

NRC Central ✓
 OELD FF (2)
 Shapar/Engelhardt
 Christenbury/Scinto
 Cunningham/Karman
 BHSmith
 JEMoore/chron (2)

WRoss, 228 Phil.
 KHerring, 440 Phil.
 GZech, 528 Phil.
 RLobel, 516 Phil.

NRC Central

July 26, 1979

Mr. Marvin Lewis
 6504 Bradford Terrace
 Philadelphia, PA 19149

In the Matter of
 Public Service Electric & Gas Company
 (Salem Nuclear Generating Station, Unit No. 1)
 Docket No. 50-272
 Proposed Issuance of Amendment to
Facility Operating License No. DPR-70

Dear Mr. Lewis:

The enclosure to this letter and its attachments are in response to the questions you asked during your limited appearance statement made at the prehearing conference at Salem on March 15-16, 1979.

I hope that the Staff's response addresses the concern you expressed at the prehearing conference.

Sincerely,

JS

Janice E. Moore
 Counsel for NRC Staff

Enclosure
 As stated

cc w/encl. w/o attachments:
 See Salem Service List

7909140084

RL

OFFICE →	OELD	OELD	DOR	EB	RS
BURNAME →	JMoore:acb	GHCunningham	GZech	KHerring	RLobel
DATE →	7/26/79	7/26/79	7/26/79	7/25/79	7/27/79

NRC STAFF RESPONSE TO MR. MARVIN LEWIS
LIMITED APPEARANCE STATEMENT AT THE SALEM
PREHEARING CONFERENCE ON MARCH 15-16, 1979

For the seismic analysis of the Salem spent fuel storage racks, the licensee stated that modal and spatial responses were combined in accordance with the U.S.N.R.C. Standard Review Plan, Section 3.7.2 and Regulatory Guide 1.92, Rev. 1, entitled "Combining Modal Responses and Spatial Components in Seismic Response Analyses." Neither the Standard Review Plan nor the Regulatory Guide endorses the algebraic summations of intramodal codirectional responses to the separate earthquake components, which is the method of combination used in the piping analyses for the five shutdown plants. The algebraic summation method as used for Salem Unit 1's piping analyses is addressed in the response of Public Service Electric and Gas Co. to Inspection and Enforcement Bulletin 79-07 (Attachment A). This response (Attachment B) is currently under NRC review. The results of this review will be provided in the form of a Safety Evaluation Report (SER).

The computer code used for the linear elastic seismic modal analyses of the racks was SAP-4, which is a public domain code in widespread use. In addition, non-linear seismic analyses to determine the effects of gaps between the storage cells and the fuel assemblies, and the rack modules and the pool structures were also performed. The effects of the gaps between the storage cell and fuel assembly were considered in the same way as for Arkansas Power and Light Company's spent fuel storage racks, approved by the NRC in December, 1976. The impact factor thus calculated to determine the effects

of the gaps between the storage cell and fuel assembly by considering only one assembly inside one fuel storage cell, did not consider any potential reduction from the fact that all assemblies would not likely be impacting at the same time and in the same direction. The impact factor from this analysis was applied to the entire module. Additionally, this impact factor was comparable to that presented in other similar spent fuel pool modification applications. The methods for consideration of the gaps between the rack modules and pool structures were felt to be conservative since test data for coefficients of friction at the interfaces of the module legs and pool liner indicate that it is highly unlikely that the modules would slide under seismic loadings. Based upon the review of the analyses coupled with the commitment of the licensee to combine modal and spatial responses in an acceptable way, and NRC approved quality control and quality assurance programs, the rack design was felt to provide adequate assurance that the public health and safety were protected and that the appropriate General Design Criteria were satisfied. No computer code verification was felt to be required. The issue of computer code verification is presently under study of the NRC Staff.

In response to the second question posed in your limited appearance statement, there is no Regulatory Guide on "fuel rod degradation," as such. The term fuel rod degradation has no precise meaning. There is a Regulatory Guide which addresses reporting requirements for fuel damage (Regulatory Guide 1.16). There are Regulatory Guides to address criteria for conditions during certain accidents

for which fuel rod failure should be assumed to occur. Fuel rod failure is defined as a breach of the cladding which allows the release of fission products contained in the fuel rod.

There is no NRC requirement for the removal of failed fuel from the reactor core. Such situations are handled on a case-by-case basis. However, there are coolant activity limits which must be met. These limits imply that the extent of fuel failure must be below a certain level. Protection is therefore given to the public in terms of radioactivity in the coolant, rather than by limiting the number of failed fuel rods in the core. This type of limit also has a practical advantage since it is not possible to accurately infer the number of failed fuel rods in an operating reactor.

Fuel rods in some reactors have experienced a phenomenon termed pellet cladding interaction which involves a mechanical interaction of the fuel pellet against the cladding which can produce a brittle failure or "crack" in some cladding. If such a failure were to occur, most fission products in the fuel rod void space would be released to the coolant and would subsequently be processed by the reactor cleanup system. The remaining fission products are retained in the fuel. Data exist which show that when this failed fuel is subsequently placed in the spent fuel pool, the fuel rod will not further degrade and no fuel rod which was not already failed in the reactor will fail in the spent fuel pool (Reference 1). A comparison of the conditions in a typical spent fuel pool with those in the reactor shows that the water

temperature, neutron and gamma ray fluxes and pressure are significantly less in the spent fuel pool than in the reactor. Therefore, no damage to the fuel rods would be anticipated in the spent fuel pool.

"Twist" is not a term used to describe fuel rod behavior. We assume that what is meant is fuel rod bowing or fuel assembly bowing. These phenomena occur in the reactor core and the NRC Staff has reviewed and approved models for conservatively accommodating these effects on reactor safety analyses (Reference 2). These phenomena are due to large temperature gradients and large neutron fluxes such as those which exist in the core during reactor operations. In a spent fuel pool, the neutron flux is a factor of 10^8 smaller than in the reactor and there are no temperature gradients. Therefore, no bowing will occur in the spent fuel pool. Residual bowing from the reactor core can be accommodated by the design of the spent fuel racks. We are aware of no case in which a spent fuel handling problem in the spent fuel pool was caused by fuel rod or fuel assembly bowing.

It is not clear what is meant by the statement that "fuel rod degradation was not originally considered because it wasn't even known." Substituting the words fuel rod failure as previously defined for the nebulous term "fuel rod degradation," there has been a constant intensive effort over the whole history of the nuclear power industry to eliminate fuel rod failures. The failure rate now is on the order of one fuel rod failure per every 10000 per reactor-year. These failures are predominantly due to statistical defects in the manufacturing process.

All known fuel rod failure mechanisms have been considered in the design and licensing of both the Salem reactor and the Salem spent fuel pool.

References

1. Impacts of Reactor-Induced Cladding Defects on Spent Fuel Storage,
A. B. Johnson, Jr., Staff Scientist, Battelle, Pacific Northwest
Laboratories, PNL-SA-6917. (Attachment C)
2. Interim Safety Evaluation Report on Westinghouse Fuel Rod Bowing,
U.S. Nuclear Regulatory Commission Staff, April, 1976.