RELATED CORRESPONDENCE

March 24, 1983

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PUBLIC SERVICE COMPANY OF	)	Docket Nos. 50-443 OL
NEW HAMPSHIRE, et al.	)	50-444 OL
	)	
(Seabrook Station, Units 1 and 2)	)	
	)	

NECNP OPPOSITION TO MOTIONS FOR SUMMARY DISPOSITION
AND NOTIFICATION OF WITHDRAWN CONTENTIONS

Applicants and the NRC Staff have filed motions for summary disposition pursuant to 10 CFR § 2.749 on 21 of NECNP's contentions in this proceeding. With the exception of the motions on emergency planning issues, they are answered below. NECNP has dropped a number of contentions because the discovery process has satisfied our safety concerns. NECNP opposes, however, summary disposition of Contentions I.A.2, I.B.1, I.D.2, I.G, and II.B. 3 and 4. Neither Applicants nor Staff have met the burden of proof that they must carry for summary disposition. 10 CFR § 2.732. For each of these issues, there remain unresolved issues of fact or law, and therefore summary judgment may not be granted. We note that where "significant health and safety issues" are involved, motions for summary disposition should be granted only if the licensing board "is convinced from the material filed that the public health and

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safety ... will be satisfactorily protected." Cincinnati Gas & Electric (William H. Zimmer Nuclear Station) LBP-81-2, 13 NRC 36, 40 (1981).

I.A.2. Qualification of Electric Valve Operators

Applicants have moved for summary disposition of NECNP Contention I.A.2, which asserts that "the Applicant has not complied with Commission standards regarding qualification tests of electric valve operators installed inside the containment." NECNP Supplemental Petition to Intervene at 8 (filed April 21, 1982).

In support of their motion, Applicants assert that "All Class 1E electric valve operators installed inside containment will be qualified in conformity with Reg. Guide 1.73 (Rev. 0, 1/74); Reg. Guide 1.89 (Rev. 0, 11/74) and NUREG-0588. NECNP does not dispute this factual assertion. The Applicants' error is legal, not factual. Applicants are required to environmentally qualify not only those electric valve operators which are "safety related" or "Class 1E," but all electric valve operators which are "important to safety." 10 CFR § 50.49(a). This includes non-safety related valve operators "whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions...by the safety-related equipment." 10 CFR § 50.49(b)(2).

As shown by Applicants' Table I.A.2-3 (attached) the only classifications considered by Applicants in qualifying electric valve operators are "safety related" and "Class 1E", a subset of safety related. There are a number of electric valve operators inside the containment which are not qualified because they are not "safety related" or "Class 1E". Applicants have not made the required determination, however, of whether they are "important to safety." The analysis performed by Applicants in determining which electric valve operators must be qualified consists only of a determination regarding which valve operators are <u>required</u> to operate during an accident. Applicants have not considered whether failure of any of the other electric valve operators could prevent satisfactory completion of safety functions. Applicants have not met the legal requirements for environmental qualification of equipment important to safety, and therefore summary disposition must be denied.

STATEMENT OF FACTS AS TO WHICH THERE IS A
MATERIAL ISSUE TO BE HEARD

1. NRC regulations at 10 CFR § 50.49 require the environmental qualification of all electric equipment important to safety.

2. The electric valve operators inside the containment include valve operators which are neither classified as "safety related" nor "Class 1E." See Table I.A.2-3

3. Applicants have only environmentally qualified electric valve operators which they consider to be "safety related" or "Class 1E".

4. With regard to the other electric valve operators, Applicants have made no determination as to whether they are nonsafety related components which are nevertheless important to safety because their failure could prevent the operation of safety functions.

I.B.1 Qualification of Residual Heat Removal Equipment

Applicants and the NRC Staff have moved for summary disposition on NECNP Contention I.B.1, which asserts that Applicants have violated GDC 4 and GDC 34 in that they have not environmentally qualified all systems that may be required to remove heat from the steam generators during an accident, including steam dump valves, turbine valves, and the steam dump control system. Applicants state, in support of their motion for summary disposition, that the Seabrook RHR system does not require the use of the steam dump valves, turbine valves, and the steam dump control system to meet the RHR requirements of GDC 34; and that the systems at Seabrook that are essential to perform or support the function of RHR are safety grade and environmentally qualified. The NRC Staff also asserts that these systems need not be qualified because they are "not required" to remove residual heat from the core during an accident.

NECNP does not dispute the facts asserted by Applicants and Staff. Those facts support a conclusion that Applicants and Staff have misapplied the NRC regulations.

The qualification of "essential" or "required" systems alone does not satisfy the Commission's requirements for

environmental qualification, and therefore does not meet GDC 4.[1/] Under 10 CFR § 50.49, Applicants must qualify all electric equipment "important to safety," which includes not only required safety related equipment, but "non safety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions ... by safety-related equipment." 10 CFR § 50.49(b)(2). The Applicants have not addressed the issue of whether heat removal systems such as the turbine valves, steam dump valves, or steam dump control system, are "important to safety." They are therefore not entitled to summary judgment. The NRC Staff, in a footnote, states that the components in question do not constitute nonsafety-related equipment which is "important to safety," but provide absolutely no support for that assertion. The Staff's conclusion is neither repeated nor explained in the "Statement of Facts as to Which There is No Genuine Issue to be Heard" or the affidavit supporting the

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[1/] In the absence of compliance with GDC 4, Applicants cannot provide a reliable heat removal system, because unqualified components which are important to safety must be assumed to fail. Therefore Applicants violate GDC 34.

motion.[2/] Without any factual support for this assertion, the NRC Staff has not sustained its burden of proof, and summary disposition must be denied.

Furthermore, the Staff's Safety Evaluation Report strongly supports a conclusion that systems which remove decay heat during normal operation, such as the turbine valves, steam dump valves, steam dump control system, main feedwater and condensate systems, condenser steam dump valves, condenser, and circulating water system, are important to safety and should be environmentally qualified. In asserting that these systems are not required for decay heat removal during an accident, the Applicants and Staff implicitly assume the integrity of the steam generators. The SER, however, recognizes the strong possibility of tube leaks in the steam generators, a recurrent problem with Westinghouse equipment. With regard to the integrity of the steam generator tubes, Unresolved Safety Issue

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[2/] In fact, in its answers to NECNP's interrogatories, the NRC Staff stated that "systems that perform the function of residual heat removal during all plant operating conditions are ...important to safety." (emphasis added) NRC Staff Response to NECNP First Set of Interrogatories at 11 (filed December 28, 1982). Those systems include the turbine valves, main feedwater and condensate systems, condenser steam dump valves, condenser, and circulating water system, which are normally used during orderly plant shutdown to remove decay heat. See NRC Staff Response to NECNP First Set of Interrogatories at 8. At the time, the NRC Staff defined "important to safety" as "structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." NRC Staff Response to NECNP Second Set of Interrogatories at 2 (filed January 21, 1983). The NRC Staff has not explained why it has apparantly changed its position, and no longer considers those systems to be important to safety.

A-3, the Staff was unable to find that "this facility can be operated before the ultimate resolution of these generic issues without endangering the health and safety of the public." Seabrook SER at C-7. Such leaks could allow radioactive primary coolant to escape into the secondary coolant system.

Because the steam generators cannot be relied upon as a barrier to the escape of radiation, other barriers which might ordinarily be considered secondary and unnecessary gain importance. Those barriers include the valves and systems used to remove heat from the steam generators during normal operation. In the presence of tube leaks in the steam generators, failure of these valves and systems could result in releases of radiation to the environment. For example, with leaking steam generators, Applicants could not rely on the atmospheric dump valves to remove heat from the steam generators during an accident, because the valves would release radiation into the environment. The Applicants would be forced to rely on other systems such as the condenser steam dump valves, which would channel the steam to the condensers instead of the atmosphere. If those valves failed, there would be no safe outlet for the decay heat from the steam generators.

NECNP has given an example of a likely situation in which equipment currently classified as nonsafety related, and currently unqualified, could be relied upon during an accident. It is not NECNP's burden to prove that this equipment must be environmentally qualified. Rather, it is the

Applicants' and Staff's burden to prove that qualification is not required because the equipment is not "important to safety." Applicants' and Staff's motions give no indication that the effect of these components' failure on the ability of safety equipment to function was ever considered. Because they have misapplied the legal standard, they have not satisfied their burden of proof, and summary disposition of Contention I.B.1 must be denied to Applicants and Staff. NECNP has attached the affidavit of Gregory C. Minor in support of this opposition.

STATEMENT OF MATERIAL FACT AS TO WHICH THERE
IS A GENUINE ISSUE TO BE HEARD

1. Applicants have only environmentally qualified those parts of the steam generator decay heat removal system which they consider necessary to remove heat from the steam generators during an accident.

2. Applicants have not performed any analysis to determine which of the nonsafety-related decay heat removal systems ae "important to safety" in that their failure would prevent operation of safety related functions.

3. In the Safety Evaluation Report for Seabrook, the NRC Staff has been unable to make a finding of safe operation pending resolution of Task A-3, Steam Generator Tube Degradation. Therefore, tube degradation, which has historically plagued Westinghouse steam generators, can be reasonbly expected to occur at Seabrook.

4. Where steam generator integrity is not assured, it can reasonably be expected that radioactivity will escape from the primary coolant into the secondary coolant. Therefore, all valves and systems which serve as a barrier between the steam generators and the environment must be classified as "important to safety" and enviromentally qualified because their failure to operate properly could result in the escape of radioactivity to the environment. 10 CFR 50.49(a). These valves and systems include the steam dump valves, turbine valves, condenser steam dump valves, and other valves on lines leading from te main steam line; control systems for valves; the condensate steam system; and the coolant system related to the condensate system. See Minor affidavit, paragraphs 4 and 5.

5. During an accident, in the event that the atmospheric dump valves could not be used to vent steam to the atmosphere because of radioactivity in the steam, Applicants would be forced to channel steam to the condensers via the turbine valves and condensate system and related cooling systems. Failure of these components could prevent the safe removal of heat from the steam generators. Therefore, they are important to safety under the definition of 10 CFR § 50.49(b)(2) and must be environmentally qualified. See Minor affidavit, paragraph 6.

AFFIDAVIT OF GREGORY C. MINOR

Gregory C. Minor, being duly sworn, deposes and says:

1. I am a consulting engineer with MHB Technical Associates in San Jose, California.

2. I have worked for over 20 years in the nuclear industry in a variety of positions including the design, manufacturing, construction, maintenance and analysis of nuclear plants, systems and components. A list of my professional qualifications is attached to this affidavit.

3. In the presence of steam generator tube leaks, there is an increased risk of radiation release to the environment through the secondary cooling system of a nuclear plant.

4. Where there is uncertainty regarding the integrity of the steam generators, all valves which constitute a barrier between the steam generators and the environment may be relied upon to prevent release of radioactivity to the environment. These valves include the steam dump valves, turbine valves, condenser steam dump valves, and other valves on lines leading from the main steam line.

5. The other systems which may be required to function to prevent release of radiation include control systems for valves, the condensate system where steam is directed, and the coolant system related to the condensate system.

6. In the event of an accident where there were tube ruptures in the steam generators, the atmospheric dump valves could not be used because they would release radioactivity to the environment. In such a case, Applicants would be forced to rely on the turbine valves and condensate system and related cooling systems to divert the steam to the condenser where its heat could be removed without release of radioactivity to the enviroment. Failure of these components could prevent the safe removal of heat from the steam generators.

Gregory C Minor
Gregory C. Minor

Subscribed and sworn to before me this 24th day of March, 1983.

Donna Marie Wilcox
NOTARY PUBLIC

My Commission Expires July [illegible]

PROFESSIONAL QUALIFICATIONS OF GREGORY C. MINOR

GREGORY C. MINOR
MHB Technical Associates
1723 Hamilton Avenue
Suite K
San Jose, California 95125
(408) 266-2716

EXPERIENCE:

1976 - PRESENT

Vice-President - MHB Technical Associates, San Jose, California. Engineering and energy consultant to state, federal, and private organizations and individusals. Major activities include studies of safety and risk involved in energy generation, providing technical consulting to legislative, regulatory, public and private groups and expert witness in behalf of state organizations and citizens' groups. Was co-editor of a critique of the Reactor Safety Study (WASH-1400) for the Union of Concerned Scientists and co-author of a risk analysis of Swedish reactors for the Swedish Energy Commission. Served on the Peer Review Group of the NRC/TMI Special Inquiry Group (Rogovin Committee). Actively involved in the Nuclear Power Plant standards Committee work for the Instrument Society of America (ISA).

1972 - 1976

Manager, Advanced Control and Instrumentation Engineering, General Electric Company, Nuclear Energy Division, San Jose, California.

Managed a design and development group of thirty-four engineers and support personnel designing systems for use in the measurement, control and operation of nuclear reactors. Involved coordination with other reactor design organizations, the Nuclear Regulatory Commission, and customers, both overseas and domestic. Responsibilities included coordinating and managing the design and development of control systems, safety systems, and new control concepts for use on the next generation of reactors. The position included responsibility for standards applicable to control and instrumentation, as well as the design of short-term solutions to field problems. The disciplines involved included electrical and mechanical engineering, seismic design and process computer control/programming.

1970 - 1972

Manager, Reactor Control Systems Design, General Electric Company, Nuclear Energy Division, San Jose, California.

Managed a group of seven engineers and two support personnel in the design and preparation of the detailed system drawings and control documents relating to safety and emergency systems for nuclear reactors. Responsibility required coordination with other design organizations and interaction with the customer's engineering personnel, as well as regulatory personnel.

1963 - 1970

Design Engineer, General Electric Company, Nuclear Energy Division, San Jose, California.

Responsible for the design of specific control and instrumentation systems for nuclear reactors. Lead design responsibility for various subsystems of instrumentation used to measure neutron flux in the reactor during startup and intermediate power operation. Performed lead system design function in the design of a major system for measuring the power generated in nuclear reactors. Other responsibilities included on-site checkout and testing of a complete reactor control system at an experimental reactor in the Southwest. Received patent for Nuclear Power Monitoring System.

1960 - 1963

Advanced Engineering Program, General Electric Company; Assignments in Washington, California, and Arizona.

Rotating assignments in a variety of disciplines:

- Engineer, reactor maintenance and instrument design, KE and D reactors, Hanford, Washington, circuit design and equipment maintenance coordination.

- Design engineer, Microwave Department, Palo Alto, California. Worked on design of cavity couplers for TWT's.

- Design engineer, Computer Department, Phoenix, Arizona. Design of core driving circuitry.

- Design engineer, Atomic Power Equipment Department, San Jose, California. Circuit design and analysis.

- Design engineer, Space Systems Department, Santa Barbara, California. Prepared control portion of satellite proposal.

- Technical Staff - Technical Military Planning Operation. (TEMPO), Santa Barbara, California. Prepare analysis of missile exchanges.

During this period, completed three-year General Electric program of extensive education in advanced engineering principles of higher mathematics, probability and analysis. Also completed courses in Kepner-Tregoe, Effective Presentation, Management Training Program, and various technical seminars.

EDUCATION

University of California at Berkeley, BSEE, 1960.

Advanced Course in Engineering - three-year curriculum, General Electric Company, 1963.

Stanford University, MSEE, 1966.

HONORS AND ASSOCIATIONS

- Tau Beta Pi Engineering Honorary Society.
- Co-holder of U.S. Patent No. 3,565-760, "Nuclear Reactor Power Monitoring System," February, 1971.
- Member: American Association for Advance of Science.
- Member: Nuclear Power Plant Standards Committee, Instrument Society of America.

PERSONAL DATA

Born: June 7, 1937
Married, three children
Residence: San Jose, California

PUBLICATIONS AND TESTIMONY

1. G.C. Minor, S.E. Moore, "Control Rod Signal Multiplexing," IEEE Transactions on Nuclear Science, Vol. NS-19, February, 1972.

2. G.C. Minor, W.G. Milam, "An Integrated Control Room System for a Nuclear Power Plant," NEDO-10658, presented at International Nuclear Industries Fair and Technical Meetings, October, 1972, Basle, Switzerland.

3. The above article was also published in the German Technical Magazine, NT, March, 1973.

4. Testimony of G.C. Minor, D.G. Bridenbaugh, and R.B. Hubbard before the Joint Committee on Atomic Energy, Hearings held February 18, 1976, and published by the Union of Concerned Scientists, Cambridge, Massachusetts.

5. Testimony of G.C. Minor, D.G. Bridenbaugh, and R.B. Hubbard before the California State Assembly Committee on Resources, Land Use, and Energy, March 8, 1976.

6. Testimony of G.C. Minor and R.B. Hubbard before the California State Senate Committee on Public Utilities, Transit, and Energy, March 23, 1976.

7. Testimony of G.C. Minor regarding the Grafenrheinfeld Nuclear Plant, March 16-17, 1977, Wurzburg, Germany.

8. Testimony of G.C. Minor before the Cluff Lake Board of Inquiry, Regina, Saskatchewan, Canada, September 21, 1977.

9. <u>The Risks of Nuclear Power Reactors: A Review of the NRC Reactor Safety Study WASH-1400 (NUREG-75/0140)</u>, H. Kendall, et al, edited by G.C. Minor and R.B. Hubbard for the Union of Concerned Scientists, August, 1977.

10. <u>Swedish Reactor Safety Study: Barsebäck Risk Assessment</u>, MHB Technical Associates, January, 1978. (Published by Swedish Department of Industry as Document SdI 1978:1)

11. Testimony by G.C. Minor before the Wisconsin Public Service Commission, February 13, 1978, <u>Loss of Coolant Accidents: Their Probability and Consequence</u>.

12. Testimony by G.C. Minor before the California Legislature Assembly Committee on Resources, Land Use, and Energy, AB 3108, April 26, 1978, Sacramento, California.

PUBLICATIONS AND TESTIMONY

13. Presentation by G.C. Minor before the Federal Ministry for Research and Technology (BMFT), Meeting on Reactor Safety Research, Man/Machine Interface in Nuclear Reactors, August 21, and September 1, 1978, Bonn, Germany.

14. Testimony by G.C. Minor, D.G. Bridenbaugh, and R.B. Hubbard, before the Atomic Safety and Licensing Board, September 25, 1978, in the matter of the Black Fox Nuclear Power Station Construction Permit Hearings, Tulsa, Oklahoma.

15. Testimony of G.C. Minor, ASLB Hearings Related to TMI-2 Accident, Rancho Seco Power Plant, on behalf of Friends of the Earth, September 13, 1979.

16. Testimony of G.C. Minor before the Michigan State Legislature, Special Joint Committee on Nuclear Energy, Implications of Three Mile Island Accident for Nuclear Power Plants in Michigan, 10/15/79.

17. A Critical View of Reactor Safety, by G.C. Minor, paper presented to the American Association for the Advancement of Science, Symposium on Nuclear Reactor Safety, January 7, 1980, San Francisco, California.

18. The Effects of Aging on Safety of Nuclear Power Plants, paper presented at Forum on Swedish Nuclear Referendum, Stockholm, Sweden, March 1, 1980.

19. Minnesota Nuclear Plants Gaseous Emissions Study, MHB Technical Associates, September, 1980, prepared for the Minnesota Pollution Control Agency, Roseville, MN.

20. Testimony of G.C. Minor and D.G. Bridenbaugh before the New York State Public Service Commission, Shoreham Nuclear Plant Construction Schedule, in the matter of Long Island Lighting Company Temporary Rate Case, September 22, 1980.

21. Testimony of G.C. Minor and D.G. Bridenbaugh before the New Jersey Board of Public Utilities, Oyster Creek 1980 Refueling Outage Investigation, in the matter of Jersey Central Power and Light Rate Case, February 19, 1981.

I.C. NECNP has withdrawn this contention.

I.D.1. NECNP has withdrawn this contention.

I.D.2. Testing of Protection System Actuation Functions

Applicants have moved for summary disposition of this contention on the ground that Reg. Guide 1.22 has been fully complied with, and that therefore General Design Criterion 21 has been satisfied.[3/] NECNP contends that, with respect to testing of the manual system actuation functions, Applicants have complied with neither GDC 21 nor Reg. Guide 1.22, and that there is a factual dispute on the reliability of the manual trip function which cannot be resolved in favor of Applicants by summary judgment.

General Design Criterion 20 requires that the protection system be designed to initiate the operation of systems and components important to safety. General Design Criterion 21 requires that the protection system be designed to permit periodic testing of its functioning when the reactor is in operation. Reg. Guide 1.22, which implements these design criteria, contains three narrow criteria for waiving the requirement for testing of power. Regulatory Position 4, page 22.2.

3/ Applicants also argue that NUREG-0737 does not apply to this contention. NECNP does not dispute this assertion.

The nuclear industry's history of numerous reactor trip failures, culminating in the near-disatrous Salem accident of February 25, demonstrates that Applicaants cannot provide the assurance required by Regulatory Position 4.b. that "the probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the equipment during reactor operation." The Salem accident demonstrated the critical role of the manual trip function in averting a serious accident. When both channes of the automatic trip function failed, the operator was required to immediately activate the manual trip in order to sut the reactor down. The February 25 incident was preceded by a similar event on February 22. The two Salem accidents were the first events in which both circuit breakers failed; however, there have been 35 reported circuit breaker failures since 1973. Philadelphis Enquirer, March 3, 1983 at 1-B. The NRC has also reported reactor trip breaker failure in two tests at San Onofre during the month of March. See IE Bulletin No. 83-04.

As a result of the Salem incident, the NRC is requiring the licensee to test the manual trip (or shunt trip) every month. Because normal operational outages typically occur at much longer intervals, this means the manual trip functions must be tested at power.[4/]

4/ The fact that testing can be done at monthly intervals (the licensee actually proposed weekly intervals) indicates that it is possible to test the manual reactor trip without damaging the reactor, contrary to Applicants' unsupported assertion in the FSAR. at § 7.1.2.5.

NECNP contends that in light of the Salem accident and the other reactor trip failures which show a low reliability of that function, the frequent testing of the manual trip function at power is required for conformance with GDC 20 and 21, and Reg. Guide 1.22. Therefore, Applicants' Motion for summary disposition should be denied.

CONTENTION I.D.2
STATEMENT OF MATERIAL FACTS AS TO WHICH THERE
IS A GENUINE ISSUE TO BE HEARD

1. General Design Criterion 21 requires that "The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety function to be performed." GDC also requires that "The Protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundance that may have occured."

2. Applicants do not provide for testing at power of the manual reactor trip function.

3. The unreliability of automatic reactor trip has been demonstrated by a large number of trip failures in recent years. See IE Bulletin 83-04 and Philadelphia Enquirer of March 3, 1983, attached. The most significant of these events was the failure of both atomatic reactor trip channels to initiate shutdown at the Salem reactor on February 22 and February 25, 1983. In those instances, the Salem operator relied on the manual trip switch to bring the plant to safe shutdown. Had the plant been operating at full power, failure of the manual trip could have caused a major accident.

4. In order to insure reliability of reactor trip, the NRC has required the institution of monthly tests of the manual trip at Salem. SECY 83-98A, Attachment 1 at 27. Since normal outages occur at much greater intervals than monthly, it may be assumed that the manual trip must be tested at power under these circumstances.

5. Aside from an unsupported assertion in the FSAR that testing the manual reactor trip at power would damage the reactor, Applicants have never given any reason for failing to test the manual trip function at power. Applicants have not stated that they are unable to test each channel of the manual trip separately, which would avoid tripping the reactor.

6. Commensurate with the requirements set forth by the Staff for the Salem plant, the Applicants should be requied to perform periodic testing of the manual trip switch at power in order to provide a reasonable assurance of its reliability in satisfaction of GDC 21.

March 14, 1983 SECY-83-98A

POLICY ISSUE
(Information)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: SALEM RESTART

Purpose: To provide the Commissioners with a report on the current status of the staff evaluation of the failure to automatically scram events of February 22 and 25, 1983 at the Salem Nuclear Generating Station and the staff action plan for authorizing restart of Units 1 and 2.

Discussion: During a briefing on March 2, 1983 concerning the Salem reactor trip system failure events, the Commissioners requested that the staff provide its plan of action to resolve the issues identified from the NRC evaluation of the Salem events.

Enclosed is the Salem Restart status report which identifies the issues related to the recent Salem events and the short- and long-term actions needed to resolve those issues. For the short-term actions, the staff has or intends to obtain specific commitments from the licensee to complete those actions and the staff will assure their satisfactory completion prior to permitting restart of either Salem unit. For satisfactory resolution of the long-term actions, the staff intends to develop with the licensee an acceptable schedule for completion of those actions, obtain necessary written commitments, and follow up their completion on the agreed upon schedule.

Contact: Gus Lainas
X-27817
R. Starostecki
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In addition to the short- and long-term actions identified in the report, the staff has also concluded that a show cause order should be issued to the licensee (see enclosure 2). The staff believes that the particular circumstances at this facility, as further detailed in the start-up report, justify requiring that these three separate but interrelated sets of actions be implemented by the licensee in a timely fashion.

Subject to satisfactory implementation of these actions, the staff has concluded that the Salem facilities can be restarted and operated without undue risk to the health and safety of the public. Enforcement actions are under active consideration by the staff and will be discussed separately with the Commission at a later date.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Salem Restart Status Report
2. Dft Show Cause Order

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technician. At this time, the CRC-2-26 lubricant cleaner was sprayed on all four UV trip attachments associated with the Unit 1 circuit breaker. This lubricant is being procured by FRC for testing purposes.

List of Investigations To Be Performed by NRC Contractor (FRC)

1. The first test will be to perform various deenergizations and energizations of the UV trip attachment and monitor the device under various conditions.

2. The second test will be to disassemble the latch mechanism to observe the surfaces of the various parts of the latch and to photograph these surfaces through a microscope to determine the various levels of wear on these surfaces.

3. The third test is to determine the effects of CRC-2-26 spray on the various types of metals used in this devices. An attempt will be made to use metals other than those in the actual attachment. If possible, the chemical consistency of this spray will be determined from the manufacturer.

To prove that the sample UV trip attachment is identical to all such Salem devices, a visual inspection of all existing Salem Unit 1 and 2 UV trip attachments will be performed. This can take place at Salem, with no disassembly needed. The inspection can be made with the devices mounted on the circuit breakers or loose. These inspections should be done as soon as possible, and Tuesday, March 8, 1983 is recommended.

If further tests are required they will be based on the results of these initial tests. All tests will be nondestructive such that the device can be used for further testing and returned to the utility.

Additional Test To Be Conducted by the Licensee, as Revised by NRC Staff

This test will require the use of a spare circuit breaker. The UV trip and shunt trip attachments will be mounted on the breaker, and the breaker will be operated repeatedly to determine the effect on the shunt and UV trip attachments. It is surmised that while the attachments are energized and the breaker trips and closes a number of times, additional friction of the trip latch may occur from the vibration. This test is described in detail in the following section.

II. REVISED SURVEILLANCE OF REACTOR-TRIP CIRCUIT BREAKER OPERATION AND VERIFICATION TESTING

The licensee proposed the following increased surveillance of reactor-trip circuit breaker operation:

1. Main and bypass breakers will be shunt-tripped weekly.

2. Main breakers will be UV-tripped monthly.

The acceptability of this revised surveillance of reactor-trip circuit breaker operation has been evaluated by NRC staff. Based on an analysis conducted by NRC staff, which considered reactor-trip system unavailability, reactor-trip circuit breaker failure rates, and test intervals, the following conclusions were drawn. First, the proposed test of each reactor-trip circuit breaker UV trip attachment once every 30 days is acceptable. Second, the proposed test of the shunt trip attachment once every seven days is considered to be excessive and may impact on the reliability of the reactor trip system by increasing the potential for a single failure. During testing, a single failure in the logic portion of the reactor trip system could prevent an automatic SCRAM. Thus, it is recommended that the shunt trip attachment be tested on the same schedule as the UV trip attachment; that is, once every 30 days. It is also recommended that the UV trip of the bypass breakers be tested prior to restart and every refueling thereafter.

Discussion

The acceptability of the proposed test intervals for the reactor-trip circuit breakers was based on NRC staff review of reactor-trip circuit breaker failure rate data obtained from Licensee Event Reports (LERs). The generic RPS unavailability of 3×10^{-5} (used in both NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," and by the ATWS Task Force and Steering Group in the development of the proposed ATWS Rule) was used in evaluating the licensee's proposed test intervals. In addition, the following considerations were incorporated into the NRC staff recommendation:

1. The shunt trip attachment provides a diverse means of tripping the reactor-trip circuit breaker, which is electrically independent of the UV trip attachment. The UV trip attachment is supplied by a 48-V dc source and is deenergized to trip. The shunt trip attachment is supplied by a 125-V dc source and is energized to trip.

2. The shunt trip attachment is an energize-to-actuate device and is not "fail safe" in that a loss of power will not cause a trip. However, the shunt trip is powered from a reliable Class 1E battery-backed source.

3. Since the shunt trip attachment is an energize-to-actuate device, it is not subject to the constant heating effects that the continuously energized UV trip attachment experiences. The heating effects may contribute to the higher failure rate of the UV trip attachment.

4. The mechanical construction of the shunt trip attachment is less complex than that of the UV trip attachment. The shunt trip attachment does not rely on the successful operation of the complex latching mechanism that has been determined to be the source of the majority of the failures of the UV trip attachment.

5. The majority of the electrical circuit breakers used in the high-voltage electrical distribution system have dc-powered energize-to-actuate shunt trip attachments. These circuit breakers are used for manual, as well as automatic, trip functions for load shedding and power switching. Reliability of energize-to-actuate shunt trips in similar applications through-

out the nuclear power industry has been shown to be significantly higher than for devices that are constantly energized.

6. Over 70% of the known reactor-trip circuit breaker failures were caused by UV trip attachment failures.

7. Most of the concerns relating to the events at Salem on February 22 and 25, 1983 are related to the operation of the UV trip attachment. During the events at Salem, the shunt trip attachment functioned properly.

8. The bypass breakers are required to trip in response to a UV trip demand signal should this occur when the main breakers are being tested. Since the test frequency of the main breakers has been increased, the bypass breakers should be tested to verify the capability to perform their backup safety function.

Verification Testing

It is recommended that a bench test be performed on one DB-50 reactor-trip circuit breaker. The purpose of the test will be to cycle the DB-50 with the UV trip and shunt trip attachments in place for a total of 2000 cycles to determine if any adverse effects can be identified and, if there are no adverse effects, show that a properly maintained breaker and its subcomponents can operate for an extended number of cycles. The breaker will be tripped, with each cycle being alternated with the UV and shunt trips. The ambient temperature should be 100°F to simulate the expected service environment, and the circuit breaker should be cycled no more often than once every 30 minutes to allow for return to steady-state conditions. The results of each circuit breaker operation will be documented and a visual check made. Additional details for this type test will be provided at a later time.

The Philadelphia Inquirer
city/suburbs/region
section B
Thursday, March 3, 1983

NRC: Salem plant's safety system failed twice

By Matt Yancey
Associated Press

WASHINGTON — Federal officials yesterday blamed poor maintenance for the unprecedented failure — twice within a week — of a South Jersey nuclear plant's automatic safety system and expressed fear that the problem could be widespread among the nation's reactors.

The Nuclear Regulatory Commission (NRC) disclosed that the system that automatically shuts down a reactor when there are indications of unsafe conditions had failed for a second time last week at Public Service Electric & Gas Co.'s plant in Salem. One of the breakdowns had come to light previously.

Both breakdowns were attributed to the simultaneous failure of two circuit breakers that immediately insert control rods into a reactor, when either one is activated, to stop the nuclear chain reaction.

Roger Mattson, director of the NRC's systems integration division, said there had been 35 incidents since 1973 in which one circuit breaker failed in an automatic safety system. But he said the two incidents at the Salem plant last week marked the only examples of both the primary circuit breaker and its backup failing simultaneously.

Because the circuit breakers are considered vital safety equipment, utilities are required to follow elaborate, detailed procedures in maintaining them.

But officials said that Public Service Electric & Gas had never received maintenance bulletins that Westinghouse, the manufacturer of the equipment, put out in 1974.

In addition, when the breakers were taken apart for maintenance in January, they were classified incorrectly as nonsafety equipment and did not get the priority they should have received, officials said.

"Obviously, we take this situation we're in very, very seriously," said Richard Eckert, Public Service's senior vice president, who blamed human error for the incorrect classification of the equipment. "We need some time to sort this out."

The disclosures created concern yesterday among the five NRC commissioners.

"There's some very serious things that happened here," said Commissioner Victor Gilinsky, who added that he wanted a report on corrections that the utility was making before allowing it to restart the plant.

"If they labeled these incorrectly, isn't there a chance they labeled other pieces of equipment incorrectly?" Commissioner John Ahearne asked.

NRC officials acknowledged that the Salem incidents raised new questions about the level of quality control in maintenance programs at nuclear plants around the country.

The commission ordered a detailed investigation of similar circuit breakers used in other plants and how they are maintained. The owners of all Westinghouse reactors were instructed to test their ciruit breakers over the weekend, and none failed, NRC officials said.

Utilities with reactors purchased from other manufacturers also have been ordered to supply data on circuit breakers of similar design and on how they are maintained.

NRC officials said the second failure at the Salem plant last Friday and the 24½ seconds that passed before the control room operator shut down the reactor manually could have caused a severe accident with possible damage to the reactor core if the plant had been operating at full power.

The last nuclear accident that caused core damage in the United States was in March 1979 at the Three Mile Island nuclear plant near Harrisburg. There, in what became the worst accident in the industry's history, the automatic safety systems worked properly, but the operator, believing that the reactor was safely covered with water, shut them off.

NRC officials said the Salem Unit 1 reactor, which was starting back up after be[illegible] down for refueling, was operating at only 12 percent of its capacity last Friday. Because of that, they said, the plant was never in real danger.

But had the plant been operating a full power, Mattson said, the pressures inside the reactor and its cooling system would have created many more problems during the 24½-second gap, including possible core damage. He said he haddnot yet conducted studies to determine just how much damage could have occurred.

Officials said a similar total breakdown of the automatic safety system at the plant occurred three days earlier, on Feb. 22. It went unnoticed, they said, because the operator, seeing indications of unsafe conditions on his instruments, shut the plant down manually only four seconds after the failure.

The only previous incident of a reactor failing to shut down completely occurred at the Browns Ferry plant in Alabama in June 1980. There, 76 of the 185 control rods failed to insert fully for six minutes during a manual shutdown.

Of the 35 circuit-breaker failures in automatic safety systems since 1973, six have occurred at the Salem plant, including the four last week, Mattson said.

There are five other plants at which there has been more than one occasion on which one of the two safety-system breakers failed to work. They involve seven failures at the Oconee plant in South Carolina, six failures at the Zion plant in Wisconsin, two failures at the Robinson plant in South Carolina and three failures each at the Point Beach plant in Wisconsin and the Arkansas I reactor at Russelville.

out-of-adjustment within the linkage mechanism of the UV trip device installed in General Electric (GE) type AK-2 (i.e., AK2A-15, 25, 50, 75, 100) circuit breakers. Failures have occurred at ANO-1, Crystal River-3, Oconee Units 1 and 3, TMI-1, and St. Lucie. As a result of these events, the NRC issued IE Bulletin No. 79-09 dated April 17, 1979 and IE Circular 81-12 dated July 20, 1981. Subsequently, failures have been reported at ANO-1 and Rancho Seco.

Required Actions for Holders of Operating Licenses for Pressurized Water Reactors:

PWR licensees with other than W DB type breakers in Reactor Protective System applications are requested to:

1. Perform surveillance tests of undervoltage trip function independent of the shunt trip function within 5 days of receipt of this Bulletin unless equivalent testing has been performed within 10 days. Those plants currently shutdown should complete this item before resuming operation or within 10 days, whichever is sooner. Those plants for which on-line testability is not provided should complete this item at the next plant shutdown if currently operating.

2. Review the maintenance program for conformance to the latest manufacturer's recommendation, including frequency and lubrication. Verify actual implementation of the program.* If maintenance does not conform, initiate such maintenance within 5 days of receipt of this bulletin or provide an alternate maintenance program. Repeat the testing required in item 1 prior to declaring the breaker OPERABLE.

3. Notify all licensed operators of the failure-to-trip event which occurred at Salem (see IE Bulletin 83-01) and the testing failures at San Onofre Units 2 and 3 described above. Review the appropriate emergency operating procedures for the event of failure-to-trip with each operator upon his arrival on-shift.

4. Provide a written reply within 10 days of receipt of this bulletin:

   a. Identify results of testing performed in response to item 1. Plants without on-line testability should report the date and results of the most recent test.

   b. Identify conformance of the maintenance program to manufacturer's recommendation and describe results of maintenance performed directly as a result of this Bulletin in response to item 2.

*IE Bulletin 79-09, dated April 17, 1979, had as an attachment an extract of General Electric (GE) Service Advice Letter No. 175(CPDD)9.3 which is applicable to GE type AK-2 breakers.

c. Provide a statement that provisions are in place to notify licensed operators of the Salem and San Onofre events and bring to their attention appropriate failure-to-trip emergency procedures upon their arrival on-shift.

d. Provide a description of all RPS breaker malfunctions not previously reported to the NRC.

e. Verify that procurement, testing and maintenance activities treat the RPS breaker and UV devices as safety related. Report the results of this verification to the NRC.

5. Any RPS breaker failure identified as a result of testing requested by this bulletin should be promptly reported to the NRC via the emergency notification system, regardless of the operating mode of the plant at the time of the failure.

The written report required shall be telefaxed to Richard C. DeYoung, Director, Office of Inspection and Enforcement within 10 days of receipt of this bulletin.** At the same time, the report shall be submitted to the appropriate Regional Administrator under oath or affirmation under provisions of Section 182a, Atomic Energy Act of 1954, as amended. The original copy of the cover letters and a copy of the reports shall be transmitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555 for reproduction and distribution.

This request for information was approved by the Office of Management and Budget under a blanket clearance number 3150-00012 which expires April 30, 1985. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C. 20503.

If you have any questions regarding this matter, please contact the Regional Administrator of the NRC Regional Office or the technical contact listed below.

James H. Sniezek for
Richard C. DeYoung, Director
Office of Inspection and Enforcement

Technical Contact: I. Villalva, IE — 301-492-9635
V. Thomas, IE — 301-492-4755

Attachment:
1. List of Recently Issued IE Bulletins

**Rapidfax (301) 492-8187 or (301) 492-7376
3M Remote Copies (301) 492-7285

I.D.3. NECNP has withdrawn this contention.

I.D.4. NECNP has withdrawn this contention.

I.F. NECNP has withdrawn this contention.

I.G. Pressure Instrument Reliability

In this contention, NECNP asserts that the Seabrook wide range pressure instruments cannot be relied upon for accurate information, and thus may lead to inappropriate operator actions jeopardizing the cooling of the reactor. This contention was based on IE Notice no. 82-11 (April 9, 1982) which reported a significant margin of inaccuracy (±363 psig actuation and ±390 psig indication) in the instruments during qualification tests in a post-accident high energy line break environment. The NRC concluded that these inaccuracies could result in "inappropriate operator actions."

In their answers to NECNP's interrogatories, Applicants identified the wide range pressure transmitters as PCT 403 and PCT 405, and supplied NECNP with a drawing (No. 9763-M-506635) showing the two transmitters to be outside the containment.

By motion of February 11, 1983, the Applicants requested the Board to summarily dispose of NECNP Contention I.G. on the sole ground that "the Seabrook wide range pressure transmitters are located outside the containment and thus are not subject to the hig[illegible] energy line break environment which caused the inaccuracies addressed in IE Information Notice 82-11."

Applicants attached an affidavit stating that the wide range pressure transmitted are located outside containment, but supplied no diagrams in support of this assertion.

Applicants' statement regarding the location of the wide range pressure transmitters is squarely contradicted by the FSAR. Figure 5.1-1, SH 2 and SH 5 of the FSAR (attached), entitled "Reactor Coolant System Loop No. 1, P & ID Diagram", show the wide range pressure transmitters (PCT 403 and 405) to be inside the containment. Applicants provide no explanation for the discrepancy between the statement in the summary judgment motion and the FSAR. Nor have Applicants explained why the drawing supplied in response to NECNP's interrogatories differs from the FSAR.

The location of the wide range pressure transmitters is the only ground raised by Applicants in moving for summary judgment. The Applicants' own documents establish a clear conflict in the evidence on this issue. Since material issue of fact remains to be resolved, Applicants are not entitled to summary judgment.

MATERIAL FACTS AS TO WHICH THERE IS A
GENUINE ISSUE TO BE HEARD

1. According to Applicants' Final Safety Analysis Report, Figure 5.1-1, SH 2 and SH5, the wide range pressure transmitters (PCT 403 and PCT 405) are located inside the containment and are therefore subject to the type of accident environment which caused the instrument ambiguities noted in IE Notice 82-11.

I.I. Environmental Qualification of a Path to Cold Shutdown

In Contention I.I., NECNP asserted that Applicants must identify and environmentally qualify a path to cold shutdown. Applicants have moved for summary disposition on the ground that the NRC's latest rule on environmental qualification 10 CFF 50.49, excluded a requirement for environmental qualification of a path to cold shutdown. We note that in doing so, the Commission stated that shutdown decay heat removal remains an Unresolved Safety Issue which the Commission continues to study. 48 Fed. Reg. at 2731 (January 21, 1983). Therefore, the adequacy of residual heat removal, including consideration of cold shutdown capacity, must still be resolved on a plant-specific basis.

In any event, NECNP considers this issue to be resolved to our satisfaction. Despite their protests in this proceeding that it is not required, Applicants have informed the NRC Staff by letter that they have qualified a path to cold shutdown, with the exception of the pressurizer heaters. Letter from J. DeVincentis, Yankee Atomic Electric Co., to George W. Knighton, U.S. NRC, re: Clarification to RAI 440.133 (Letter No. SBN480) (February 25, 1982). In light of this commitment, NECNP withdraws contention I.I.

II. B.1. NECNP has withdrawn this contention.

II.B.3. Independence of QA program

NECNP Contention II.B.3. asserts that the Applicants' Quality Assurance ("QA") organization for operations does not have the independence required by 10 CFR Appendix B, Criterion 1. As reflected in the basis for the contention and in NECNP's responses to Applicants' interrogatories on the point, the contention is based upon and limited to the fact that the Nuclear Quality Manager, who is responsible for the QA program, reports to the Vice President - Nuclear Production, who is responsible for the operation of the Seabrook station, and therefore is responsible for assuring that Seabrook remains on line as much as possible to maximize the financial benefit to the company.

Criterion I of Appendix B requires, in pertinent part, that

> [t]he persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. <u>Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations are provided.</u>

(Emphasis supplied).

As previously noted, at Seabrook, the Nuclear Quality Manager reports directly to the Vice President - Nuclear Production. FSAR 17.2.1.3.a.1. However, the FSAR also establishes that the Vice President - Nuclear Production "is responsible for the operation and operational support of

Seabrook Station." FSAR 13.1.1.1., FSAR 17.2.1.3.a. This description alone, particularly when read in light of the title of the position, establishes that the Vice President for Nuclear Production is responsible for maximizing the operation of Seabrook Station so that it produces electricity and contributes to the Applicants' rate base to the maximum extent possible. Neither the Applicants nor the Staff has denied that this individual is responsible for maximizing electricity production. Moreover, a review of Chapters 13 and 17 of the FSAR reveals no other manager above the site-based staff who has this responsibility. The necessary conclusion is that the Vice President - Nuclear Production has it, and that this is his or her primary responsibility.

Neither the Applicants nor the Staff contest this. Rather, they rely upon (1) the fact that NUREG-0731 shows a structure in which the QA Department reports to a Vice President - Nuclear Production (Applicants' Motion, Material Facts #2, Affidavit #2(b)), (2) the fact that the Seabrook organization arguably comports with the type of organization suggested in the Standard Review Plan, which is used by the Staff in reviewing license applications (Staff Motion, Material Facts #5), and (3) the opinion of a Staff member that, because the SRP suggestion is met, the QA program at Seabrook has sufficient independence under Appendix B.

These assertions avail the Applicants and Staff of nothing. Both NUREG-0731 and the SRP are merely Staff

documents. They have no force of law. They reflect no review of Seabrook itself. More important, they do not refute the fact that at Sebrook, according to the FSAR, the QA program would be reporting directly to the individual with a major vested interest in assuring that QA problems do not cause shutdown or delayed operation, which would interfere with electricity production, this Vice President's primary responsibility.

Other portions of the FSAR are revealing and disturbing in this regard. According to FSAR 17.2.1.3.a.1, the Nuclear Quality Group has the authority to "request work stoppages or remedial actions if conditions adverse to quality are encountered" (emphasis supplied). This conflicts with a reference at FSAR Page 17.2-6 that the Quality Assurance Department may exercise stop-work authority. At a minimum, the FSAR and the record now before the Board are unclear on the extent of the authority vested in the Nuclear Quality Group.

In addition, all disputes between QA and production personnel are to be mediated by the Vice President - Nuclear Production, the very individual responsibile for maximizing the operation of Seabrook. Although the dispute may be appealed to the Executive Vice President - Engineering and Production, the FSAR gives no indication that the QA department has direct access to that officer. Moreover, to appeal any dispute to this relatively impartial officer, the Nuclear Quality Manager would have to contest a decision by his immediate superior, who appears from the FSAR to be responsible for all aspects of the

QA function, including whether the Nuclear Quality Manager keeps his job. This has a serious chilling effect on any necessary appeals to the first individual in the line of command who appears to be sufficiently removed from production pressures to render an independent opinion.

Nothing presented by the Staff or Applicants has resolved or even approached the factual dispute here, which involves the independence of a QA program under the direct control of a production-oriented official. Compliance with two non-regulatory Staff documents indicates nothing. In the absence of evidence refuting the foregoing discussion, which is based on the language of the FSAR, these motions must be denied.

II.B.4. QA of Replacement Parts

NECNP Contention II.B.4. asserts that the Seabrook QA program for operations as described in the FSAR does not demonstrate how the Applicants will assure that replacement parts are equivalent to original parts and installed in accordance with proper procedures. It also asserts that the QA program does not demonstrate how repaired or reworked items will be adequately inspected and tested. It is based on the absence of any demonstration in the FSAR itself.

Applicants move for summary disposition on the ground that these matters are covered in the FSAR at §§ 17.2.4 and 17.2.15. The supporting affidavit similarly references those sections. In addition, it references a matrix of procedures being prepared to address various quality assurance requirements. According to the affidavit, "These procedures are presently being developed and previewed; will be available prior to fuel load; and will provide the implementation of the Quality Assurance Program."

The Staff's motion cites further FSAR references, including a cryptic statment by Applicants that it complies with certain Regulatory Guides. These provisions involve general commitments to meet QA requirements.

There is no factual dispute as to the language of the FSAR or the various referenced documents. The issue may thus be decided as a matter of law based on that language.

The governing regulation is 10 C.F.R. § 50.34(b)(6)(ii), which requires that the FSAR meet the following standard:

> The information on the controls to be used for a nuclear power plant...shall include a discussion of how the applicable requirements of Appendix B will be satisfied.

(Emphasis supplied). The FSAR falls woefully short of this standard, and the affidavit supplied by the Applicants, as well as Applicants' response to NECNP's interrogatories on this issue, demonstrate the Applicants have not yet even determined how the QA requirements will be met. We address in turn (1) the alleged QA program for replacement parts, and (2) the alleged QA program for rework and repair.

The FSAR discussion of QA for replacement parts appears at § 17.2.4. That section, a total of only three pages of text, is purported to demonstrate how QA requirements will be met for replacement parts. It does no such thing. To the contrary, it includes only a very general discussion describing generally the framework of what the Applicants expect to do in this area.

The discussion indicates, for example, that all purchase requests will include, " as appropriate," "Quality Assurance Requirements." But there is no indication what those requirements will be or how Applicants will assure that they are met by the vendor. There follows a mention of the "quality review," but there is no indication of what that review might be. Indeed, for "off-the-shelf" items, Applicants appear to indicate that they have not decided how they will perform

quality control. Rather, they will develop special requirements for each item. (FSAR Page 17.2-21).[5/] Not only is there not even the slightest outline of what these requirements might be, there is no indication of how they will be developed. Who will be responsible for developing them? Who will review them when they are developed? Who has initial approval responsibility? Who has ultimate approval authority? The FSAR provides none of this information.

The same is true of the rest of the discussion. It is simply impossible to determine from the sketchy information in the FSAR <u>how</u> the QA program will be implemented for replacement parts. The discussion constitutes nothing more than a general assertion that the Applicants intends to implement it. That is not enough under 10 CFR § 50.34(b)(6)(ii).

With respect to repair and rework, the Applicants rely on less than one page of text in FSAR § 17.2.15. Again, this is only the most general discussion of the point. There is no

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5/ In fact, the FSAR does not identify who will develop these requirements. It says simply that they "will be established." This leaves open the possibility that the Applicants will rely on the vendors to do the work, although the Applicants are responsible. The FSAR commits the Applicants to nothing more than that.

indication of how or by whom the inspections will be performed. Beyond the cryptic mention of nonconformance reports, there is no indication of how the QA will be documented. Perhaps more important, there is no indication of how or with what frequency the crucial trend analysis will be performed to prevent repetition of significant quality problems. The FSAR states only that some such system will be maintained, but does no explain how it will be implemented.

The Staff has added little to the Applicants' discussion. The fact that the Applicants have commited to meeting various Regulatory Guides or other standards does not demonstrate how the Applicants will do so. That is what the regulation requires.

Finally, as previously noted, the Applicants' affidavit establishes that they have not yet determined, or at least nave not yet chosen to reveal, how the QA program will be implemented. Similarly, in response to NECNP's interrogatories, Applicants stated that

> The procedures which implement the QA program are not yet fully approved and are still under review and development. They will be made available when finally approved.

Applicants' Answers to "NECNP Second Set of Interrogatories and Requests for Documents to Applicants on Contentions I.D.1., I.D.4., I.F., I.I., I.L., and II.B" and Motion for Protective Order, filed January 29, 1983, at 15. These procedures are precisely what are needed to determine how the QA requirements will be met for replacement parts. According to Applicants' affidavit, we can expect them only sometime "prior to fuel load."

The very purpose of detailed procedures in any complex operation is to establish how the relevant task will be accomplished. It may be that something less than the procedures themselves would suffice, but the information does not appear in the FSAR, as required by the regulation. Thus the Applicants are avoiding public scrutiny of one of the most serious issues of reactor operation - the quality assurance program that is necessary to safety when actions are taken to alter or repair the reactor.

For these reasons, we urge the Board to deny the motions for summary disposition filed by the Applicants and the Staff.

CONTENTION II.B.4
STATEMENT OF MATERIAL FACT AS TO WHICH THERE
EXISTS A GENUINE ISSUE TO BE HEARD

1. The portions of the FSAR cited by the applicant and Staff establish that the required program for inspection and testing of repaired or replaced parts is not complete. FSAR §§ 17.2.4, 17.2.15; Applicants Answer to "NECNP Second Set of Interrogatories...on Contentions I.D.1, I.D.4., I.F., I.I., I.L., and II.B" and Motion for Protective Order, filed January 29, 1983, at 15.

II.B.5. NECNP has withdrawn this contention.

Respectfully submitted,

Diane Curran

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CERTIFICATE OF SERVICE

I certify that on March 24, 1983, copies of NECNP OPPOSITION TO MOTIONS FOR SUMMARY DISPOSITION AND NOTIFICATION OF WITHDRAWN CONTENTIONS were served by first class mail or as otherwise indicated on the following:

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