioning.

Q.4. How have the Applicants assessed the costs and effects of decommissioning CRBRP?

A.4. Contrary to the NRDC contention:

1. The Applicants have estimated the cost of decommissioning the CRBRP based on prior decommissioning experience and on the results of recent detailed studies of the cost of decommissioning large LWR plants;

2. The Applicants have performed analyses of neutron activation products in the CRBRP design process and have extended this analysis to consider the implications for decommissioning the CRBRP.

3. The Applicants have specifically examined the

implications of Ni-59 and other relevant trace element activation products.

Q.5. What information or data are available for use in analyzing the cost of decommissioning of nuclear reactor facilities?

A.5. Considerable experience exists in decommissioning nuclear reactor facilities. Although this experience has involved small reactors, (less than 200 megawatts thermal) the technology developed is directly applicable to the decommissioning of larger reactors.

In addition to this actual experience, the NRC has sponsored comprehensive studies of the methods and costs of decommissioning large LWR power stations which examined the three modes of decommissioning identified in NRC Regulatory Guide 1.86.[1] The three modes are called dismantling (DECON), mothballing (SAFSTOR), and entombment (ENTOMB).

- o DECON is the prompt removal from the site of all materials containing or contaminated with radionuclides at levels greater than permitted for unrestricted use of the property.
- o SAFSTOR is the establishment and maintenance of the reactor power plant in a state of protective storage.

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1 NRC Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors.

o ENTOMB is the encasement and maintenance of the residual radioactive materials in a structure to ensure retention of the radionuclides until they have decayed to levels that permit unrestricted release of the site.

In the cases of SAFSTOR and ENTOMB, possession-only licenses are established and surveillance is maintained to ensure the public health and safety.

Q.6. Has a decommissioning technique been selected for CRBRP?

A.6. At present, the particular technique or combination of techniques that will eventually be utilized in the decommissioning of the CRBRP has not been selected. As the CRBRP approaches the end of its useful lifetime, which is designed to be about 30 years, a decommissioning plan will be prepared and submitted to the NRC. The mode of decommissioning selected will be determined based upon conditions and requirements at that time and could range from total prompt dismantling to mothballing.

Q.7. Is there sufficient data available to estimate the cost of decommissioning CRBRP?

A.7. The history of experience and the availability of recent studies provide a sufficient basis to reasonably estimate the costs associated with decommissioning the CRBRP. In general, taking into account the time value of money, dismantling (DECON) is indicated as the most expensive mode and mothballing (SAFSTOR) the least expensive mode with entombment (ENTOMB) at some intermediate cost.

Dismantling, therefore, is conservatively chosen as the reference mode in estimating the costs of decommissioning since it represents the upper bound cost.

The CRBRP is designed to have a 30-year operating life, including a 5-year demonstration period. The most severe decommissioning conditions from the standpoint of total radioactive contamination would occur upon the completion of this 30 years of operation. Decommissioning the CRBRP at some earlier point in time would not increase the costs or complications associated with decommissioning since they are directly related to the levels of radiation in and around the facility at the time of decommissioning. Thus, it was assumed that decommissioning would occur at the end of the 30-year operating life.

Q.8. Has the technology been developed which would permit successful decommissioning of CRBRP?

A.8. Experience gained in decommissioning eight domestic power reactors, as well as several smaller test reactors, shows that the technology exists to successfully accomplish decommissioning. Comparison of the experience in decommissioning the Fermi 1 reactor and the Hallam reactor (both sodium cooled plants) with the BONUS reactor and the Elk River Reactor (both light water cooled reactors) indicates that there are no significant factors affecting cost and techniques in the decommissioning of sodium cooled plants such as the CRBRP.

The successful dismantlement of the Elk River Reactor through the use of remote handling equipment effectively demonstrates that the technology exists to allow for partial or for total dismantlement of the CRBRP even if inventories of radioactive materials are so high as to preclude hands-on removal.

Q.9. What do you consider the upper bound cost estimate for decommissioning the CRBRP?

A.9. Comparisons with previous nuclear decommissioning efforts indicate that the DECON mode of decommissioning identified in Regulatory Guide 1.86 could eventually be utilized for the CRBRP within reasonable bounds of cost. The highest cost for decommissioning using the DECON method should not exceed about $110 million in 1980 dollars. This upper bound estimate is substantiated by reference to two additional estimates. First, experience with the Elk River Reactor decommissioning provided one basis for estimating the cost of decommissioning CRBRP. Considering that the significant tooling development efforts for the Elk River Reactor dismantlement would not be repeated, the unit costs for removal of the CRBRP reactor vessel and the materials would be comparable to the unit costs for removal of the Elk River vessel and internals. The Elk River costs for this task including tooling development were $1.8 million in 1974 dollars, which would be approximately $2.85 million in 1980 dollars.

The cost of removal of the CRBRP reactor vessel and internals, scaled by weights on a linear basis relative to Elk River is conservatively estimated to be about $33 million in 1980 dollars. Since this part of the work generally represents between 30% to 40% of the total cost of decommissioning, the total cost would be estimated to be about $110 million.

Second, the conservatism of the $110 million estimate also is supported by the results of NRC sponsored studies which estimate the cost of decommissioning large LWR plants to range from about $40 million to $50 million in 1980 dollars. The dimensions and weights of pieces of large LWRs are comparable to the CRBRP (e.g., both plants have approximately 500-600 tons of components in the reactor enclosure and some 100,000 to 150,000 feet of piping). The pumps and heat exchangers are comparable in size and number for a comparable number of loops. Removal and disposal of the CRBRP systems would be no more complicated than the removal and disposal of these systems for an LWR of comparable size.

Q.10. What analyses were performed to identify all neutron activation products which could affect decommissioning?

A.10. In the CRBRP engineering design activities, the Applicants have systematically developed material specifications and applications to limit the introduction of trace elements or isotopes and their neutron activation products. Detailed specifications of the limits of trace elements in

the materials of construction are routinely used in nuclear power plant design to control neutron activation products and material structural properties. Limitations in trace elements (e.g., Co-59) are specified to minimize neutron activated corrosion products and therefore, minimize radiation exposure and assure maintainability of the reactor system.

Based on the specifications for the materials of construction and the CRBRP reactor systems configurations, the Applicants have implemented engineering analysis tools in a normal design analysis procedure to predict the spatial and energy distribution of the neutron flux in the CRBRP and the neutron activation resulting from that distribution for specific CRBRP plant materials. This analysis identified all neutron activation products which could affect normal operation or maintainability of the CRBRP.

For purposes of decommissioning, the analysis described above was extended to include a comprehensive review of recent studies of neutron activation products as related to PWR and BWR decommissioning. This effort was undertaken to determine whether any previously unidentified elements or isotopes could significantly affect decommissioning of CRBRP.

Q.11. What neutron activation products were identified as a result of these anlayses?

A.11. The systematic materials analysis and the review of recent studies of neutron activation products noted above, did not reveal any neutron activation products other than Co-60 and Nb-94 which could significantly affect CRBRP decommissioning.

Q.12. Will Ni-59 have any significant impact on CRBRP decommissioning?

A.12. Ni-59 will have no significant impact on CRBRP decommissioning.

Q.13. In reaching this conclusion, did you consider the NYPIRG Report identified in NRDC's Contention 8?

A.13. In Contention 8, NRDC asserted that the time required for the radionuclide Ni-59, and not Co-60, to decay to safe levels should be the limiting factor utilized in determining the length of the isolation period required for a facility site following decommissioning. NRDC has used results of the NYPIRG report[2] in an attempt to substantiate their contention that neutron activation products have not been properly analyzed in CRBRP decommissioning studies.

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2 "Decommissioning Nuclear Reactors," by S. Harwood, K. May, M. Resnikoff, B. Schlenger, and P. Tames, New York Public Interest Research Group (NYPIRG), unpublished, January 1976.

The principal claim of the report is that Ni-59 presents a radiological problem so severe as to negate past reactor decommissioning experience since the potential effects of Ni-59 had not been considered during these decommissionings. In fact, the radiation effects associated with Ni-59 had been considered in prior decommissionings. Because the quantities of Ni-59 existing at those sites at decommissioning presented no significant hazard to the public health and safety, further explicit discussion of Ni-59 in the decommissioning reports was unnecessary. In fact, there is no significant additional radiological risk, environmental impact or economic cost associated with Ni-59.

Previous decommissioning experience demonstrates that Co-60 is the controlling isotope for the first 100 years after reactor shutdown. However, the NYPIRG Report on which NRDC relies maintains that the isolation period necessary to allow radionuclides at the site to decay to safe levels should be based on Ni-59 and not Co-60 since there would be a significant inventory of Ni-59 present in reactor component materials which would result from neutron activation during the reactor operating lifetime. Based upon the 80,000 year half-life of Ni-59, the NYPIRG Report concluded that for a large power reactor an isolation period of up to 1.5 million years would be required before the Ni-59 present in the reactor

components would decay to the same safe level of radiation that Co-60 would achieve in 120-200 years.

The NYPIRG Report used a simplified model to calculate the Ni-59 radiation dose rate from activated materials in a nuclear reactor. This calculational technique is in error by a factor of 1,000,000 due to the following:

(1) The point source model used by NYPIRG concentrates the Ni-59 activation products at a point with no self absorption in the material in which the parent isotope was activated. In reality, the Ni-59 activation product would be dispersed in the material, thus resulting in lower dose rates.

(2) The NYPIRG analysis selected a single average energy at which the Ni-59 decay gamma energy due to K-electron capture is released. In reality a continous spectrum of gamma rays emanates from each decay event. Selection of a single average energy rather than a spectrum results in underestimation of self-absorption of low energy gamma rays and thus an overestimation of dose rates.

The results of detailed analysis of the CRBRP reactor indicate that the highest radiation field for CRBRP occurs at the inside surface of the fixed radial shield (FRS), at core midplane height. The total dose rate from Ni-59 at this point is 8.3 mrem/hour with 5.1 mrem/hour originating from the isotope's continuous spectrum of K-electron

capture gamma-rays and 3.2 mrem/hour from its 8.3 keV X-ray. This result is in marked contrast to the NYPIRG report which concluded that the Ni-59 dose rates at plant shutdown would be 1.47×10^4 rem/hour from the 8.3 keV X-Ray from Ni-59 decay and 3.4 rem/hour from K-electron capture gamma rays. In regard to CRBRP the NYPIRG analysis would erroneously over predict dose rates by a factor of 1,000,000.

Q.14. Would you describe the impact of Nb-94 on CRBRP decommissioning?

A.14. In the systematic search for neutron activation products which could impact the CRBRP decommissioning modes discussed previously, Nb-94 which is produced from neutron activation of Nb-93 was identified. The parent Nb-93 is a trace element or alloying constituent in the materials of construction used in the CRBRP. The prediction of the Nb-94 radioactivity in CRBRP components used the same analytical approach as described for Ni-59. Nb-94, which has a more energetic decay gamma emitter than Ni-59, produces higher dose rates than Ni-59 adjacent to the CRBRP permanent components. Thus, in the time frame of interest, any conclusions drawn as to the significance of Nb-94 would also apply to Ni-59.

The Nb-94 dose rate calculated at the FRS inner surface at the core midplane is 2.2 rem/hour. In contrast, the Co-60 dose rate calculated at the FRS core midplane inner surface is 1.3×10^5 rem/hour. This level

would require approximately 100 years to decay to levels below that of Nb-94. Thus for the first 100 years, Co-60 would be the isotope controlling the dose rate, after which Nb-94 would control.

Q.15. In the case of prompt dismantlement, DECON, what would be the controlling isotope?

A.15. For prompt dismantlement, Co-60 would be the controlling isotope in the decommissioning process and the presence of either Ni-59 or Nb-94 would not affect the cost or environmental impact associated with decommissioning. Prior decommissioning experience and the estimated cost for CRBRP decommissioning, both of which are implicitly based upon consideration of Co-60 as the controlling isotope, are not affected by consideration of Nb-94.

Q.16. In the case of decommissioning by prompt dismantling, would your conclusions be affected if there were multiple long-lived isotopes of concern in the CRBRP materials of construction?

A.16. No. More than 99% of the radiation level associated with the activated structures would be the result of decay of Co-60. The techniques used for prompt dismantlement would be controlled by the radiation level associated with Co-60.

Q.17. For the disposal of the activated materials at a repository site, would your conclusions be affected if there were multiple long-lived isotopes in those materials?

A.17. No. Our analysis of decommissioning costs for prompt dismantling recognizes that long term isolation of Co-60 is required. The other isotopes, regardless of numbers, would all be contained in the same activated material and would be packaged together for long term isolation at a repository site.

Q.18. What are your conclusions regarding Contention 8?

A.18. Our conclusions are:

(1) The CRBRP can be decommissioned by prompt total dismantlement;

(2) For reasons given above, that the upper bound cost for decommissioning the CRBRP should not exceed about $110 million in 1980 dollars;

(3) For reasons given above, that the implications of Ni-59 and other relevant trace element activation products would not significantly affect the cost estimate for decommissioning the CRBRP.

STATEMENT OF QUALIFICATIONS

James F. Murdock

Current Position: Reactor Engineer
Reactor and Plant Components Branch
U.S. DOE - CRBRP Project Office

Mr. Murdock received a B.S. degree in Metallurgical Engineering from Virginia Polytechnic Institute in 1959 and a M.S. degree from the University of Tennessee in 1964.

From 1950 to 1957 he was employed at the Oak Ridge National Laboratory as a cooperative engineering student in the Solid State Division. His work involved assembly and disassembly in hot cells of experiments from test reactors.

From 1959 to 1967 he was employed as a metallurgist in the Metals and Ceramics Divison of the Oak Ridge National Laboratory. He performed research studies of radioisotope tracer diffusion in pure metals and alloys.

From 1967 to the present, he was employed by the United States Atomic Energy Commission (then ERDA and now DOE). After an intensive training program in reactor theory and practical operation in the Operations Division of the Oak Ridge National Laboratory, he was assigned to the position of Site Representative for the Division of Reactor Development and

Technology (DRDT) at the Boiling Nuclear Superheat Reactor (BONUS) near Rincon, Puerto Rico. He was liaison between the contractors at BONUS and the various AEC Program and contracting organizations during the decommissioning of the BONUS facility. He provided review and comment on the decommissioning plans and approval of the detailed procedures for decommissioning. Upon completion of the BONUS decommissioning in 1970 he was assigned the position of Site Representative for DRDT at the LaCrosse Boiling Water Reactor, Genoa, Wisconsin. He was liaison between the operating and support organizations at LACBWR and the AEC Program and contracting organizations providing on-site monitoring of the operating plant. In 1972 he assumed the additional duties of Site Representative for the decommissioning efforts at the Elk River Reactor, Elk River, Minnesota. His responsibilities at Elk River were the same as his responsibilities at BONUS. Upon the sale of LACBWR to the Dairyland Power Cooperative in 1973 and subsequently the completion of the Elk River dismantlement in 1974 he was assigned the position of Site Representative for the Chicago Operations Office (CH) at the Westinghouse Advanced Reactors Division (W-ARD), Madison, Pennsylvania. He performed liaison duties between the CH Demonstration Project Office and the Division of Reactor Research and Development and the W-ARD for the CRBRP and the FFTF. In 1975 he was assigned to the Engineering Division of the CRBRP Project Office. He was responsible for the engineering management of the reactor internals design and the supporting

development programs and engineering planning of the Operations, Maintenance and Testing Program of the CRBRP. From 1978 to the present time he has been assigned to the Reactor and Plant Components Branch of the Engineering Division with responsibility for the design, fabrication, and procurement of reactor and plant hardware.

STATEMENT OF QUALIFICATIONS

Albert A. Weinstein

Current Position: Manager of Engineering
S. M. Stoller Corporation

Mr. Weinstein is manager of Engineering at SMSC responsible for the areas of reactor servicing and operations, mechanical systems design, and real-time systems applications.

As Manager of Engineering he provides consulting services to nuclear utilities on refueling systems analysis, plant operational and maintenance support, design review and quality assurance, and plant arrangement and system design. In this context he is responsible for safety analysis and licensing support in special areas, such as cask drop, and fuel handling accidents, and leads SMSC efforts in support of facilities design review, including fuel storage pool arrangements and fuel inspection and reconstitution facilities.

Mr. Weinstein also leads activities at SMSC related to nuclear power plant decommissioning. He served on the AIF task force assigned to monitor and review the preparation of AIF/NESP-009SR, "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives". He has reviewed LWR plant designs for ease of decommissioning, and has participated in the

actual decommissioning of the BONUS nuclear plant in Puerto Rico, where he served as the resident engineer, responsible for program definition, planning and scheduling. Mr. Weinstein has directed the preparation of decommissioning cost estimates for both PWR and BWR nuclear power plants.

He is presently also Project Manager for the D.C. Cook NPP combined RE&M/Security System and as such, is responsible for system development and installation, client and subcontractor interface and other project related activities. Mr. Weinstein has been extensively involved in the design and application of the RE&M System since its beginnings and continues in this capacity. He is also responsible for service administration after installations are complete.

Mr. Weinstein entered the nuclear business in 1957 at Combustion Engineering Company, where he participated in the mechanical design and analysis of the SIC reactor core with responsibility for startup and testing procedures relating to mechanical safety of the core. He assisted in subsequent reactor disassembly and inspection of radioactive core components. In 1960 he joined the United Nuclear Corporation, and in 1976 Mr. Weinstein was named Manager of the Engineering Section of the Mechanical Design Department.

Mr. Weinstein has directed the design of fuel inspection equipment and supervised the on-site inspection of commercial irradiated fuel, utilizing underwater video equipment and measuring devices. While at UNC Mr. Weinstein served on the

Technical Support Team under contract to the AEC to provide technical assistance for the Elk River, BONUS, and LaCrosse Nuclear Reactor Plants. He was assigned lead responsiblity for the technical support for the BONUS Nuclear Reactor in Puerto Rico.

Mr. Weinstein received a B.S. in Civil Engineering from the University of Southern California in 1954, and a M.S. in Applied Mechanics from the University of Connecticut in 1959. He has performed additional graduate work in engineering mechanics and is a licensed professional engineer in the State of Connecticut.

STATEMENT OF PROFESSIONAL QUALIFICATIONS

Richard K. Disney

Current Position: Manager, Shielding Analysis
Nuclear Systems Engineering
Engineering Department
Westinghouse Electric Corporation
Advanced Reactors Division

Education: B.S. in Nuclear Engineering, Kansas State University, 1958 Graduate Studies, Washington University of St. Louis, 1962.

Since 1971, I have been employed at the Westinghouse Advanced Reactors Divsion, Waltz Mill Site at Madison, Pennsylvania, where I have been assigned various management positions.

I have been Manager of the shielding analysis group at ARD since 1974 with primary responsibility for the Clinch River Breeder Reactor Project radiation protection, radiation analysis, and shielding design efforts. In this capacity, I have been responsible for efforts to define the radiation shielding and radiation protection philosophy of CRBRP. Included in the responsibilities are prediction of radiation environments in

CRBRP reactor systems, development of the ALARA review program for CRBRP, review of radiation shielding designs of CRBRP plant component and systems, technical guidance of experimental programs related to LMFBR radiation shielding, review of the refueling system radiation protection and shielding design efforts, and development of radiation source terms for components, systems, and radiological and safety analyses or studies.

In 1972-1973 I was responsible for the radiation shielding design, radiation environment predictions, and personnel radiation exposure predictions for the Fast Flux Test Facility sodium-cooled nuclear test reactor. Responsibilities included review of plant shielding, definition and review of radiation shielding experimental programs, definition of component radiation environment, definition of radiation source terms, analysis of reactor system shielding systems, prediction of radiation shielding performance, and personnel radiation exposure predictions.

In 1971 I was responsible for the development of advanced radiation transport methods for use in LMFBR radiation shielding analysis and the implementation of these methods on large scale mainframe computer systems.

In the period 1968-1971 I was assigned to a management position in the NERVA (Nuclear Engine for Rocket Vehicle Application) Project at the Astronuclear Laboratory of Westinghouse Electric Corporation at Large, Pennsylvania. In

this capacity, I was responsible for the analysis of nuclear radiation shielding experiments and for the development and evaluation of advanced analytical techniques and data for nuclear, radiation, and shielding design in the NERVA Project.

In the period 1965-1968 I was assigned to a variety of projects including the NERVA Project. I was responsible for the development of advanced computational methods including advanced discrete ordinate transport methods. I was also responsible for predictions of radiation environments in the large-size propulsion reactors for space vehicle applications. In addition, I established radioisotope fuel form and radiation shielding requirements for an implantable circulatory support system (artificial heart).

In the period 1962-1965 I was assigned to the Advanced Projects Department of the Astronuclear Laboratory. I was responsible for radiation and shielding analysis of advanced nuclear-electric propulsion systems and nuclear rockets, nuclear design of compact liquid metal reactors, and shielding analyses for the Lunar Base Nuclear Power Plant conceptual design study.

During the period 1961-1962 I was employed by the Internuclear Company of Clayton, Missouri. In this capacity I was responsible for the design and specification of the radiation shielding for two university research reactors, the 5 Mwt Kansai Research Reactor at Kyoto, Japan and the 10 Mwt Missouri University Research Reactor at Columbia, Missouri.

In the period 1958-1961 I was employed by the Babcock &

Wilcox Company at the Atomic Energy Division, Lynchburg, Virginia. Job responsibilities were in radiation source definition and radiation shielding design on the following projects: (1) the Consolidated Edison No. 1 Nuclear Power Plant, (2) the N.S. Savannah Nuclear Propulsion Merchant Ship, and (3) the Advanced Test Reactor materials test facility in Idaho Falls, Idaho.