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)	)	

AFFIDAVIT OF JACQUES S. BOEGLI,
EDWARD F. BRANAGAN, JR., AND RICHARD JOHN SERBU
IN SUPPORT OF SUMMARY DISPOSITION OF DES CONTENTION 19

1. I, J.S. Boegli, being duly sworn, do depose and state: I am an employee of the U.S. Nuclear Regulatory Commission in the Effluent Treatment Systems Branch, Office of Nuclear Reactor Regulation. I am responsible for the review and evaluation of radioactive waste treatment and effluent control systems and for the calculation of effluent source terms for nuclear power reactors. My professional and education qualifications are attached to this statement. I certify that I have personal knowledge of the matters set forth herein with respect to the above areas for which I am responsible, and that the statements made are true and correct to the best of my knowledge.

2. I, Edward F. Branagan, Jr., being duly sworn, do depose and state: I am a Health Physicist in the Radiological Assessment Branch, Division of Systems Integration within the Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached. I certify that I have personal knowledge of the matters set forth herein

with respect to assessment of the impact from exposure of the public to radioactive effluents from spent fuel stored at Catawba. The statements made are true and correct to the best of my knowledge.

3. I, Richard John Serbu, being sworn, do depose and state: I am an employee of the U.S. Nuclear Regulatory Commission (NRC). My present position is Health Physicist, Radiological Assessment Branch, Division of Systems Integration within the Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached. I certify that I have personal knowledge of the matters set forth herein with respect to assessment of occupational exposures to on-site personnel and that the statements made are true and correct to the best of my knowledge.

4. This affidavit addresses DES Contention 19, which reads as follows:

> "Failure to evaluate the environmental costs of operation of Catawba as a storage facility for spent fuel from other Duke facilities compromises the validity of the favorable cost-benefit balance struck at the construction permit phase of this proceeding. Since the CP stage hearing, Duke Power has considerably expanded the Catawba spent fuel pool capacity and provided for denser storage of irradiated fuel. FSAR Table 1.2.2-1. Applicants intend to use Catawba for storage of irradiated fuel from the McGuire and Oconee nuclear facilities of Duke Power Company. FSAR 9.1.2.4; OL Application, pp. 11-12."

In admitting this contention, the Licensing Board stated that ". . . the primary focus of DES 19 would be on the environmental effects of routine releases from such [Oconee and McGuire] transhipped fuel during normal operations at Catawba." (Memorandum and Order, February 25, 1983, p. 9).

5. In the FES, the Staff analyzed "the environmental costs of operation of Catawba as a storage facility for spent fuel from other Duke facilities" in the following manner. The major environmental pathways of exposure of humans were considered. Tables D.1 and D.4 included

releases from spent fuel from Catawba and the spent fuel that is expected to be stored at Catawba from Oconee and McGuire. In its review, the Staff determined that the releases of radioactive materials from fuel stored in the Catawba Spent Fuel Storage Facility (SFSF) were very small fractions of the total releases from normal operations of the entire Catawba facility. Similarly, dose commitments to a maximally exposed individual and to the population from operating Catawba included releases from the storage at Catawba of spent fuel from Catawba, Oconee and McGuire. (See Tables D.6, D.7, D.8). Finally, estimates of dose to workers from normal handling of spent fuel casks from Oconee and McGuire were evaluated at Sec. 5.9.3.1.2 of the FES (p. 5-19). The Staff concluded that the systems as now designed and built are capable of controlling effluent releases, including those from stored spent fuel from other Duke facilities, to meet the dose-design objectives of Appendix I to 10 CFR 50. In addition, the estimated doses to individual members of the public and to the general population from exposure to all effluents from the facility were very small fractions of the annual doses from exposure to background radiation. Further, the Staff, in its review, determined that estimated doses to individual members of the public and to the general population from exposure to effluents from the SFSF were very small fractions of the estimated doses from exposure to all effluents.

6. This affidavit sets forth in more detail than in the FES the sources and amounts of routine releases of radioactive materials, and resultant dose commitments both onsite and offsite which may be expected from the SFSF. In the following analysis the Staff has evaluated the environmental impact associated with the receipt and storage of spent

fuel in the SFSF by addressing (1) the types of releases from the SFSF (liquid, gaseous, solid) leading to possible exposure of the public, and (2) the possible occupational doses to workers associated with spent fuel storage and fuel handling. The Staff's environmental evaluation encompasses the contribution of the expanded spent fuel storage facility, at full capacity, and includes the contribution attributable to receipt and storage of Oconee and McGuire spent fuel at Catawba.

Routine Releases

7. The amount of radioactivity which will be released into the environment from the SFSF, and the amount of radioactivity which may be attributed to the storage of Oconee and McGuire spent fuel in the Catawba SFSF, may be determined based on the capacity of the SFSF, the release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous streams, and the plant-specific design features used to treat and store radioactive material from fluid streams by collection on media for disposal as solid waste. Such estimates of routine radioactive releases are called the "source term" and are derived by techniques presented in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gases and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code) using inplant measurements at many of the operating nuclear power plants. The SFSF at Catawba, Unit 1 and Unit 2, are independent, as required by 10 CFR Part 50, Appendix A, GDC No. 5. However, they have similar design features such that the routine releases may be calculated on a per unit basis.

Capacity of the SFSF

8. In determining the routine releases per unit, the Staff considered the SFSF to be full at its design capacity of 1418 fuel assemblies. Although the environmental statement published upon consideration of the Catawba construction permit application was based on a SFSF with 265 fuel assemblies, the FES and Safety Evaluation Report published for the operating license application are based on the increased capacity. This increase would allow for 193 fuel assemblies from off-loading a fully loaded core at Catawba at any time, plus storage space for approximately 258 fuel assemblies from Catawba that are less than five years out-of-core, and storage space for approximately 967 fuel assemblies that would be over five years out-of-core. Applicants have proposed to store spent fuel from Oconee and McGuire at Catawba which would be at least five years out-of-core. However, for the purpose of calculating routine releases, the Staff considered for the FES that the Catawba fuel assemblies and those from McGuire and Oconee to be equivalent after five years out-of-core since the assemblies are approximately the same size (8.4 x 8.4 x 144 inches), the same materials (Zircaloy-4/Inconel 718), contain the same UO_2 fuel (about 1150 pounds UO_2/assembly) are used to about the same burnup rate (33 MWD/Kg U) and would have approximately the same fission product inventory after the same amount of time in storage. Therefore, the origin of fuel assemblies over five years out-of-core would not impact the routine releases. Although the date at which maximum capacity would be reached would be advanced by the storage of McGuire and Oconee fuel assemblies in the SFSF, the Staff bases for routine releases considered a full SFSF at any time during the operating

life of the Catawba Nuclear Station. Therefore, the routine releases based on a full SFSF will not be increased by the shipment of five year old fuel assemblies from McGuire and Oconee, and the routine releases at a time where the SFSF is below capacity will be less than at full capacity.

Release Mechanisms and Treatment Provisions

9. The release mechanisms for routine releases are the same for the underwater storage of fuel assemblies at all facilities. During the movement and storage of fuel assemblies in the SFSF, both volatile and nonvolatile radioactive materials may be transferred to the SFSF pool water from the outer surface of the fuel assemblies or from defects in the fuel assembly cladding. Most of the outer surface material consists of activated corrosion products, such as Co-58, Co-60, Mn-54 and Fe-59, which are nonvolatile. The Staff estimates that this outer surface material constitutes about 0.001 Ci/assembly and that most of the material is insoluble. The spent fuel pool cleanup system removes the insoluble material transferred to the pool water by continuous recirculation through filters and removes any soluble material by demineralization. Most of the surface material is removed during the first few months of storage in the pool water such that there would be little contamination of the SFSF by assemblies shipped from McGuire or Oconee that have been stored in their respective spent fuel pools at least five years prior to shipment to Catawba. None of the surface corrosion products are volatile since they are salts and metal oxides.

10. The radioactive materials that may be transferred to the SFSF pool waters from cladding defects are generally nonvolatile fission

products, such as Cs-134, Cs-137, Sr-89 and Sr-90. The abundance of fission products transferred into the SFSF water is dependent on the fission yield, the time since irradiation in core, the size of any cladding leak and the temperature of the cladding. The Staff estimates that the fission products transferred to the pool water due to cladding defects constitutes about 0.01 Ci/assembly at the time of unloading from the core, less than 0.001 Ci/assembly after about 20 days in storage, and an undetectable amount after several years in storage. These soluble and insoluble radioactive materials are continuously removed from the pool water by the demineralizers and filters in the spent fuel pool cleanup system. Measurements at operating plants have shown that most defects or pinhole perforations in the fuel cladding are self-sealing when the cladding temperature is relatively cool, approximately 180°F. This self-sealing condition occurs in about 20 days after a fuel assembly is removed from the core, and together with radioactive decay, greatly reduces the net effect of nonvolatile fission products in the SFSF pool water. Since Oconee and McGuire spent fuel received and stored at Catawba would be at least 5 years out-of-core, the contribution of radioactive materials due to defects in such fuel would be undetectable.

11. The release mechanism for volatile fission products is the same as for the nonvolatile; however, their solubility in the SFSF pool water must be considered. Generally the SFSF water is maintained below 140°F to reduce the cladding temperature, reduce water evaporation and to increase the solubility of gases and thereby contain most of the volatile materials. Radioiodines and most of the noble gases are

reduced by radioactive decay, together with the self-sealing condition of the fuel cladding. Tritium produced in the reactor coolant and within the fuel assembly is not a significant nuclide, since in the case of Catawba, there is no major mixing of reactor coolant water or fuel assembly transfer cask water into the SFSF. Operating experience has demonstrated that after 4 to 6 months there is no significant release of volatile fission products from fuel assemblies and the only significant noble gas nuclide attributable to long term storage of fuel assemblies would be Kr-85, which is at undetectable concentrations in the plant effluent after two years out-of-core.

12. The SFSF pool water is recirculated and continuously cooled, filtered and demineralized. The treatment removes radioactive materials from the water by filter and exchange media such that nonvolatile materials are collected for disposal as solid waste, and solids disposal is to a licensed burial site. During routine servicing and maintenance operations, excess water from the SFSF is treated in a similar manner by the liquid radwaste treatment system. There are no releases of SFSF pool water. Radioactive materials in gaseous effluents from the SFSF are collected by the fuel building ventilation system, treated by filters and absorbers for particulate and radioiodine removal (if any), monitored and released to the atmosphere via the plant vent. (These effluents are discussed further in the section entitled "Calculated Releases and Dose Impact".)

13. Prior to the publication of the FES, the Staff reviewed the release mechanisms for radioactive materials from spent fuel assemblies and the provisions for treating the liquid, gaseous and solid waste

generated by the operation of the Catawba SFSF. The Staff found that the release mechanisms were not altered by the number of fuel assemblies in the SFSF, and the treatment provisions were designed and installed adequately to meet the requirements of a full SFSF. Since there are no detectable transfers of radioactive material from the fuel assemblies to the pool water after five years of storage, there would be no difference between a full SFSF containing only Catawba spent fuel assemblies and a full SFSF containing fuel assemblies from McGuire and Oconee. The pool water radioactivity would be essentially the same.

Calculated Releases and Dose Impact

14. There are estimated to be essentially no liquid releases from the SFSF since it is a closed recirculation/treatment system. Therefore, radioactive materials in liquid effluents calculated for the FES and the SER for Catawba, Unit Nos. 1 and 2 did not include SFSF releases, and the proposal to store fuel assemblies from McGuire and Oconee would not change this conclusion.

15. Solid radioactive wastes, generated by the Catawba SFSF pool water filter and demineralizer treatment system are packaged and shipped to a licensed burial site. The Staff estimated that the volume of solid waste generated by disposal of filters and demineralizer exchange media from the spent fuel cleanup system would amount to about 6 cubic feet per year per unit at Catawba, containing approximately 0.1 Ci/cubic feet. The environmental impact of the transportation and disposal of these low level wastes are accounted for by the generic values in Table S-3, 10 CFR 51.20, as stated in the FES. As a conservative estimate, the Staff assumed that the amount of solid waste generated by the storage of

McGuire and Oconee fuel assemblies at the Catawba SFSF may be increased by 6 cubic feet per year per unit. There would be no increase in activity of this solid waste since there is no increase in the radioactivity of the pool water. The annual average amount of solid waste to be shipped from each unit at Catawba is estimated to be about 20,000 cubic feet per year. The total solid waste volume generated by the storage of McGuire and Oconee fuel assemblies at Catawba is estimated to be less than 0.1% of the preceding value and would not have any significant environmental impact not already considered in the FES and the SER for the Catawba Nuclear Station.

16. Due to the release mechanism for volatile fission products from fuel assemblies after several years out-of-core (described in paragraph 11, above), it is estimated that there would be no measureable releases of noble gases in the plant effluent. Using computer models, the Staff calculated in the FES, Table D.1 on page D-5, that the annual average release of Kr-85 for the auxiliary building stack, which includes releases from the SFSF, would be less than 1 Ci/yr averaged over the 30 year operational life of the plant, with the SFSF less than at full capacity some of this time. The Staff conservatively estimated that if the SFSF were at full capacity all of the time, the maximum routine release of Kr-85 would be less than 1 Ci/year or no more than 0.5% of the total annual release of Kr-85 from either unit.

17. Therefore, if fuel assemblies from McGuire and Oconee were shipped after 5 years out-of-core and stored in the Catawba SFSF a conservative estimate of the gaseous releases would be less than 1 curie per unit per year of Kr-85. The estimated doses to the total body and

skin of a maximally exposed individual are estimated to be much less than 0.1 mrem/year. The Staff's method for calculating the total body dose impact is provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I." Revision 1, October 1977. This dose is not significant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation and small compared to the fluctuation in the annual dose that an individual would receive from natural background radiation.

18. The dose to the total body of the population within a 50-mile radius of the Catawba Nuclear Station (estimated to be about 1,700,000 persons in the year 2000, FES, p. D-9) due to normal operation of the SFSF, with the storage of McGuire and Oconee fuel assemblies, is estimated to be less than 0.1 man-rem/year. This dose is a very small fraction of the annual dose of about 160,000 person-rems (FES, p. D-9) that this population would receive from natural background radiation. Thus, the Staff concludes that normal operation of the SFSF including the proposed storage of fuel assemblies from McGuire and Oconee in the Catawba SFSF will not have any significant impact on exposures offsite.

Occupational Dose

19. In addition to the effluent pathways, the DES/FES and SER considered occupational doses which might be associated with the handling and storage of spent fuel, including fuel transferred from other facilities. The SER, in Section 12.5, considered occupational doses and ALARA/radiation protection practices associated with fuel

handling and storage operations at the SFSF at Catawba, which was evaluated by the NRC Staff as follows:

> The storage of spent fuel at Catawba has been evaluated in accordance with Standard Review Plan Sections 12.2 and 12.3-12.4. The applicant has provided information which satisfactorily meets with our guidance and positions in the Standard Review Plan, including the requirements in 10 CFR Part 50, Appendix A; General Design Criteria-61; 10 CFR Part 70, 70.24; and Regulatory Guide 8.12, and is therefore acceptable. Additionally the radiation protection program, organization, and policies have been evaluated as indicated in Sections 12.1 and 12.5 of the Safety Evaluation Report, and are acceptable for the transfer and storage of spent fuel from Oconee and McGuire. No significant additional occupational doses should result from the storage of additional spent fuel at Catawba, since direct doses from stored fuel provide only a fractional contribution to spent fuel pool area dose rates in comparison to radioactivity in pool water. Similarly, dose increases due to the handling of spent fuel casks at Catawba would contribute only a very small fraction to the total projected dose for the facility.

20. The Staff evaluated and found acceptable Applicants' estimates of worker doses associated with spent fuel pool operations. These estimates are based on work area dose rates, the nature and location of work to be performed, and the time spent performing work in the pool area. Applicants' estimates reflect source terms, dose rates, work, and work times which are consistent with those measured and observed for similar facilities and operations throughout the industry.

21. Dose rates of 1 mrem/hr to 2 mrem/hr are typically measured at the surface of spent fuel pools, primarily due to the presence of contaminants such as Co^{60} and Co^{58} in the pool water. These contaminants are introduced into the spent fuel pool water as fuel is moved about in the pool and activated corrosion products (which typically adhere to in-core fuel assemblies) are disturbed and transfer to the water. However, as stated in paragraph 13, supra, Oconee and McGuire fuel, being stored over five years prior to storage at Catawba, would not

transfer defectable amounts of such containments to the Catawba SFSF. Therefore, while occupational dose rates from operations at Catawba are expected to reflect industry experience, generating dose rates in the range of 0.5 mrem/hr to 5 mrem/hr--and most probably generating dose rates of 1 mrem/hr to 2 mrem/hr during stable storage periods with the pool cleanup system in operation--no significant amount of these dose rates would be attributable to the Oconee and McGuire spent fuel proposed to be transshipped to Catawba.

22. Due to the shielding effect of water, the fuel assemblies themselves will contribute little to the overall dose rate at the surface of the spent fuel pool - a fraction of the dose rate originating from the pool water itself. A specific water level for radiation shielding is not required by regulations. Design guidelines for spent fuel pools, ANSI N 210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," recommends that radiation dose to personnel in normally occupied areas of spent fuel pools be maintained as low as reasonably achievable below 2.5 mrem/hr whole body dose during normal operations. This design guideline can be met by maintaining a minimum of 10 to 12 feet of water over fuel in storage--enough water to reduce the gamma dose rate from fuel assemblies to the range from 5 mrem/hr to 0.5 mrem/hr at the pool surface. However, dose rates at the surface of the pool with a typical water depth of 24 feet are on the order of only 10^{-6} mrem/hr due to direct radiation from recently discharged fuel assemblies.

23. At the CP stage Applicants made the following commitment:

> During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is 2.5 mr/hr or less. This is

accomplished by maintaining a minimum of 10 feet of water above the top of the fuel pellets in the fuel assembly during all handling operations.

(Source: Applicant's PSAR, Section 9.1.4.3.4, "Radiation Shielding")

To provide further assurance that a minimum 10 feet of water is maintained above the fuel, keeping dose rates below 2.5 mrem/hr, the Applicant has committed to provide limited maximum lift height for handling equipment used to raise and lower spent fuel. (Source: Applicant's FSAR, Section 9.1.4.1, "Design Bases"). The Applicants' assumptions and bases for shielding design and operations applicable to the SFSF were evaluated by the Staff, and the Applicants' design methods, including the use of source terms, cross section data, shield and source geometries, and radiation transport calculational schemes, were found to be consistent with accepted practice. (Source: Catawba SER, Section 12.3.2). The storage of spent fuel from McGuire and Oconee at Catawba does not impact the Applicants' commitments or the Staff's findings, since fuel storage assessments were based on recently discharged fuel (and full capacity) rather than five year old spent fuel (and less than full capacity).

24. As noted in the FES (Table D.9), the Staff has estimated 480 person-rems as the total body dose to plant workers for a year of operation for a single unit at Catawba. Normal fuel handling operations in the fuel handling building for Catawba are expected to result in an average total body dose of about 1.5 person-rems per year per unit. As stated in Section 5.9.3.1.2., additional handling of spent fuel from Oconee and McGuire is estimated to result in a total body dose of 0.029 person-rem per spent fuel shipment, or 8.7 person-rems for 300 shipments

(the maximum number of shipments per year proposed by Applicants). Doses from fuel handling are thus only a small fraction of total dose for the Catawba facility.

25. In summary, the Staff evaluated in the FES the environmental impacts of spent fuel stored in the Catawba SFSF, as expanded since the original construction permit application. This evaluation included the operation of the Catawba SFSF as a storage facility for spent fuel from Oconee and McGuire. The Staff's evaluation is contained in the FES, at pp. 5-19, 9-7, 9-8, 9-12, 9-13, and in Appendix D. A more detailed exposition of that analysis has been presented above.

26. The conclusions of the Staff's evaluation are as follows:

(a) The releases of radioactive material from fuel stored at Catawba, including fuel from Oconee and McGuire, are estimated to be very small fractions of the total releases from normal operations at Catawba.

(b) The Catawba effluent treatment systems as now designed and built are capable of controlling effluent releases, including releases from stored spent fuel from Oconee and McGuire, to meet the dose-design objectives of Appendix I to 10 CFR 50.

(c) The doses to individual members of the public and the general population exposed to effluents from fuel stored at Catawba are very small fractions of the annual doses from background radiation.

(d) Occupational doses attributable to spent fuel storage and handling operations, including handling and storage of spent fuel received from Oconee and McGuire are a small fraction of the total worker dose for the Catawba facility.

(e) As a result, the proposed operation of the Catawba SFSF has been fully evaluated, to include receipt and storage of Oconee and McGuire spent fuel, and found to have a small impact on the environment.

Edward F. Branagan, Jr.

Jacques S. Boegli

Richard John Serbu

Subscribed and sworn to before me
this 23 day of June, 1983

Notary Public

My Commission Expires: 7/1/86

J. S. Boegli
Professional Qualifications
Office of Nuclear Reactor Regulation

My name is J. S. Boegli. I am a lead nuclear engineer in the Effluent Treatment Systems Branch in the Office of Nuclear Reactor Regulation.

I attended Case Institute of Technology and was granted a B.S. in Chemical Engineering from Indiana Technical College in 1951. In 1952, I received a M.S. Degree in Chemical Engineering from Kansas State College and in 1955 to 1956 I completed advanced courses in chemical and nuclear engineering at the University of Michigan and applied Health Physics training at the Oak Ridge National Laboratory.

From 1953 to 1973 I was employed by the National Aeronautics and Space Administration and held positions as research engineer in heat and mass transfer, design engineer in nuclear reactor coolant, utilities, ventilation and radwaste systems, process systems supervisor, and technical consultant at the NASA Plum Brook Reactor in Ohio.

In July 1973, I joined the NRC (formerly AEC) as a senior nuclear engineer in the Effluent Treatment Systems Branch. I evaluate nuclear power plant systems and equipment for fission product removal, treatment of gas, liquid and solid radioactive waste, and radiation safety proposals as provided by license applicants to meet applicable NRC regulations. The duties involve development analytical models and performing calculations on the effectiveness of proposed radwaste systems, studying technological improvements and developing criteria governing radwaste processing, monitoring, shielding and handling. I evaluate the impact of radioactive effluents on the environs and prepare the radwaste section of the Environmental Statement for nuclear facilities.

EDWARD F. BRANAGAN, JR.
OFFICE OF NUCLEAR REACTOR REGULATION

PROFESSIONAL QUALIFICATIONS

From April 1979 to the present, I have been employed in the Radiological Assessment Branch in the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission (NRC). As a Health Physicist with the Radiological Assessment Branch, I am responsible for evaluating the environmental radiological impacts resulting from the operation of nuclear power reactors. In particular, I am responsible for evaluating radio-ecological models and health effect models for use in reactor licensing.

In addition to my duties involving the evaluation of radiological impacts from nuclear reactors, my duties in the Radiological Assessment Branch have included the following: (1) I managed and was the principal author of a report entitled "Staff Review of 'Radioecological Assessment of the Wyhl Nuclear Power Plant'" (NUREG-0668); (2) I serve as a technical contact on an NRC contract with Argonne National Laboratory involving development of a computer program to calculate health effects from radiation; (3) I serve as the project manager on an NRC contract with Idaho National Engineering Laboratory involving estimated and measured concentrations of radionuclides in the environment; (4) I serve as the project manager on an NRC contract with Lawrence Livermore Laboratory concerning a literature review of values for parameters in terrestrial radionuclide transport models; and (5) I serve as the project manager on an NRC contract with Oak Ridge National Laboratory concerning a statistical analysis of dose estimates via food pathways.

From 1976 to April 1979, I was employed by the NRC's Office of Nuclear Materials Safety and Safeguards, where I was involved in project management and technical work. I served as the project manager for the NRC in connection with the NRC's estimation of radiation doses from radon-222 and radium-226 releases from uranium mills, in coordination with Oak Ridge National

Laboratory which served as the NRC contractor. As part of my work on NRC's Generic Environmental Impact Statement on Uranium Milling (GEIS), I estimated health effects from uranium mill tailings. Upon publication of the GEIS, I presented a paper entitled "Health Effects of Uranium Mining and Milling for Commercial Nuclear Power" at a Conference on Health Implications of New Energy Technologies.

I received a B.A. in Physics from Catholic University in 1969, a M.A. in Science Teaching from Catholic University in 1970, and a Ph.D. in Radiation Biophysics from Kansas University in 1976. While completing my course work for my Ph.D., I was an instructor of Radiation Technology at Haskell Junior College in Lawrence, Kansas. My doctoral research work was in the area of DNA base damage, and was supported by a U.S. Public Health Service traineeship; my doctoral dissertation was entitled "Nuclear Magnetic Resonance Spectroscopy of Gamma-Irradiated DNA Bases."

I am a member of the Health Physics Society.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

RICHARD JOHN SERBU

PROFESSIONAL QUALIFICATIONS

I am presently assigned as a health physicist with the Radiation Protection Section of the Radiological Assessment Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

I graduated from the State University College of New York at Potsdam with a Bachelor of Arts Degree in Chemistry. I have work in a professional capacity in the field of radiation protection/health physics in association with nuclear power reactors since June 1973.

From June 1973 to April 1980, I held positions as Project Engineer, Dosimetry and Health Physics; Manager, Radiological Monitoring; Project Engineer, Radiological Training; Radiological Controls Supervisor; and Instructor, Chemistry and Radiological Controls at Knolls Atomic Power Laboratory. My responsibilities at KAPL included development, implementation, and management of radiological training programs, operational health physics/ALARA programs, and dosimetry programs. This includes broad experience in all aspects of reactor health physics/radiation protection; familiarity with reactor systems; radiation protection aspects of reactor startup; radiation protection for maintenance and refueling/overhaul; chemistry control programs; and compliance with established requirements. Since April of 1980, I have been with the Nuclear Regulatory Commission as a radiological engineer. In this capacity, I am responsible for the review and evaluation of radiation protection/ALARA (As Low As Reasonably Achievable) aspects of nuclear power reactor facility equipment and design, planning and procedure programs, and techniques and practices which are employed by nuclear reactor licensees and license applicants in meeting the standards for protection against radiation of 10 CFR Part 20.