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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
HAROLD R. DENTON, DIRECTOR

In the Matter of	)	
	)	
PUBLIC SERVICE ELECTRIC & GAS COMPANY, ET AL.	)	Docket No. 50-311
(Salem Nuclear Generating Station,	)	
Unit No. 2)	)	

DIRECTOR'S DENIAL OF REQUEST UNDER 10 CFR 2.206

By petition dated August 3, 1979, and a supplement filed on August 31, 1979, Alfred C. Coleman and Eleanor G. Coleman requested that the issuance of the operating license for Salem Nuclear Generating Station, Unit No. 2 be stayed until various questions they raise, set forth in ten contentions, are resolved. They also requested that an adjudicatory hearing be held to consider the issues raised in their contentions. The Coleman's petition has been treated as a request for action under 10 CFR 2.206 of the Commission's regulations. Notice of receipt of the petition was published in the Federal Register, 44 Fed. Reg. 50932 (August 30, 1979).

Each of the Coleman's contentions are dealt with, in turn, below.

Contentions 1, 2 and 8

(1) The Nuclear Regulatory Commission has failed to act on information already known to it regarding projected needs for the PJM grid. The actual assumptions used, calculations performed, and results obtained to justify licensing Salem Unit No. 2 are ambiguous and inadequate.

(2) The Nuclear Regulatory Commission has failed to act on information already known to it regarding projected plant capacity, maintenance, and operating costs for similar facilities (cost/benefit analysis).

(8) The Nuclear Regulatory Commission has failed to require of the licensee cost-benefit analysis and consideration of alternative conversion of Salem No. 2 to natural gas or coal. (Final Environmental Impact Statement - Docket Nos. 50-272 and 50-311 - April 1973 - Pages 10 Alternatives, , 10-1 through 10-17 and 12-9 (12A and 12-16 (12X)). The NRC has failed to require in their analysis of "Request for Additional Financial Information Concerning Unit No. 2" (NRC request to PSE&G, April 18, 1978 - Olan D. Parr to R. L. Mittl) the alternative of conversion to natural gas or coal.

Response

All three of these contentions appear to be related to the Commission's obligations under the National Environmental Policy Act of 1969 (NEPA): need-for-power, operation and maintenance costs, cost/benefit analyses, and consideration of the alternative of conversion to natural gas or coal.

NEPA requires balancing of environmental costs against the expected benefits of major federal actions which significantly affect the environment before the actions are taken. "Need-for-power" is a shorthand expression for a primary aspect of the "benefit" side of the cost-benefit balance which NEPA mandates in considering the licensing of a nuclear power plant.[1/] "A nuclear plant's principal 'benefit' is of course the electric power it generates. Hence, absent some 'need for power', justification for building a facility is problematical." Duke Power Company (Catawba Nuclear Station, Units 1 & 2), ALAB-355, 4 NRC 397, 405 (1976).

The Commission has recognized, however, that uncertainty is inherent in any prediction of the need for or demand for the electricity to be generated by a nuclear plant.

> "[E]very prediction has an associated uncertainty and ...long range forecasts of this type are especially uncertain in that they are affected by trends in usage, increasing rates, demographic changes, industrial growth or decline, the general state of the economy, etc. These factors exist even beyond the uncertainty that inheres in demand forecasts: assumptions on continued use from historical data, range of years considered, the area considered, extrapolations from usage in residential, commercial, and industrial sectors, etc." Carolina Power & Light Company (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 & 4), CLI-79-5, 9 NRC 609, 610 (1979).

As the Atomic Safety and Licensing Appeal Board has stated, "[g]iven the legal responsibility imposed upon a public utility to provide at all times adequate, reliable service - and the severe consequences which may attend upon a failure to discharge that responsibility - the most that can be required is that the

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1/ Public Service Co. of New Hampshire (Seabrook Station, Units 1&2), ALAB-422, 6 NRC 33, 90 (1977).

forecast be a reasonable one in the light of what is ascertainable at the time made." Kansas Gas & Electric Company (Wolf Creek Generating Station. Unit 1), ALAB-462, 7 NRC 320, 328 (1978) (citation omitted).

In the course of fulfilling this obligation under NEPA, the Commission staff prepared an Environmental Impact Statement 2/ for Salem, Units 1 & 2 which concluded that the power to be generated by the facilities was needed to meet the applicants' future demands for electric power. 3/ No environmentally preferable alternatives were found to be available 4/ and the results of the cost/benefit analysis for the Salem facilities found that the environmental costs of the facility were outweighed by the benefits to be derived from the facility. 5/ The Commission, therefore, made a good faith assessment of the need for the Salem facility based on the information available to it and considered possible alternatives to the construction and operation of the facility.

Now the Colemans are requesting that the Commission re-open the need-for-power determination and its consideration of alternatives. Previous Director's Decisions have set forth the standard which is followed in consideration of such a request, i.e., whether the new information presented represents a significant new environmental impact or information which would clearly mandate a change in the

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2/ Final Environmental Impact Statement, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Units 1 & 2, April 1973.

3/ FES, supra, Section 9.6.

4/ FES, supra, Section 10.1.

5/ FES, supra, Section 11.

Commission's original determination of the need for the facility and the acceptance of the nuclear generation alternative.[6/]

The Colemans have not submitted any specific information on projected needs or costs of operation of Salem Unit 2 versus other, e.g., coal or natural gas, facilities. Rather, they merely assert that the Commission has failed to act on information known to it regarding costs and need for power and failed to require the licensee to do a cost-benefit analysis on the alternative of converting Salem Unit 2 to natural gas or coal.

While the exact nature of their assertion of failure to act on information regarding costs of similar facilities is unclear, it should be noted that under the Atomic Energy Act, Congress did not make this agency responsible for assessing whether a proposed nuclear plant would be the most financially advantageous way for a utility to satisfy its customers need for power.[7/] Furthermore, under our NEPA obligations, cost is relevant only to the extent an environmentally preferable alternative exists. If one does exist, then costs are considered to determine if they outweigh the environmental advantages to be gained. [8/]

6/ The staff has applied this standard in previous Director's denials under 10 CFR 2.206. See, e.g., Public Service Company of Indiana (Marble Hill Nuclear Generating Station, Units 1 & 2), DD-79-10, 10 NRC 129 (1979); Georgia Power Co. (Alvin W. Vogtle Nuclear Plant, Units 1 & 2), DD-79-4, 9 NRC 582 (1979). The staff believes that this standard is consistent with NEPA and is appropriate in considering under 10 CFR 2.206 petitions to reopen the record in a proceeding in light of the well-recognized need for finality in the administrative process. See Greene County Planning Board v. FPC, 559 F.2d 1227, 1233 (2d Cir. 1976), cert. denied, 434 U.S. 1086 (1978); Cleveland Electric Illuminating Co. (Perry Nuclear Power Plant, Units 1 & 2), ALAB-443, 6 NRC 741, 750-51 (1977).

7/ Consumers Power Company (Midland Plant, Units 1 & 2), ALAB-458, 7NRC 155, 162 (1978); see also Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, Docket Nos. 50-338SP, 50-339SP, Slip Op. at 10-16, (March 24, 1980) .

8/ Consumers Power Company, supra.

In the Environmental Impact Statement, the staff concluded that the alternatives of coal and natural gas did not reasonably exist.[9/] Even assuming they reasonably exist, when the sunk costs of the essentially completed Salem Unit 2 facility are considered,[10/] it would be extremely difficult to find that the benefits to be derived from converting the plant would be outweighed by the costs of such an action. Moreover, as set forth below, the staff has concluded that an already constructed nuclear unit is cheaper to operate than existing fossil fuel units because of lower fuel, operation and maintenance costs.

The 1980 nuclear fuel cost for Salem Unit 1 is estimated at 4.3 mills/kilowatt hours (kWh).[11/] The 1980 operation and maintenance cost is estimated at 2.0 mills/kWh.[12/] The Public Service Electric and Gas Company's 1980 average fuel cost for an oil unit is 29.34 mills/kWh, for a coal unit it is 14.7 mills/kWh.[13/] The 1980 average operation and maintenance cost for an oil unit is 5.9 mills/kWh while the coal unit's average 1980 operation and maintenance cost is 4.32 mills/kWh.[14/] The 1980 weighted average fuel cost for the oil and coal baseload units

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9/ FES, supra, Sections 10.1.5 and 10.1.6.

10/ Public Service Company of New Hampshire (Seabrook Station, Units 1 & 2), CLI-77-8, 5 NRC 503, 530-6 (1977).

11/ Uniform Statistical Reports - Year Ending December 31, 1978 - for Public Service Electric and Gas Company, April 24, 1979, Schedule XIX.

12/ "A Procedure for Estimating Non-Fuel Operation and Maintenance Cost for Large Steam-Electric Power Plants," ORNL/RM 6467.

13/ Uniform Statistical Reports, supra.

14/ "Steam Electric Plant Construction Cost and Annual Production Expenses 1977" DOE/EIA-0033/3(7).

is 23.07 mills/kWh and the weighted operation and maintenance cost is 5.22 mills/kWh. (The majority of Public Service Electric and Gas Company's baseload capacity is generated by oil fired units.)[15/]

The staff generally assumes using a 40 percent capacity factor for the initial year of operation of nuclear plants, 65 percent for the second year and 70 percent for the third through the 15th years of operation. It was also assumed that both nuclear and fossil fuel costs escalated at 10 percent per year and that operation and maintenance costs escalated at 8 percent per year.

Based on these capacity factors, Salem Unit 2 would be capable of producing 3.90 billion kWh of electricity the first year, 6.3 billion kWh the second year, and 6.8 billion kWh of electricity for the third through the 15th year. If the equivalent energy is generated by existing oil and coal units, the economic penalty in increased production costs alone would be $372 million in 1980 dollars for the first three years that Salem Unit 2 was not allowed to operate. Actually, the staff believes the applicants would use Salem Unit 2 to replace its more expensive oil base load capacity and not a combination of oil and coal. The resulting economic penalties in 1980 dollars for the first three years, considering the replacing by oil fired units alone, would be $487 million. In subsequent years the economic penalty would, in all probability, be even greater because even equivalent escalation rates impact more heavily on oil and coal which start at larger base values than nuclear production costs. Therefore, even in the highly unlikely event that demand

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15/ Uniform Statistical Reports, supra.

did not grow and Salem's output could be provided by existing units, the operation of Salem 2 would result in substantial production cost savings to the rate payers served by Public Service Electric and Gas.

Consequently, the Staff does not believe that the Colemans' petition and the information referred to therein represents a major new environmental impact or change in facts which would warrant re-opening consideration of the original NEPA analysis and its consideration of need for power alternatives and attendant cost/benefit analyses.

Contention 3

The Nuclear Regulatory Commission has failed to act on information already known to it regarding unresolved safety issues. "Public safety is the first, last and a permanent consideration in any decision on the issuance of a construction permit or a license to operate a nuclear facility." Power Reactor Development Corp. v. International Union of Electrical Radio and Machine Workers, 367 U.S. 396, 402, 81 S Ct 1529, 1532 (1961).

Response

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors, research results, NRC staff and Advisory Committee on Reactor Safeguards safety reviews, and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

Where it is concluded to be necessary, immediate action is taken to assure safety, e.g., the shutdown of nuclear reactors due to piping seismic design deficiencies in 1979. In other cases interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue

prior to making licensing decisions. In most cases, however the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. These issues have also been referred to as "unresolved safety issues." However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions of the facility(s) under consideration while the longer term generic review is underway.

These longer generic studies were the subject of a decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. In Gulf States Utilities Co. (River Bend Station, Units 1 & 2), ALAB 444, 6 NRC 760 at 775 (1977), the Appeal Board set forth the manner in which the staff should deal with unresolved generic safety questions for a particular facility.

The Appeal Board stated:

> "In short, the board (and the public as well) should be in a position to ascertain from the SER itself -- without the need to resort to extrinsic documents -- the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information would likely shed light on such alternatively important considerations as whether (1) the problem has already been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to insure that continued operation (if permitted at all) would not pose an undue risk to the public."

Since the issuance of the ALAB-444 the NRC has addressed this matter in its SERs (Safety Evaluation Reports) as they relate to specific applications.

With respect to Salem Unit 2, we have reviewed the generic safety issues in accordance with ALAB-444. Our evaluation of this matter will be addressed in

a supplement to the Safety Evaluation Report which will be issued prior to a decision to issue the operating license.[16]

Contention 4

The NRC has failed to consider the outstanding adjudicatory hearing on Salem Unit No. 1 with regard to expansion of the spent fuel pool, as it pertains to expansion at Salem Unit No. 2 located at a multi-nuclear complex.

Response

Public Service Gas and Electric Company has requested an amendment to the operating license for its Salem Unit 1 facility to provide additional storage capacity in the Salem Unit 1 spent fuel pool (SFP). That amendment as petitioner has correctly stated, is currently the subject of an adjudicatory proceeding. Further, by letter dated April 12, 1978, the licensee submitted Amendment No. 42 to the Application for License for Construction and Operation of the Salem Nuclear Generating Station Unit 2, which stated that the design changes proposed for the SFP at Salem Unit 1 would be made at Unit 2 as well.

In the course of evaluating the proposed spent fuel pool expansion for Unit 1, the Office of Nuclear Reactor Regulation prepared an Environmental Impact

[16] See Appendix C, Supplement No. 4, Safety Evaluation Report for Salem Nuclear Generating Station Unit 2, Docket No. 50-311, April __, 1980.

Appraisal (EIA) of the proposed amendment. That EIA was issued on January 15, 1979. The Salem Station Final Environmental Statement (FES) which was issued in April 1973 considered the environmental impacts of the Salem Station rather than for Salem Unit 1 alone. Since PSE&G has indicated it will make identical modifications to the SFP at Unit 2, the EIA addressed the cumulative environmental impacts of the expansion of both SFPs.

The Commission concluded in the EIA that the environmental impacts associated with the proposed modification to both facility spent fuel pools will not be significantly changed from those analyzed in the FES for Salem Units 1 and 2 issued in April 1973. Consequently, the cumulative environmental impacts of the expansion of the spent fuel pools at the two Salem facilities have been adequately assessed.

With respect to the spent fuel pool expansion of Unit 1, our safety evaluation is presented in "Unit 1 Modification of Spent Fuel Pool Storage," dated January 15, 1979. We concluded that since the proposed modifications to the Unit 2 spent fuel storage and spent fuel pool facilities are identical to those at Unit 1, they are acceptable on the basis of the Unit 1 Evaluation.

Contention 5

The Nuclear Regulatory Commission has failed to require an "independent" and separate "fire protection" water backup system for Salem Unit No. 2.

Response

We have reviewed the Salem Nuclear Generating Station Units 1 and 2 fire protection program and fire hazards analysis submitted by the licensee. This submittal, which was in response to our request for an evaluation of the fire protection program against the guidelines of Appendix A to BTP ABSCB 9.5-1, states that a common yard fire main loop may serve multi-unit nuclear power plant sites, if cross-connected between units. Sectional control valves would permit maintaining independence of the individual loop around each unit. For such installation, common water supplies may also be utilized with the water supply sized for the largest single expected flow. For multiple reactor sites with widely separated plants (approaching 1 mile or more), separate yard fire main loops should be used.

The Salem Units are not widely separated plants and, therefore, do not require separate and independent yard fire main loops. The fire protection water supply system is common to both units and consists of two full capacity diesel-engine-driven fire pumps. Each pump has a separate discharge header that is connected to the yard fire main loop. Post type indicator valves have been provided to isolate them in the pumps' discharge headers in the yard loop and in the yard loop itself to provide sectionalization so that independence of the loop around each unit can be maintained. The water supply source to the pumps is from two 350,000 gallon water tanks (each tank has 300,000 gallons dedicated to fire protection). The fire suppression system with the greatest demand is the 1400 gpm deluge system (primary) plus the 1000 gpm for the

manual hose station (backup). This 2400 gpm demand is within the design capacity of 2500 gpm for the system.

In addition to the above, the automatic sprinkler system and manual hose station hose standpipe systems are fed by the main yard loop with multiple connections to interior fire protection systems header. Each sprinkler system and manual hose station has an independent connection to the fire protection header fed from two directions, therefore, a single failure cannot impair both the primary and backup fire protection system.

Based on our review, we find that the Fire Protection Program for the Salem Nuclear Generating Station is adequate and with the scheduled modifications committed to by the applicants, will meet the guidelines contained in Appendix A to Branch Technical Position ASB 9.5-1 and the General Design Criterion 3 "Fire Protection."

Contention 6

The recommendations from the NRC Task Force contains 23 recommendations for administrative and design changes to Salem Unit No. 2, proposed requirements arising from "Lessons-Learned: Study of the Accident at TMI." These should be completed prior to licensing and commercial start-up as well as additional corrective action on potential defects.

Response

Over the past several months following the Three Mile Island accident, the NRC staff has been conducting an intensive review of the design and operational aspects of nuclear power plants and the emergency procedures for coping with potential accidents. The purpose of these efforts was to identify measures that should be taken in the short-term to reduce the likelihood of such accidents and to improve the emergency preparedness in responding to such events.

The TMI-2 related requirements for near-term operating license (NTOL) applications were initially identified in the January 5, 1980 memorandum from the Executive Director for Operations to the Commissioners, "TMI Action Plan Prerequisites for Resumption of Licensing." On February 6, 1980, a revision of this list of requirements based on the latest draft of the Task Action Plans as of February 6, 1980 was prepared and discussed with the Commission. These requirements were listed in two categories; those required prior to fuel load and low power testing operation up to five percent power (designated as FL) and those required prior to operation above five percent power (designated as FP).

These requirements were developed from all available sources such as the recommendations of the Bulletins and Orders Task Force, the Presidential Commission to Investigate TMI-2, and the NRC Special Inquiry Group, and those which resulted from the Lessons Learned Task Force Short Term Recommendations (NUREG-0578), and the Lessons Learned Task Force Final Report (NUREG-0585).

Those requirements in the February 6, 1980 list which resulted from the Lessons Learned Task Force Short Term Recommendations (NUREG-0578), and those resulting from the Advisory Committee on Reactor Safeguards (ACRS) review of that document and the additional requirements of the Director, Office of Nuclear Reactor Regulation, were previously approved by the Commission. On September 27, 1979, a letter was issued transmitting these requirements to all pending operating license applicants. On November 9, 1979, a letter clarifying these requirements was issued to all pending operating license applicants to assist in their understanding of our requirements.

The response of the Public Service Electric and Gas Company to our letters has been the subject of staff review since October 1979. Meetings were held with the applicants in Bethesda on November 20 and December 11, 1979, and February 26, 1980. Site visits were made on January 10 and 11, and February 27, 1980 to check hardware installation, review proposed support centers, and to review specific administrative procedures relating to operating personnel and accident response.

In addition, for all the remaining items in the February 6, 1980 listing of requirements, the staff and the applicants have had ongoing reviews and meetings concerning these requirements and the applicants' responses to these additional items. Further site visits were held, for example, the March 5-7, 1980 visit by a team headed by an Office of Inspection and Enforcement leader and composed of the NRR licensing project manager, the Office of Inspection and Enforcement site representative, and technical members from NRR. They evaluated the

onsite and offsite support centers and their staffing and the installed communitions system between the plant and NRC Incident Response Center. This evaluation included the review of licensee management organization and managerial capabilities.

Our evaluation regarding Three Mile Island matters will be presented in Supplement Number 4 of the Salem Unit 2 Safety Evaluation Report which will be issued prior to a decision to issue the operating license for the Unit 2 facility.[17]

Contention 7

The NRC has failed to consider the "menu for disaster" track record of Salem Unit No. 1 as it relates to known shutdown and power reductions (forced) for the following reasons:

A. Equipment Failure

B. Maintenance or Test

C. Refueling

D. Regulatory Restriction

E. Operator Training and License Examination

F. Administrative

G. Operational Error

H. Other

[17] See Part II of Supplement No. 4, Safety Evaluation Report, Salem Nuclear Generating Station Unit 2, Docket No. 50-311, April __, 1980.

as it affects the performance of Salem Unit No. 2. Additionally, the NRC has failed to recommend changes to Salem Unit No. 2 as a result of "Lessons Learned" at Salem Unit No. 1 mentioned operating status and "Reportable Occurrences" as filed in License Event Reports (LERs) since fuel loading 1976 to date. This corrective action as a result of "Lessons Learned" from Salem Unit No. 1 should be completed prior to licensing and commercial startup of Salem Unit No. 2.

Response

With respect to Salem Unit 1, actions for deficiencies identified by any means such as a reportable occurrence are manifested in one of two ways; design change or procedure modification. These items are routinely verified for Salem Unit 1 through our Office of Inspection and Enforcement inspection program.

In addition, inspection by our Office of Inspection and Enforcement has been conducted to verify that such corrective measures have been applied to Salem Unit 2. The enclosed applicable portions of the Office of Inspection and Enforcement inspection reports 50-311/78-47 and 50-311/79-23 (Enclosure 2) are examples of such inspection effort. It should be additionally noted that the basis for the Salem Unit 2 operating procedures has been the Unit 1 operating procedure, complete with all changes and iterations which have accrued from three years of use.

The Office of Inspection and Enforcement inspection program has verified, through sampling inspection, that corrective measures taken at Salem Unit 1 have been considered for applicability at Salem Unit 2 and where applicable, have been incorporated.

Contention 7A

The Nuclear Regulatory Commission has failed to require the licensee and/or the manufacturer of reactor/steam generators to retrofit, as a result of testing, evaluating and analysis from "Lessons Learned" from the 1974 incident in Switzerland (Westinghouse reactor) and Davis-Besse Unit No. 1, Ohio (Licensee: Toledo Edison Company - Docket No. 50-346).

Response

At Davis-Besse Unit No. 1 and at the reactor in Switzerland of a Westinghouse design, both failures of the relief and/or safety valves to close resulted in small break loss-of-coolant accidents. In both of these cases actuation of engineering safety features and an appropriate reactor operator action prevented the event from evolving into a situation similar to that experienced at Three Mile Island Unit 2, even though there were a number of similarities between the Three Mile Island Unit 2 event and the events at these two reactors.

In NUREG-0578, the NRC staff's Three Mile Island Unit 2 Lessons Learned Task Force has disclosed a number of actions in the areas of design, analysis, and plant operations that will deal with the events similar to the ones that took place at Davis-Besse Unit No. 1 and at the Westinghouse designed reactor in Switzerland. The response to Contention 6 addresses the requirements which have been imposed on the applicants and the staff's evaluation of the implementation of those requirements.

Contention 8

The Nuclear Regulatory Commission has failed to require of the licensee cost-benefit analysis and consideration of alternative conversion of Salem No. 2 to natural gas or coal. (Final Environmental Impact Statement - Docket Nos. 50-272 and 50-311 - April 1973 - Pages 10 Alternatives, 10-1 through 10-17 and 12-9 (12A and 12-16 (12X). The NRC has failed to require in their analysis of "Request for Additional Financial Information Concerning Unit No. 2" (NRC request to PSE&G, April 18, 1978 - Olan D. Parr to R. L. Mittl) the alternative of conversion to natural gas or coal.

Response

See response to Contentions 1 and 2.

Contention 9

The NRC has failed to review and compel the licensee to explain apparent discrepancies in seismic findings by Dames and Moore for PSE&G and Delmarva Power and Light Company. (Summit Nuclear Plant - Delaware) as it relates to the effect of a possible earthquake. The final Safety Analysis Report reflects there is no earthquake fault in the vicinity of Artifical Island, site of Salem Nuclear Generating Station Nos. 1 and 2.

This appears to be in contrast to the study and findings of the University of Delaware which states there is a fault down the middle of the Delaware River.

This study is available to the NRC staff. The NRC staff order for seismic inspection of 29 reactors failed to include reactor containment structure, fuel handling, and spent fuel facilities. This must be determined prior to licensing Salem Unit No. 2. (Attachment - Article from "Today's Sunbeam," August 24, 1979). The NRC is already aware of the condition of the containment building (reactor) (cracks - NRC inspection report) and is unable to determine width, depth, extent or cause because of sand blasting by the licensee prior to NRC inspection.

Response

The geology and seismology of the Salem site were reviewed during the construction permit stage by the U.S. Atomic Energy Commission (USAEC) staff (now the NRC), the U.S. Geological Survey (USGS) and the U.S. Coast and Geodetic Survey (USC&GS), the seismological review group of which is now part of the USGS. The conclusions from that review are summarized in the Safety Evaluation Report for the Salem Nuclear Generating Station Units 1 and 2 dated July 16, 1968. In that report it was concluded that there were no identifiable geologic structures that could be expected to localize earthquakes in the site vicinity, and that 0.2g and 0.1g for the safe shutdown earthquake and operating basis earthquake would provide adequate earthquake protection for the plant.

On September 25, 1968 the Commission issued provisional construction permits for Units 1 and 2. Subsequent to this action, in 1972 and 1973, Dr. N. Spoljaric of the Delaware Geological Survey reported faulting along the Fall Zone in the Newark, Delaware area and in the Red Lion area. Those faults were investigated by the staff in considerable detail in regard to the Summit

Nuclear Power Station site studies. As the result of its review of data from these studies and based on advice from the USGS, the NRC staff concluded that the oldest unfaulted strata overlying any of these faults was at least 65 million years old. Therefore these faults are not considered capable within the meaning of the NRC seismic and geologic siting criteria, Appendix A to 10 CFR Part 100 and were not considered significant to the nuclear sites on Artificial Island, which includes Salem 1 and 2 and Hope Creek 1 and 2. Because consideration of these faults did not alter the original conclusion regarding the seismic safety of Salem 1 and 2, the staff did not address specific geologic anomalies in the Safety Evaluation Report for the operating license of Salem 1 and 2, but simply restated the original conclusion.

In regard to the fault down the middle of the Delaware River proposed by the University of Delaware, we assume that you are referring to one of the faults discussed in a 1976 article by Spoljaric and others, entitled "Inference of Tectonic Evolution from LANDSAT-1 Imagery," which was published in Photogrametric Engineering and Remote Sensing, Vol. 52, No. 8, pages 1069-1082. While this paper postdates publication of the Salem SER, we have considered it and reviewed its significance. The fault in the Delaware River discussed in the referenced article, which is based primarily on the interpretation of LANDSAT-1 imagery, is believed by the authors to be equivalent to one of the fault systems described by Spoljaric in 1972 and 1973. This fault system was investigated during the Summit site studies and shown to be at least 65 million years old. Based on the results of that investigation, we see no reason to change our conclusion arrived at during the CP review, that is, there are no known geologic structures that could tend to localize earthquakes in the site vicinity, and the SSE of 0.2g and the OBE of 0.1g are acceptable.

The NRC staff order regarding seismic inspection of 29 reactors was related to the specific area of the design of safety related piping (Office of Inspection and Enforcement Bulletin 79-07, "Seismic Stress Analysis of Safety-Related Piping") and does not include matters related to the seismic design of safety-related structures.

Our review of the seismic design of all Category I (safety-related) structures including the containment structure and fuel building is presented in Section 3.7 of the Safety Evaluation Report and Section 3.7 of Supplement No. 3 to the Safety Evaluation Report. As stated in Section 3.7 of Supplement No. 3, we require additional information regarding the seismic design as it relates to (1) a comparison of the response spectra and damping values between those currently adopted by us and those adopted by the applicants; (2) a justification of the use of a ± 10 percent peak width increment; (3) criteria used for the selection of lumped masses; and (4) criteria used for either coupling or decoupling a subsystem to its supporting system.

In letters dated January 21, 1979 and February 6, 1979, the applicants provided the necessary information. On the basis of our review of these matters, we concluded that the information provided was acceptable and consider the matter related to the seismic design of Category I structures resolved. We have not issued a supplement to the SER since that time. Our evaluation of these matters will be presented in a supplement to the Safety Evaluation Report prior to a decision concerning the issuance of an operating license.[18]

[18] See Supplement 4, Section 3.7, Safety Evaluation Report for Salem Nuclear Generating Station Unit 2, Docket No. 50-311, April , 1980.

With respect to the fuel handling system and spent fuel pool facilities, our evaluation of the seismic design of this system and facilities for Unit 1 is presented in a safety evaluation related to the Unit 1 modification of the Spent Fuel Storage Pool dated January 15, 1979. Since the Unit 2 fuel handling system and spent fuel pool facilities are identical to the Unit 1 fuel handling system and spent fuel pool facilities, we conclude that they are acceptable on the basis of the Unit 1 Evaluation.

With respect to the containment structural integrity test (CSIT) the Office of Inspection and Enforcement indicated in their report 50-311/78-51 that crack patterns were located on the exterior of containment utilizing the test procedure and Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments" requirements. The area of each crack pattern exceeded the Regulatory Guide 1.18 requirement of 40 square feet. A grid network of one foot squares was superimposed on the crack pattern area as an aid for transcribing crack details.

The inspection report (50-311/78-51) indicated that the applicants sandblasted the surface of the containment structure at the crack pattern areas to remove a coating of Modac in order to expose the actual concrete surface. During the inspector's observations of the crack pattern areas, it was noted that the sandblasting operations had weathered the edges of the existing cracks, thus making it somewhat difficult to obtain consistent crack width readings. The inspectors alerted test personnel of their concerns in this area. The inspectors further observed the initial crack survey at atmospheric pressure just prior to the start of pressurization and expressed their concern to the applicants about the methods being used to measure cracks. In response the applicants conducted additional training of each crack mapping team.

The report (50-311/78-51) further indicates that based on interviews, observations, and independent measurements performed, the inspectors felt that the test was conducted in accordance with the test procedure and that valid test data were obtained.

In the Office of Inspection and Enforcement Report 50-311/79-10, it is indicated that the inspector reviewed the test records relative to the containment structural integrity test. These final data were compared to the acceptance criteria stated in the Final Safety Analysis Report and also to the results of the Salem Unit 1 containment structural integrity test.

With respect to crack measurements, it is stated in the Office of Inspection and Enforcement Report that:

> "The FSAR limit of 0.030 inches maximum crack growth during pressurization from 0 psig to 54 psig was exceeded in a total of 39 instances. Of these 39 instances, 30 were in the area of the equipment hatch and 6 were in the area of the personnel hatch. The licensee elected to chip away approximately 1/8" of surface concrete in the area of 4 of these cracks to determine their depth. This was accomplished at the maximum test pressure of 54 psig. It was determined that in none of the 4 cracks explored did the crack width exceed 0.015 inches 1/8" deep into the concrete.
>
> The FSAR limit of 0.020 inches maximum residual crack growth after depressurization was exceeded in one instance in the area of the equipment hatch. The licensee intends to explore this crack.

The above results are not consistent with Unit 1. However, the FSAR stated acceptance criteria for the containment structure is "...demonstration that the overall structure exhibits elastic behavior throughout the test range." The licensee feels, based on the values of the other measurements and the results of the crack exploration, that the containment structure did demonstrate elastic behavior and that the crack growth data is anomalous. The displacement and strain measurement data examined by the inspectors appeared to support this conclusion. The licensee will offer an explanation for the crack growth in the final CSIT report. The licensee has stated that the CSIT test results will be submitted by letter to NRR with an evaluation of the cracks."

On April 24, 1979, the Public Service Electric and Gas Company submitted to the Office of Nuclear Reactor Regulation (NRR) the report "Structural Integrity Tests, Unit 2 Containment, Salem Nuclear Generating Station" dated February 26, 1979.

We have reviewed the information provided by the applicants and our evaluation regarding this matter is presented in the following paragraphs.

The applicants have tested the Salem 2 containment following the non-prototype test requirements identified in Regulatory Guide 1.18, with some expanded measurements which utilized already installed strain gauges. These additional measurements were used to correlate test results between Units 1 and 2.

The verification of the containment design was made by comparing the measured displacements and strains with the computed values. The applicants have shown

in the subject report that the measured displacements and strains were within the acceptable range of their corresponding computed values. The crack pattern and widths were shown to be in general agreement with the computed values. Some cracks exceeded the acceptable width limit. This problem was investigated by the applicants to determine whether these larger cracks might effect the structural integrity of the containment. It was determined that these cracks were shallow surface cracks which were not induced by excessive rebar strains.

Based on the review of the information contained in the subject report, we conclude that the applicants have adequately demonstrated that the concrete containment is capable of withstanding the postulated pressure loads with no adverse effects to its functional integrity and the subject test is judged acceptable.

Contention 10

The NRC has failed to require the licensee to consider, evaluate, and analyze the possible effects of a Class 9 accident for the Salem Nuclear Generating Station.

The term "Class 9 accident" derives from the Commission's December 1971, proposed rulemaking on "Consideration of Accidents in Implementation of the National Environmental Policy Act of 1969," 36 Fed. Reg. (1971). The proposed rulemaking would have added an Annex to Appendix D of 10 CFR Part 50, to set forth the manner in which various categories of accidents should be taken into account in the environmental review. In the proposed Annex, the

Commission divided into classes a theoretical spectrum of accidents ranging in severity from "trivial" (Class 1) to "very serious" (Class 9). Each class of accidents, except Classes 1 and 9, is required to be analyzed in environmental reports and statements. According to the proposed Annex, Class 1 accidents need not be considered because of their trivial consequences. Accidents within Classes 2 through 8 "found to have significant adverse environmental effects shall be evaluated as to probability, or frequency of occurrence, to permit estimates to be made of environmental risk or cost arising from accidents of the given class" 36 Fed. Reg. 22852 (1971). With regard to "Class 9" accidents, the proposed Annex states:

> "The occurrences in Class 9 involve sequences of postulated successive failure more severe than those postulated for the design basis for protective systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is so small that their environmental risk is extremely low. Defense in depth (multiple physical barriers), quality assurance for design, manufacture, and operation, continued surveillance and testing, and conservative design are all applied to provide and maintain the required high degree of assurance that potential accidents in this class are, and will remain, sufficiently remote in probability that the environmental risk is extremely low." 36 Fed. Reg. 2286 (1971).

Accordingly, the Annex does not require discussion of Class 9 accidents in environmental reports and statements.

Although the Annex has never been formally adopted by the Commission, the Commission noted upon publication that the Annex would be useful as "interim guidance" until the Commission took further action on the Annex. 36 Fed. Reg. 22851 (1971). Upon promulgation of 10 CFR Part 51 in 1974, the Commission stated that the adoption of Part 51 did not affect the proposed Annex, which was "still under consideration by the Commission." 39 Fed. Reg. 26279 (1974). Reliance on the Annex has been upheld by decisions of the Commission's adjudicatory panels and by Federal Courts. See Offshore Power Systems (Floating Nuclear Power Plants), CLI-79-9, 10 NRC 257, 259 n. 6 (1979), and cases cited therein; Pennsylvania Power & Light Company (Susquehanna Steam Electric Station Units 1 & 2), LBP-79-29, 10 NRC 586, 590 (1979).

The Colemans now request the Director of NRR to reverse the Commission's existing policy and to require the Public Service Electric & Gas Co., to consider the possible effects of a Class 9 accident for the Salem Unit 2 facility.[19] I do not find such a course of action appropriate in light of the Commission's expressed intention in its recent decision in Offshore Power Systems, supra. Although the Commission ruled that consideration of Class 9 accidents was

[19] The staff has previously indicated in a Director's Decision under 10 CFR 2.206 that the fact that the Three Mile Island accident occurred will not itself cause the staff to institute a proceeding to consider generally the environmental effects of Class 9 accidents at a facility. Public Service Company of Indiana (Marbel Hill Nuclear Generating Station Units 1 & 2), DD-79-21, 10 NRC 717 (Docket Nos. 50-546 & 50-547; November 27, 1979). Although at least two Licensing Board panels have acknowledged, consistent with the proposed Annex, the admissibility of "Class 9 contentions" involving a specific accident sequence based on the Three Mile Island accident, these same Boards have recognized that general consideration of the consequences of Class 9 accidents at land-based reactors would be inconsistent with Commission policy. Metropolitan Edison Co. (Three Mile Island Nuclear Station Unit 1), First Special Prehearing Conference Order (Restart Proceeding) Slip Op. at 11 (Docket No. 50-289, December 18, 1979); Pennsylvania Power 7 Light Co., supra, 10 NRC at 591.

proper in the environmental review of floating nuclear power plants, the Commission did not alter the status of the proposed Annex as the Commission's "interim guidance" pending completion of the rulemaking on the proposed Annex. Moreover, the Commission expressed its intent "to complete the rulemaking begun by the Annex and to re-examine Commission policy in this area." Id., Slip. Op. at 9. The Commission cautioned, however, that it was not "expressing any views on the question of environmental consideration of Class 9 accidents at land-based reactors," specifically noting that "[s]uch a generic action is more properly and effectively done through rulemaking proceedings in which all interested persons may participate." Id.[20] In the meantime, the Commission requested in Offshore Power Systems that the staff:

> "1. Provide us with its recommendations on how the interim guidance of the Annex might be modified, on an interim basis and until the rulemaking on this subject is completed, to reflect development since 1971 and to accord more fully with current staff policy in this area; and
>
> 2. In the interim, pending completion of the rulemaking on this subject, bring to our attention any individual cases in which it believes the environmental consequences of Class 9 accidents should be considered." 10 NRC at 262-63.

[20] The Commission reaffirmed this view in its recent decision Public Service Company of Oklahoma (Black Fox Station Units 1 & 2), CLI-80 (Docket Nos. 50-556 & 50-557, March 21, 1980).

The Commis on has under consideration a proposed revised policy and pending consideration and guidance by the Commission on the proposed policy, the staff plans to withhold completion of any unissued Environmental Impact Statements on cases under current review.

We believe that the course of developing a revised general NRC policy on reviewing the risks of nuclear accidents, taking into account the suggestions in the Lewis Committee Report and the lessons learned from the accident at Three Mile Island, will in the long run result in sounder reviews than if we attempted to supplement reviews for individual plants before the general policy was determined. The Commission itself has said, "[W]e did not believe that the NRC's generic policy on consideration of Class 9 accidents would properly be developed ruling on a case-by-case basis. Such piecemeal consideration is not appropriate to such an important policy area, and we decline to adopt such an approach now." Public Service Company of Oklahoma (Black Fox Station Units 1 & 2), CLI-80-8, Docket Nos. 50-556 and 50-557, Slip. Op. at 3-4 (March 21, 1980).

The staff is mindful, however, that the Commission also requested the staff to bring to the Commission's attention "any individual cases in which it believes the environmental consequences of Class 9 accidents should be considered." Off Shore Power Systems, supra. 10 NRC at 263. See also Public Service Company of Oklahoma supra, at 3 & n.3. The staff has reviewed information concerning the Salem facility to determine whether "special circumstances"

exist which might warrant a detailed Class 9 accident evaluation.[21] The results of the staff's review follow:

As noted in Section 1.2 of the Safety Evaluation Report,[22] the nuclear steam supply system for each Salem unit will consist of a pressurized water reactor using a four-loop reactor coolant system. The Salem facility is a typical light water reactor facility similar to several other reactor designs of the Westinghouse Electric Corporation licensed for construction and operations, and therefore is not a novel reactor design.

In Offshore Power Systems, the unique design and unique siting mode consisted of a nuclear power plant mounted on a floating barge. There would be no soil

[21] The staff's review is similar to one undertaken in a recent decision under 10 CFR 2.206, in which the staff reviewed information concerning the Seabrook Station in light of the special circumstances identified in the staff's brief to the Commission (dated January 12, 1979) in Offshore Power Systems. Public Service Co. of New Hampshire (Seabrook Station, Units 1 & 2), DD-80-6, Docket Nos. 50-443 & 50-444, Slip. Op. at 10-12 (Feb. 11, 1980). In the brief submitted in Offshore Power Systems, the staff listed three special circumstances:

> "To date, only three types of special circumstances have been identified that would trigger a detailed Class 9 accident evaluation: a high population density for the proposed site (above the trip points in the Standard Review Plan and Regulatory Guide), a novel reactor design (a type of power reactor other than a light water power reactor), or a combination of a unique design and a unique siting mode (a floating nuclear plant)." Brief at 47.

See also Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, NUREG-75-087 (Sept. 1975); General Site Suitability Criteria for Nuclear Power Stations, Regulatory Guide 4.7 (Nov. 1975). In Public Service Co. of Oklahoma, supra, at 3, the Commission noted in addition to these three criteria that proximity of a plant to a "man-made or natural hazard" might also represent "the type of exceptional case that might warrant additional consideration."

[22] "Safety Evaluation of the Salem Nuclear Generating Station Units 1 and 2, Docket Nos. 50-272 and 50-311", U.S. Atomic Energy Commission, Directorate of Licensing, October 11, 1974.

structure to retard the release and dispersal of activity beneath the plant following a core melt accident as would be the case for land based plants. The staff concluded that the most likely population exposure from the liquid pathway for a floating nuclear plant is significantly greater than for a land based plant because of the inability to interdict releases in the vicinity of the floating nuclear plant.

The Salem Nuclear Generating Station is located on Artificial Island, a man-made peninsula in the Delaware River estuary in Salem County, New Jersey. This estuary is the nearest surface water body which could be affected by liquid release from a Class 9 accident.

The most likely groundwater path to the estuary would be through a permeable sand layer approximately 30 feet below the surface. Groundwater velocity is conservatively estimated to be about 3 feet per day.

The time for contaminated liquids, generated by a postulated core-melt accident, to travel the groundwater pathway (approximately 780 feet) to the estuary would be in excess of 8 months. Due to this slow rate of groundwater movement, the staff concludes that there are no unusual features or special circumstances with regard to the groundwater contamination interdiction characteristics of this site that would distinguish it from other land based light water reactor sites to the extent that, under the present Commission policy, would warrant consideration of environmental consequences of Class 9 accidents.

However, the Task Action Plans contained in Draft NUREG-0660 (TMI Lessons Learned) as proposed to the Commission, identify Task Action Plan III.D.2.3, liquid pathway interdiction (an in-depth study of one of the special factors of Class 9 events). Assuming approval of this plan, Salem and all other plants would be analyzed as part of Task Action Plan III.D.2.3. If that should result in the liquid pathway being identified as a unique consideration at Salem, and the Commission's interim policy on Class 9 accident consideration has not yet clarified the situation in this regard, methods of interdiction and mitigation will be identified. Based upon the Liquid Pathway Study and preliminary discussions with Argonne National Laboratory on liquid pathway mitigation methods, it is possible to interdict within the time period identified above and reduce or prevent the migration of contaminated groundwater to the river.

Several methods of mitigation, including pumping and construction of slurry walls to prevent migration are available. However, site specific techniques if required, will be identified as a part of the Liquid Pathway Interdiction review.

On the question of proximity to man-made or natural hazards, the Staff concluded, in Section 2.2 of Supplement 1 to the Safety Evaluation Report, on the basis of its analysis of site characteristics, that the site was acceptable relative

to seismology, geology and foundations.[23] The Staff also concluded that the probability of damage to safety-related portions of the nuclear power plants on Artificial Island due to accidents occurring to waterborn commerce involving hazardous cargoes on the Delaware River was sufficiently low that these accidents need not be the bases for the design of the Salem facility.[24] In a recent decision which considered the issue of hazards due to water traffic on the Delaware River for the Hope Creek facilities, which are also located on Artificial Island, the Appeal Board concluded that there need not be any modification to the design of those facilities to accommodate possible hazards.[25] Moreover, conditions were included in the construction permits for the Hope Creek facilities which require reports to the Commission periodically of any changes in actual or projected traffic on the Delaware River.[26] These reporting requirements will provide information directly relevant to Salem Unit 2 and will keep the Commission informed of any changes that might affect the above conclusions.

With regard to the high population special circumstance, the staff's brief in Offshore Power Systems noted that the "special attention" called for by the

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[23] Supplement 1, Safety Evaluation Report, Salem Nuclear Generating Station, Docket No. 50-311, June 28, 1976, Section 2.5. See also the response to Contention 9 herein.

[24] Supplement 1, Safety Evaluation Report, Salem Nuclear Generating Station Docket No. 50-311, June 28, 1976, Section 2.2.

[25] Public Service Electric and Gas Company, et al. (Hope Creek Generating Station, Units 1 and 2), ALAB-518, 9 NRC 14, 37 (1979).

[26] Id. at 39-40

Standard Review Plan and Regulatory Guide 4.7,[27] in the case of sites exceeding the population level "trip points" entails a consideration of comparative population exposures for Class 9 accidents at the proposed site and alternative sites. The "trip points" apply to proposed new sites at the construction permit stage and were not evaluated nor proposed for plants beyond the construction permit stage. The consideration of population exposures for Class 9 accidents has been utilized by the staff in assessing the relative differences between a proposed site and candidate alternative sites. The consideration of population exposure for Class 9 accidents is not used as an absolute site-specific criterion for evaluating the suitability of a proposed site and sites are not necessarily found unsuitable if they exceed the population density guidelines given in the Standard Review Plan and Regulatory Guide 4.7. As indicated by the staff criteria in Regulatory Guide 4.7 and described in the Pilgrim final environmental statement,[28] a site that exceeds the population density guidelines can nevertheless be selected and approved if, on balance, it offers advantages compared with available alternative sites when all of the environmental, safety, and economic aspects of the proposed site and the alternative sites are considered.

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[27] Section C.3 of Regulatory Guide states:

> "If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square miles averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with lower population densities."

[28] "Final Supplement to the Final Environmental Statement related to construction of Pilgrim Nuclear Power Station Unit No. 2," (NUREG-549) May 1979.

It is current staff practice to assess the relative differences in population exposures from a Class 9 accident at a proposed new site and the alternative sites, using population distribution and population density as a surrogate for accident consequences. The consequences of radiological accidents, from minor or trivial releases up to and including severe events, is directly related to the number of people surrounding a particular site and to the distance of the population from the reactor location. The staff recognizes that the population distribution of a site is a relatively crude measure of the risk associated with the accidental releases of radioactivity. The risk from any accidental releases would depend not only upon the population distribution of a site but also upon many other factors that would enter into the determination of the actual consequences of the accident. However, insight gained in the evaluation of the relative consequences of accidents in the Perryman alternative site study (SECY-78-137, Enclosure D) led the staff to conclude that (1) the relative differences in the population distribution between sites is a reasonable measure of the relative magnitude of potential consequences, (2) relatively large differences in the population densities between two sites are required to exist before significant differences in accident risks would be expected to be discernible, and (3) the risk is not uniform for all members of the population regardless of distance from the site but is higher for those persons relatively close to the site and generally decreases with distance away from the site.

The 1970 population density in the vicinity of the Salem site was less than 100 persons per square mile within 10 miles, and was about 320 persons per square mile at 30 miles. Population projections for the year 2000 indicate

that the population density is expected to increase to about 130 persons per square mile within 10 miles, and to a value of about 450 persons per square mile at 30 miles. Based on these data, it is clear that the population density in the vicinity of the Salem site does not exceed the population level "trip points" of Regulatory Guide 4.7, and that the population density cannot, therefore, be considered to be a special circumstance that would trigger a detailed Class 9 accident evaluation.

CONCLUSION

Based on the foregoing analysis of the Contentions, I have determined that each has been adequately resolved. Consequently, the Colemans' request for a stay of the issuance of the operating license for Salem Nuclear Generating Station Unit 2, is denied.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555 and in the local Public Document Rooms for the Salem Unit 2 facility located at Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079. A copy of this decision will also be filed with the Secretary for review by the Commission in accordance with 10 CFR 2.206(c). of the regulations of the Commission.

As provided in 10 CFR 2.206(c), this decision will constitute the final action of the Commission twenty (20) days after the date of issuance, unless the Commission on its own motion institutes the review of this decision within that time.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 16th day of April, 1980