STATE OF NEW JERSEY)
: SS.
COUNTY OF ESSEX

AFFIDAVIT OF ROBERT P. DOUGLAS

ROBERT P. DOUGLAS, being first duly sworn according to law, deposes and states:

1. I am employed by Public Service Electric and Gas Company as Licensing Manager and Acting Environment Manager. In that capacity, I was responsible for the calculation of offsite radiological doses for the Salem Generating Station, including demonstration of compliance with the requirements of 10 C.F.R. Part 20 and 10 C.F.R. Part 50, Appendix I. With regard to the application to the NRC to install new spent fuel racks in the Salem Unit 1 spent fuel pool capable of holding 1170 elements, I developed the radiological dose sections and supervised the response to questions in that area. A copy of my professional qualifications is attached, hereto, as Appendix A and incorporated by reference herein. I have reviewed the allegations made regarding the Coleman's Contention 9 in this proceeding.

2. I have assessed the offsite radiological effects of increasing the capacity of the Salem Unit 1 fuel pool. The results of such an evaluation show that the additional storage capacity causes only an extremely small increase in offsite doses.

3. Initially, contrary to the assertion contained in Coleman's Contention 9, the fact that the storage capacity is increased by a factor of 4.5 does not mean that the offsite doses will be correspondingly increased by the same factor. The percentage increase in offsite dose will be significantly less.

4. Most of the releases of radioactive material which contribute to offsite doses occur as the result of the initial transfer of fuel from the reactor to the pool, the initial storage and again during its transfer from the fuel storage pool to the shipping cask for shipment offsite. Inasmuch as these activities would occur whether or not the storage capacity were increased, i.e., the spent fuel rack modification increases only the storage capacity and not the frequency or the amount of fuel to be replaced for each fuel cycle, such doses should not be associated with the requested change.

5. Radioactive gases which might be released from the spent fuel pool consist of radioactive xenons such as Xe-131m, Xe-133, and Xe-135, radioactive iodines such as I-131 and I-133, Kr-85, and tritium (H-3).

6. During the period of interest, because of the half lives of these isotopes (except H-3) relative to Kr-85, the curies released for these isotopes will be substantially lower than for Kr-85. The release of these isotopes will occur during the first few months of fuel storage. Hence, increased fuel

storage time, i.e., beyond four years, will not result in any increase in releases to the environment of other than Kr-85 and H-3. No dectable additional tritium release is expected as a result of increased fuel storage time. See Paragraph 11, infra.

7. As part of the evaluation to assure compliance with 10 CFR Part 50, Appendix I, using the GALE Code contained in Regulatory Guide 1.109, a release from each Auxiliary Building of less than one curie per year with the original racks in place was calculated. [Application Revision 1 at 10.]

8. If it assumed that the release rate of Kr-85 is increased by a factor of 4.5 to correspond to the increase in the number of fuel elements being stored, a conservative assumption inasmuch as the release of Kr-85 is most likely to occur during the initial handling and first year of storage, and that all Kr-85 releases from this building were attributable to releases from the fuel pool, the maximum release from the auxiliary building would be 4.5 curies, an increase of approximately 3.5 ci/yr. The total plant releases of Kr-85 initially projected was 280 ci/yr. Thus, the maximum percentage increase due to spent fuel storage pool expansion would be consistently less than 1.25%. The maximum offsite dose resulting from the additional Kr-85 would be $1.6x10^{-6}$ man-rem/year.

9. The NRC Staff, using even more conservative assumptions, has also calculated the additional dose due to the change in spent fuel racks using even more conservative assumptions regarding Kr-85 for both Salem Units 1 and 2. The Staff concluded:

> With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time (beyond 4 years) would be krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed a few months, there is no significant release of fission products from defective fuel. However, as a measure of conservatism, we assumed that an additional 114 Curies per year of krypton-85 would be released from both units when the modified pools are completely filled. This assumption is based on the expected annual reload cycle and the total number of fuel assemblies that could be stored in the modified pool. This would result in an additional total body dose to an individual at the site boundary of less than 0.005 mrem/year. Such a dose would be insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. Furthermore, the additional total body dose to the estimated population within a 50-mile radius of the plant that would result from this assumption would be less than 0.005 manrem/year. Such a dose would be less than the natural fluctuations in the annual dose that this population would receive from natural background radiation. Under our conservative assumptions, these exposures represent an increase of less than 0.5% of the exposures from the station evaluated in the Salem 1/2 FES for an individual at the site boundary and the population. Based on the above scoping evaluation, we conclude that the proposed modifications will not have any significant impact on exposures offsite. [EIA at 7]

The increase in the maximum calculated dose to an individual of 0.005 and the increase of 0.005 man-rem/year within 50 miles are truly insignificant even considering the modification of the spent fuel pools for both Salem Units 1 and 2.

10. The Staff also concluded that since the I-131 inventory in the fuel will have decayed to negligible levels during the first four years of storage presently possible with-

out these modifications, the I-131 release will not be significantly increased. [EIA at 8]

11. The NRC Staff also considered the offsite doses due to I-131 and H-3 assuming that the peak bulk spent fuel pool water temperature may go as high as 134° F and may be above 120° F for as long as 32 days following the final incremental discharge of fuel that fills the pool to capacity. The Staff concluded in this regard:

> Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than would the SFP water. Therefore, even if there were a temporary higher evaporation rate from the spent fuel pool, the resulting increase in tritium and iodine released from the station would be small compared to the amount normally released from the station without these modifications as was previously evaluated in the FES. In addition, the station radiological effluent Technical Specifications, which will not be affected by this action, will limit the total releases of gaseous activity including those from stored spent fuel. If levels of airborne radioiodine become too high, the air over the SFP can be routed through charcoal filters for the removal of radioiodine before release to the environment. [EIA at 8]

12. Thus, even considering the cumulative radioactive releases from Salem Units 1 and 2, resulting from the installation of the larger capacity spent fuel racks, they are insignificant. However, even were the spent fuel capacity for the Hope Creek units increased and the increase of radioactive effluents were comparable to those from the Salem Generating Station, the total released from the Artificial Island units would still be extremely small. As an example, for four units,

even utilizing the Staff's conservative assumptions, the offsite dose to an individual resulting from the increase storage in all four units would still be in the order of 0.01 mrem per year and the man-rem increase would be in the order of 0.01 man-rem.

13. Ultimately, compliance with each facility's technical specifications which implements the requirements of 10 CFR Part 50, Appendix I assures that the total releases from that facility, including those associated with the increased storage in the spent fuel pool, are in the as low as reasonably achievable range.

14. Based upon my knowledge of the specific impacts associated with fuel pool expansion for Unit 1 and my general knowledge of the Salem and Hope Creek units, aside from the very minor increase in radioactive effluents should all of the units' spent fuel pools be expanded, I am aware of no other cumulative environmental impacts of significance associated with such action.*

Robert P. Douglas
ROBERT P. DOUGLAS

Sworn and subscribed to)
before me this 21st day)
of February, 1979.)

W. A. VanderClock
W. A. VANDERCLOCK
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Mar. 18, 1979

*The hypothesis that Kr-85 releases would increase by a factor of 9 for the two Salem units and by a factor of 18 if the Hope Creek units are considered is incorrect. If the assumption is made that Kr-85 release increase by a factor of 4.5 for one unit, then the factor increase is still 4.5 regardless of the number of units considered. For example, if two units would release 9 curies of Kr-85 with the fuel pool expansion and 2 curies without the expansion (one per unit), the overall factor increase is 4.5.

TECHNICAL QUALIFICATIONS
ROBERT P. DOUGLAS
LICENSING MANAGER
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
APPENDIX A

My name is Robert P. Douglas. My business address is 80 Park Place, Newark, New Jersey. I am Licensing Manager in the Licensing and Environment Department of Public Service Electric and Gas Company. I also am Acting Environment Manager. In this position, I manage all the technical and administrative matters of the Licensing and Analysis Division and the Environment Division of the Licensing and Environment Department. The Licensing and Analysis Division is involved with safety analysis of nuclear and non-nuclear PSE&G facilities, coordination and preparation of reports required for the licensing activities including permit applications, safety analysis reports, and topical technical reports, analysis of radiological impact of generating station operation, coordination of meteorological and radiological monitoring data collection programs and other licensing related responsibilities.

I was graduated from Cooper Union with a B.S. degree in Mechanical Engineering in 1964. In 1966, I received a Master of Science degree in Nuclear Engineering from Massachusetts Institute of Technology. In 1967, I received the Degree of Nuclear Engineer from Massachusetts Institute of Technology. I joined PSE&G in 1967 as an Assistant Engineer in the Mechanical Division of the Electric Department. From 1967 to 1974, my responsibilities included the radiological evaluation of PSE&G nuclear generating stations, safety analysis, site selection studies, environmental program considerations and other areas. In 1974, I assumed responsibility as head of the Nuclear Licensing Group in the Mechanical Division. In 1977, I was promoted to my present position. I have either participated in directly or supervised the preparation of the radiological impact evaluation of Salem Nuclear Generating Station, including analyses required for the PSAR, FSAR, Environmental Report; Appendix I to 10CFR50 evaluation and the radiological impact of the spent fuel pool expansion.

I am a member of the American Nuclear Society, the American Society of Mechanical Engineers, and am a registered professional engineer in New Jersey.