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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
GEORGIA POWER COMPANY, ET AL.	)	Docket Nos. 50-424
	)	50-425
(Vogtle Electric Generating Plant,	)	
Units 1 and 2)	)	

APPLICANTS' RESPONSE TO GANE AND CPG SUPPLEMENTS TO PETITIONS FOR LEAVE TO INTERVENE

I. Introduction

By Memorandum and Order dated March 9, 1984, the Atomic Safety and Licensing Board established a schedule for the filing of proposed contentions and responses. On April 11, 1984, Petitioners Campaign For a Prosperous Georgia (CPG) and Georgians Against Nuclear Energy (GANE) filed supplements to their petitions for leave to intervene, each setting forth thirteen proposed contentions.1/ Applicants herein provide

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1/ Georgians Against Nuclear Energy Supplement to Petition for Leave to Intervene and Request for Hearing (April 11, 1984) (hereinafter GANE Supplement); Campaign for a Prosperous Georgia Supplement to Petition for Leave to Intervene and Request for Hearing (April 11, 1984) (hereinafter CPG Supplement).

their response to the two supplements. In the following section II, Applicants address the existing standards for admissibility of proposed contentions; and in section III, Applicants apply those standards to Petitioners' contentions.

II. Standards For the Admissibility of Contentions

The Commission's Rules of Practice, at 10 C.F.R. § 2.714, require that a petitioner submit a list of contentions which petitioner seeks to have litigated and set forth the basis for each contention with reasonable specificity. This standard requires that a contention state a cognizable issue with particularity, Alabama Power Company (Joseph M. Farley Nuclear Plant, Units 1 and 2), ALAB-182, 7 A.E.C. 210, 216-17 (1974), and include a "reason" in support. Houston Lighting and Power Company (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 N.R.C. 542, 548 (1980).

As a general proposition, a Licensing Board should not address the merits of a contention in determining admissibility. Houston Lighting and Power Company (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 N.R.C. 542, 548 (1980). However, a contention and its basis must be scrutinized to determine if a specific, litigable issue has been pleaded. Joseph M. Farley, supra, ALAB-182, 7 A.E.C. at 216. Such scrutiny is necessary 1) "to assure that the proposed issues are proper for adjudication," 2) "to help assure at the

pleading stage that the hearing process is not improperly invoked," and 3) "to help assure that other parties are sufficiently put on notice so that they will know at least generally what they will have to defend against or oppose." Philadelphia Electric Company, et al. (Peach Bottom Atomic Power Station, Units 2 and 3), ALAB-216, 8 A.E.C. 13, 20-21 (1974) (footnotes omitted). In this regard, there must be strict observance of the requirements governing intervention. Id. at 21. Elaboration of principles particularly applicable to Petitioners' contentions follows.

A. The Issues Must be Proper for Adjudication in the Proceeding

At the broadest level, a contention is not cognizable unless it addresses a matter that is within the scope of the issues set forth in the Commission's Notice and Opportunity for Hearing in the proceeding. See Northern Indiana Public Service Company (Bailly Generating Station, Nuclear 1), ALAB-619, 12 N.R.C. 558, 565 (1980); Portland General Electric Company, et al. (Trojan Nuclear Plant), ALAB-534, 9 N.R.C. 287, 289-90 n.6 (1979); Public Service Company of Indiana, Inc. (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-316, 3 N.R.C. 167, 170-71 (1976). A contention must be relevant and material to those matters defined in 10 C.F.R. § 2.104(c).

To establish relevance, a petitioner, in setting forth the basis for its contention, should establish a nexus between the

substance of the contention and the statutory and regulatory scope of the Board's jurisdiction. Public Service Company of New Hampshire, et al. (Seabrook Station, Units 1 and 2), LBP-82-106, 16 N.R.C. 1649, 1654 (1982). With respect to any safety issue, it should specify a regulation with which applicant is allegedly not complying and provide sufficient detail to permit the Board to determine how the regulation is being violated; or it should allege with particularity the existence and detail of a substantial safety issue on which the regulations are silent. Id. at 1656. This requirement is often referred to as the "legal basis" for a contention.

To establish materiality, the contention must provide a foundation sufficient to warrant further exploration. Peach Bottom, supra, ALAB-216, 8 A.E.C. at 21; Duquesne Light Company (Beaver Valley Power Station, Unit No. 1), ALAB-109, 6 A.E.C. 243, 246 (1973). See also Seabrook, supra, LBP-82-106, 16 N.R.C. at 1655, citing Consumers Power Company (Midland Plant, Units 1 and 2), CLI-74-5, 7 A.E.C. 19, 32 n.27 (1974), rev'd sub nom., Aeschliman v. NRC, 547 F.2d 622 (D.C. Cir. 1976), rev'd sub nom., Vermont Yankee Nuclear Power Corp. v. NRDC, 435 U.S. 519, 553-54 (1978), for the proposition that a contention must be sufficient to require reasonable minds to inquire further. In this regard, the basis for the contention should provide either a reasonably logical and technically credible explanation (Philadelphia Electric Company (Limerick Generating

Station, Units 1 and 2), ALAB-765, 19 N.R.C. ______, slip op. at 13 (March 30, 1984), or a plausible and referenced authority for the factual assertions in the contention. The petitioner's personal opinion alone is not adequate for this purpose. This materiality requirement is often referred to as "factual basis."

A contention must also have application to the facility in question. Beaver Valley, supra, ALAB-109, 6 A.E.C. at 246 n.5. A contention should refer to and address documentation, available in the public domain, that is relevant to this facility. Cleveland Electric Illuminating Company, et al. (Perry Nuclear Power Plant, Units 1 & 2), LBP-81-24, 14 N.R.C. 175, 184 (1981). The Commission itself has recently emphasized petitioners' duties in this regard. See generally Duke Power Company, et al. (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 N.R.C. 1041 (1983). The Commission held that a petitioner has an ironclad obligation "to diligently uncover and apply all publicly available information to the prompt formulation of contentions." Id. at 1048. This requirement for specific reference to relevant documentation applies with special force to an applicant's Final Safety Analysis Report and Environmental Report, but may also include applicable NRC Staff regulatory guides and other published reports. If a contention inaccurately describes an applicant's proposal or misstates the content of licensing documents, it should be rejected. See

Carolina Power & Light Company, and North Carolina Eastern Municipal Power Agency (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 N.R.C. 2069, 2076 (1982); Duke Power Company, et al. (Catawba Nuclear Station, Units 1 and 2), LBP-32-107A, 16 N.R.C. 1791, 1804 (1982); Philadelphia Electric Company (Limerick Generating Station, Units 1 and 2), LBP-82-43A, 15 N.R.C. 1423, 1504-05 (1982).

B. The Hearing Process Must Not Be Improperly Invoked

A contention is not appropriate for litigation if it collaterally attacks a Commission rule or regulation. 10 C.F.R. § 2.758.2/ Similarly, as a general proposition, Licensing Boards should not accept in individual proceedings contentions which are or are about to become the subject of general rulemaking by the Commission. Potomac Electric Power Company

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2/ A party may obtain a waiver of a rule, but only if it submits to the Licensing Board for certification to the Commission a petition setting forth with particularity special circumstances with respect to the subject matter of the particular proceeding which are such that the rule or regulation (or provision thereof) fails to serve the purposes for which the rule or regulation was adopted. The petition must be accompanied by supporting affidavit. Opportunity is provided for other parties to respond to the petition, including the submission of reply affidavits. If the Licensing Board determines that a prima facie showing has been made in support of waiver or exception, it shall, before ruling, certify directly to the Commission for a determination on the matter. If the Licensing Board does not determine that such a prima facie showing has been made, it must deny the petition. 10 C.F.R. § 2.758; Potomac Electric Power Company (Douglas Point Nuclear Generating Station, Units 1 and 2), ALAB-218, 8 A.E.C. 79, 89 (1974).

(Douglas Point Nuclear Generating Station, Units 1 and 2), ALAB-218, 8 A.E.C. 79, 85 (1974). This policy avoids wasteful duplication of effort (id.), and it also avoids regulatory inconsistency. In the same vein, Commission policy statements and policy declarations are binding on the Boards. Mississippi Power & Light Company, et al. (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-704, 16 N.R.C. 1725, 1732 (1982); Northern States Power Company (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 N.R.C. 41, 51 (1978), remanded on other grounds sub nom., Minnesota v. NRC, 602 F.2d 412 (D.C. Cir. 1979).

The general proposition that Boards should not decide issues which are the subject of rulemaking, however, is not absolute. Some issues may be litigated. The key to this determination is whether the Commission intended complete deferral of all issues to the future rulemakings. See Cleveland Electric Illuminating Company, et al. (Perry Nuclear Power Plant, Units 1 and 2), ALAB-675, 15 N.R.C. 1105, 1112 (1982).

C. The Parties Must Be Put on Notice of the Issues

The notice aspect of the requirement is a natural outgrowth of fundamental notions of fairness applied to the party with the burden of proof. As the Atomic Safety and Licensing Appeal Board has observed:

> The applicant is entitled to a fair chance to defend. It is therefore entitled to be told at the outset, with clarity and precision, what arguments are being advanced and what relief is being asked. . . . So is the Board below. It should not be necessary to speculate about what a pleading is supposed to mean.

Kansas Gas and Electric Company, et al. (Wolf Creek Generating Station, Unit No. 1), ALAB-279, 1 N.R.C. 559, 576 (1975) (emphasis added; footnote omitted). Moreover, the Licensing Board is entitled to adequate notice of a petitioner's specific contentions to enable it to guard against the obstructionism of its processes. As noted by the Supreme Court in upholding the Commission's requirements for a threshold showing of materiality:

> . . . [I]t is still incumbent upon intervenors who wish to participate to structure their participation so that it is meaningful, so that it alerts the agency to the intervenors' position and contention. . . . Indeed, administrative proceedings should not be a game or forum to engage in unjustified obstructionism by making cryptic and obscure reference to matters that "ought to be" considered. . . .

Vermont Yankee Nuclear Power Corporation v. Natural Resources Defense Council, 435 U.S. 519, 553-554 (1978).

The specificity requirement transcends "notice pleading" allowed in the federal courts, which has been found to be insufficient for NRC licensing proceedings. See Wolf Creek, supra, ALAB-279, 1 N.R.C. at 575 n.32 (1975). It does not, however, require the petition to detail the evidence which will

be offered in support of each contention. Peach Bottom, supra, ALAB-216, 8 A.E.C. 13, 20 (1974).3/ In short, the standard falls somewhere in between, and "[t]he degree of specificity with which the basis for a contention must be alleged initially involves the exercise of judgment on a case-by-case basis." Id.

III. Petitioners' Contentions

Petitioners have adopted a uniform format in pleading their contentions. Each contention is set out in single-spaced format, and is followed by a discussion of the contention.4/ GANE contentions 5 through 13 are identical to CPG contentions 5 through 13. Applicants discuss below GANE contentions 1 through 4, then GANE/CPG contentions 5 through 13, and then CPG contentions 1 through 4.

Petitioners' discussions of their contentions appear to serve two purposes. First, they purport to provide Petitioners' basis for each contention; and second, they serve in most

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3/ See also Missouri Power and Light Company (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 A.E.C. 423, 426 (1973); Houston Lighting and Power Company (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 N.R.C. 542, 548-549 (1980).

4/ GANE contentions 1 and 2 are not set out in a single-spaced format. However, Applicants assume GANE intended the same format for its contentions 1 and 2, and that the first paragraphs after the headings "GANE 1" and "GANE 2" are the actual contentions.

instances to provide much needed specificity. Without the discussion, virtually all the contentions would lack any basis whatsoever and would also be so broad as to be unlitigable.5/ Accordingly, any contention admitted should be limited to the specific subjects addressed by the Petitioners in their discussion of that contention.

A. GANE-1: Radioactive Releases

GANE-1 contains three allegations: (1) Applicants have not correctly assessed the potential routine and accidental releases of radionuclides from Vogtle; (2) Applicants have not correctly assessed the health effects of ionizing radiation; and (3) Applicants underestimate the human cost in the NEPA cost benefit analysis.

1. Radioactive releases.

GANE does not explain why Applicants' release estimates are incorrect, but simply asserts Applicants fail "to meet the requirements of 10 C.F.R. 50.34, 50.36, 20.103, 20.203, and Appendix I of Part 50." GANE Supplement at 1.

10 C.F.R. §§ 50.34 and 50.36 pertain to the contents of license applications and to technical specifications, and they

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5/ See, in particular, Applicants' discussion of GANE-8/CPG-8, GANE-10/CPG-10, and GANE-11/CPG-11. For example, in GANE-11/CPG-11, Petitioners allege inadequate consideration of "generic defects in the Westinghouse PWR." The discussion of the contention, however, addresses several specific Unresolved Safety Issues.

are not specifically applicable to radioactive releases. Presumably, GANE intended to refer to 10 C.F.R §§ 50.34a, which requires estimates of radioactive releases during normal plant operation, and to 10 C.F.R. § 50.36a, which requires as a technical specification Applicants' compliance with 10 C.F.R. § 20.106 and the ALARA (as low as reasonably achievable) standards. None of these provisions applies to releases during an accident.

Applicants comply with 10 C.F.R. § 50.34a. Applicants' calculated releases of radioactive materials in liquid effluents during normal plant operation are set forth in Table 11.2.3-1 of the FSAR.6/ Calculated airborne releases are set forth in Table 11.3.3-2. Both sets of calculations were derived by standard methodology -- the use of the NRC's PWR-GALE

6/ Applicants use the following abbreviations to refer to licensing documents:

FSAR - Applicants' Final Safety Analysis Report

PSAR - Applicants' Preliminary Safety Analysis Report

CP-ER - Applicants' Construction Permit Stage Environmental Report

OL-ER - Applicants' Operating License Stage Environmental Report

CP-FES - The NRC Staff's Construction Permit Stage Final Environmental Statement

CP-SER - The NRC Staff's Construction Permit Stage Safety Evaluation Report

computer code and plant specific parameters. FSAR, §§ 11.2.3.2 and 11.3.3.3. Accidental releases are evaluated in the FSAR at § 15.7 and Appendix 15A. (See also OL-ER, §§ 7.1, 7A.) GANE makes no attempt to address these estimates and points to no error in the methodology or result. In fact, GANE provides no specificity and no basis to conclude that any of the release estimates are in error.

GANE's reference to 10 C.F.R. §§ 20.103, 20.203, and 10 C.F.R. Part 50, Appendix I, and its attempted reference to 10 C.F.R. § 50.36a, could be read to imply that GANE, in addition to disputing Applicants' calculated releases, is also asserting that Applicants' calculated releases exceed permissible limits. If so, the contention falls far short of the specificity demanded by the Commission's Rules of Practice. Applicants should not be required to guess what issues GANE seeks to raise. Moreover, GANE's citations to sections in 10 C.F.R. Part 20 are peculiar. 10 C.F.R. § 20.203 governs warning signs and labels; and 10 C.F.R. § 20.103 applies to exposures in restricted areas.

Nevertheless, even if one assumes that GANE is indeed attempting to assert that Applicants' estimated releases exceed permissible levels and ignores that its assertion is insufficiently articulated, its pleadings are still inadequate. In Tables 11.2.3-2 and 11.3.3-3 of the FSAR, Applicants compare their calculated airborne and effluent concentrations with 10

C.F.R. Part 20 limits, and these tables indicate that the calculated concentrations are very small fractions of the 10 C.F.R. Part 20 limits. In Tables 11.2.3-4 and 11.3.3-4 of the FSAR, Applicants estimate annual doses to individuals from liquid, gaseous, and particulate effluents, and demonstrate compliance with 10 C.F.R. Part 50, Appendix I. (See also OL-ER, Tables 5.2-5 and 5.2-6.)

GANE does not address or refute these comparisons; instead, in its discussion of GANE-1, GANE asserts that Applicants fail "to thoroughly assess the risk from routine and accidental radioactive releases on the food chain in its assessment of recreation activities (VEGP-OLSER-2, 2.1-8 and VEGP-FSAR-2, 2.1.3-2) in violation of 10 C.F.R. 20.106. . . ."7/ GANE Supplement at 2. But 10 C.F.R. § 20.106 does not require risk assessment; 10 C.F.R. § 20.106 requires Applicants to meet the limits on concentrations of radioactive materials in air and water specified in 10 C.F.R. Part 20, Appendix B,8/ as Applicants have demonstrated they will do.

7/ The sections of the FSAR and OL-ER to which GANE refers are merely site descriptions, and are not intended to assess risk or potential releases.

8/ The 10 C.F.R. Part 20 standards are based on the recommendations of the Federal Radiation Council (FRC), and parallel recommendations of the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP). The radiation protection guides established by the FRC for individual members of the

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GANE also cites J. Gofman, Poisoned Power (1971), for the proposition that "an adult man consuming two pounds of fish per year from this stretch of Savannah River is exposed to cesium-137 levels in excess of the maximum allowable amount." GANE Supplement at 2. However, Gofman's statement in this book actually reads as follows:

> These levels [in 10 C.F.R. Part 20, Appendix B] are set so that a whole-body dosage of 0.5 rads per year would result from breathing such air for one year, or drinking about two quarts of such contaminated water per day. But what do such levels

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public are 500 millirem per year to the total body and bone marrow (one tenth the maximum occupational dose) and 1500 millirem per year to the thyroid and bone. The guide for average dose to the population is 5 rem in 30 years to the gonads (an annual average dose of 170 millirem per person averaged over the population). These guidelines apply to exposures from all sources other than medical procedures and natural background. See Rulemaking Hearing: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, CLI-75-5, 1 N.R.C. 277, 279-80 (1975). The concentrations in 10 C.F.R. Part 20, Appendix B are the estimated concentrations of the individual radionuclides which may be permitted in air or water used continuously by an average or "standard" man without resulting in a radiation dose that would exceed the maximum permissible individual dose. Rogers & Gamertsfelder, USA Regulations for the Control of Releases of Radioactivity Into the Environment in Effluents from Nuclear Facilities, reprinted in, International Atomic Energy Agency, Environmental Aspects of Nuclear Power Stations 128, 129 (Vienna, 1971). The Commission has determined that "any biological effects that might occur at the low levels of these standards have such a low probability of occurrence that they would escape detection by present-day methods of observation and measurement. Rulemaking Hearing, supra, CLI-75-5, 1 N.R.C. at 280.

really mean in terms of what could occur, and probably will occur if such levels are allowed in an unrestricted area where people live?

* * *

Let's look at the concentration in water. The MPC [Maximum Permissible Concentration] is based upon the calculation that a 150-lb. standard man consuming 2200 grams of water at the MPC per day would receive a dose of 0.5 rad. . . . The concentration of Cs-137 in fish flesh, caught in a river, would be 1000 times higher than the concentration in the water. Thus a man eating 1-lb. of fish a week, grown in water at the MPC, would receive a dosage of 15 rad/yr or 30 times the 0.5 rad guideline and 90 times the 0.17 rad guideline.

J. Gofman & A. Tamplin, Poisoned Power 120-21 (1979 ed.). GANE's source, therefore, refers to an adult man eating 1 pound of fish per week grown in water at the MPC, not to an adult man consuming two pounds of fish per year grown in Savannah River water, as GANE implies.9/

The radioactive concentrations in Savannah River due to Vogtle releases will be far below MPC. For cesium-137, the estimated concentration in the Savannah River will be less than 1/10,000 of the MPC!10/ See FSAR, Table 11.2.3-2. GANE does

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9/ Furthermore, Gofman was clearly attacking the adequacy of 10 C.F.R. Part 20, and GANE is merely incorporating by reference this challenge to the rules. On this ground alone, the assertion is not cognizable in this proceeding.

10/ To achieve the dose that concerned Gofman, a man would have to eat several tons of fish per week from the Savannah River.

not address this calculated concentration. Furthermore, Applicants have not ignored bioaccumulation in fish and the fish-man exposure pathway. Maximum doses from liquid releases were calculated using the LADTAP II Code and Regulatory Guide 1.109 (Rev. 1, October 1977). FSAR, § 11.2.3.5. Regulatory Guide 1.109, § C.1.b. and Appendix A, provide guidance for assessing the aquatic food pathway and give a bioaccumulation factor 2x10+U3+D. Using this methodology, and assuming that an adult would consume 21 kilograms of Savannah River fish per year, Applicants calculated the contribution from the fish exposure pathway to the annual total body dose of an adult to be .791 millirem (FSAR, Table 11.2.3-4), or 1/632 of the 500 millirem NRC standard for individuals. See note 8 supra. In addition, neither this dose nor the total maximum dose from all exposure pathways exceed the ALARA standard in 10 C.F.R. Part 50, Appendix I.11/

11/ The ALARA standard in 10 C.F.R. Part 50, Appendix I does not supplant the 10 C.F.R. Part 20 standards. The Part 20 limits are still those at which "any biological effects that might occur have such low probability of occurence that they would escape detection." Rulemaking Hearing, note 8 supra, CLI-75-5, 1 N.R.C. at 280. Instead, the ALARA concept embodies a policy decision that despite the safety of Part 20 concentrations, every effort should be made to minimize exposure. Accordingly, the low as reasonably acheivable limits were derived by considering the technological and economic feasibility of methods to reduce radioactive releases. See generally id. The result, 10 C.F.R. Part 50, Appendix I, established ALARA design objectives for light-water reactors. For gaseous effluents, the design objective is 5 millirem per year per unit total body dose, 15

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GANE next refers to airborne particulates and asserts that the milk exposure pathway "poses significant health risks not assessed by applicant." Again, GANE makes no attempt to address Applicants' licensing documents. Applicants did indeed assess the dose to the nearest individual from this pathway in the FSAR at Table 11.3.3-4 (sheet 4), using the GASPAR Code, site specific parameters, and the guidance in Reg. Guide 1.109, § C.3 and Appendix C. FSAR, § 11.3.3.6. GANE does not refute these calculations or the methodology.12/ The estimated doses are well within the ALARA limits.

Finally, GANE asserts that Applicants fail "to include doses to which pregnant or lactating women would be exposed." Applicants submit that the doses these women may receive are the same doses any other adult would receive, and as stated above, Applicants have calculated maximum doses to individuals. FSAR, Tables 11.2.3-4 and 11.3.3-4. To the extent that GANE's

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millirem per year per unit dose to the skin; for radioiodines and particulates, the design objective is 15 millirem per year per unit to any organ; and for liquid effluents, the design objective is 3 millirem per year per unit total body dose, 5 millirem per year per site total body dose, and 10 millirem per year per site dose to any organ.

12/ GANE also makes reference to a 1961 accidental release of radioiodine from the Savannah River Plant but provides no explanation of the relevance of this accident to Vogtle, and no indication that 23 years later a similar accident is likely to occur at what can be assumed to be a vastly different facility.

contention may have meant to refer to the health effects of doses to pregnant and lactating women, Applicants discuss health effects issues below.

2. Health Effects

GANE alleges that Applicants fail to meet the requirements of 10 C.F.R. §§ 50.34, 50.36, 20.103, 20.203, and 10 C.F.R. Part 50, Appendix I, by not adequately assessing the somatic, teratogenic, and genetic effects of ionizing radiation. GANE Supplement at 1. Not one of those regulations requires Applicants to assess health effects. Applicants must comply with 10 C.F.R. Part 20 and 10 C.F.R. Part 50, Appendix I (ALARA). If they do so, operation of the plant is deemed safe with respect to radiological releases. See Rulemaking Hearing, note 8 supra, CLI-75-5, 1 N.R.C. at 280 (1975). (The obligations of Applicants and the Staff with respect to including an assessment of health effects in cost benefit analyses under 10 C.F.R. Part 51 are discussed in the next section.)

3. NEPA Cost Benefit Analysis

With respect to health effects, GANE alleges that Applicants underestimate "the human cost of the project in the cost-benefit analysis required by 10 C.F.R. 51.21, 51.20(b) and (c), and 51.23(a)."13/ GANE Supplement at 1. This statement

13/ A new 10 C.F.R. Part 51 has been promulgated, but has not yet taken effect. 49 Fed. Reg. 9352 (1984).

suggests that perhaps GANE is attempting to raise Applicants' alleged failure to adequately assess health effects as a NEPA issue.

Neither 10 C.F.R. § 51.20 (which governs Applicants' Construction Permit Stage Environmental Report) nor 10 C.F.R. § 51.21 (which governs Applicants' Operating License Stage Environmental Report) requires Applicants to attempt to quantify health effects. With respect to releases from normal plant operation, an applicart typically addresses radiological effects by providing release and dose estimates. 10 C.F.R. § 51.23, on the other hand, governs the Draft Environmental Statement, a document generated by the NRC Staff. Accordingly, there is no legal basis for GANE's implying that the environmental information that Applicants have reported is deficient.

10 C.F.R. § 51.23 does require the NRC Staff to consider in its cost-benefit analysis the radiological effects of the facility, and projected health effects are normally provided in the Staff's environmental statements. However, the scope of review at the operating license stage does not require reevaluation of environmental matters considered before the construction permit was issued unless there are sufficiently changed circumstances. Philadelphia Electric Company (Limerick Generating Station, Units 1 and 2), LBP-82-43A, 15 N.R.C. 1423, 1459, 1461 (1982), citing Calvert Cliff's Coordinating Committee, Inc. v. AEC, 449 F.2d 1109, 1128 (D.C. Cir. 1971).

Accordingly, absent allegations of significant, new information, the health effects associated with the operation of the Vogtle plant are not a proper issue in this proceeding. Cf. id. at 1461. See also Detroit Edison Company, et al. (Enrico Fermi Atomic Power Plant, Unit 2), LBP-79-1, 9 N.R.C. 73, 86 (1979).

GANE refers to no significant new information on health effects in general that would necessitate a further evaluation by the NRC Staff, much less to some significant new information peculiar to Vogtle's releases which might prompt a Staff reevaluation.14/ The Staff previously concluded that the health effects of radioactive releases from Vogtle were insignificant. CP-FES, § 10.1.2.5. In the Construction Permit proceeding, the Licensing Board concurred with this conclusion. Georgia Power Company (Alvin W. Vogtle Nuclear Plant, Units 1, 2, 3, and 4), LBP-74-39, 7 A.E.C. 895, 907-908 (1974). GANE's reference to J. Gofman, Poisoned Power (1971) (GANE Supplement

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14/ In Public Service Company of Oklahoma (Black Fox Station, with Units 1 and 2), CLI-80-31, 12 N.R.C. 264 (1980), the Commission permitted litigation of the health effects of ionizing radiation as a NEPA issue in a construction permit proceeding. However, the Commission stated that a full NEPA record is already in existence (from the 10 C.F.R. Part 50, Appendix I rulemaking proceeding) and "no useful purpose would be served by litigating health effects of radioactive releases when there is no serious contest as to the result." Accordingly, even in a construction permit proceeding, a petitioner must allege significant new scientific evidence belying the insignificance of routine releases.

at 2), for the assertion that embryos are more susceptible to leukemia when exposed to radiation is certainly not new information.

GANE also alleges, in its discussion of GANE-1, that Applicants have failed to assess the cumulative effects of Vogtle and Savannah River Plants, as required by 10 C.F.R. § 51.20(b) (applicable to the Construction Permit Stage Environmental Report). GANE Supplement at 1. This allegation is redundant of GANE-2, which deals solely with cumulative impacts; accordingly, it should be rejected pursuant to 10 C.F.R. § 2.714(e). Aside from its redundancy, this allegation in GANE-1 once again fails to address relevant documentation. Applicants extensively assessed the existing radiological burden, including the Savannah River Plant releases, in its Construction Permit Stage Environmental Report. CP-ER, §§ 2.8 et seq. Applicants then evaluated the combined population doses from Vogtle, the Savannah River Plant, and other nearby facilities. The combined dose was 0.05 percent of the population dose due to natural background. CP-ER, § 5.4.4. The NRC Staff also assessed the radiation background due to nearby nuclear facilities. CP-FES, § 2.8.2. The Staff concluded that the combined population dose from these facilities would be less than 0.2 percent of the dose from natural background radioactivity. Id., § 11.7.7. In GANE-1, GANE provides no new information or circumstances15/ that might change these

15/ The proposed reopening of the L-Reactor at the Savannah River Plant, raised as an issue in GANE-2, is addressed below in Applicants' response to that contention.

conclusions made at the construction permit stage.

4. Conclusion

For all of the foregoing reasons, GANE-1 should be rejected in its entirety.

B. GANE-2: Cumulative Effects

GANE-2 alleges that Applicants have failed to assess the combined environmental and health effects of Vogtle and the Savannah River Plant. GANE cites "10 C.F.R. 20.103, 50.34(a)(4), 51.23(b), 104, 105, 106, and 201." GANE Supplement at 3. 10 C.F.R. § 50.34(a)(4), which pertains to the preliminary safety analysis report, is irrelevant and is not further discussed. 104, 105, 106 and 201 presumably refer to 10 C.F.R. §§ 20.104, 20.105, 20.106, and 20.201.

1. Environmental Assessment

10 C.F.R. § 51.23(c), (not § 51.23(b) which GANE cites), requires the NRC Staff (not Applicants)16/ to assess "cumulative impacts as may reasonably appear significant." As discussed above with respect to GANE-1, the Staff performed such an assessment at the construction permit stage and concluded that the cumulative impact of the Vogtle and Savannah River Plants, as well as other nearby facilities, was

16/ Note, however, that Applicants' preoperational radiological monitoring program fully assessed the existing radiological burden in the vicinity of the site. See Applicants' OL-ER, § 6.4 et seq.

insignificant. The only supposedly new information offered by GANE to resurrect this issue is the proposed restart of the L-Reactor.17/ Since the Staff has not yet issued its operating license stage draft environmental statement, it is premature to allege the Staff's noncompliance. Nevertheless, Applicants submit that a supplemental assessment by the Staff of cumulative effects is unnecessary. The cumulative effects of the L-Reactor and Vogtle are already being fully evaluated by the Department of Energy in its environmental impact statement for the L-Reactor.18/ NEPA does not require federal agencies to conduct duplicative evaluations. As the Supreme Court has indicated, an agency can evaluate the environmental impact of the first of several proposals by itself, and leave evaluation of the cumulative effects of subsequent proposals to the environmental impact statements on those subsequent proposals. Kleppe v. Sierra Club, 427 U.S. 390, 415 n.26 (1976).

17/ The information Applicants provided in their Construction Permit Stage Environmental Report on gamma and beta radiation, and on iodine and tritium, from the Savannah River Plant included figures from six years in which the L-Reactor was in operation. CP-ER, § 5.5.

18/ See U.S. Department of Energy, "Draft Environmental Impact Statement, L-Reactor Operation, Savannah River Plant" § 5.2 (Sept. 1983). GANE does not address the evaluation of cumulative effects in this DEIS, although GANE is obviously aware of this evaluation. (GANE cites comments on the L-Reactor DEIS). Nor does GANE attempt to show that because of the releases from the L-Reactor, the effects attributable to Vogtle will be significant. Without such a showing, cumulative effects need not be evaluated. Westside Property Owners v. Schlessinger, 597 F.2d 1214, 1217 (9th Cir. 1979).

Here, at the construction permit stage of Vogtle, cumulative effects of Vogtle together with nearby facilities were evaluated. To the extent additional or different facilities are proposed for operation thereafter, Kleppe teaches that the need for an updated cumulative assessment may be considered in conjunction with that subsequent facility. DOE is apparently doing just that in its evaluation of the L-Reactor. To the extent GANE quarrels with that evaluation, its recourse is in that proceeding, not here in the Vogtle proceeding. This is particularly appropriate in the instant circumstances where the focus is on the L-Reactor, whose releases (the precise nature of which may be sensitive from a disclosure standpoint) and additions to cumulative effects are best explored in a proceeding involving its operator, DOE.

2. Safety Consequences

GANE-2 refers to assessment of health effects separate from assessment of environmental effects. GANE also references sections of 10 C.F.R. Part 20, Standards for Protection Against Radiation. Part 20 establishes maximum permissible concentrations of radioactive materials in air and water in restricted and unrestricted areas. It is possible, therefore, that GANE-2 is attempting to assert that the combined effects of the Savannah River Plant and Vogtle releases create a safety hazard in violation of these regulations. However, such vagueness in pleading denies Applicants reasonable notice of the issues against which they must defend.

Furthermore, if GANE is indeed attempting to raise a safety issue, it fails the basis requirement utterly. Applicants have calculated the releases and concentrations that will result from operation of the Vogtle plant. These estimates are located at FSAR, Table 11.3.3-3. GANE, however, ignores these figures.

Section 20.106(e) does authorize NRC to reduce the release concentrations otherwise allowable for Vogtle if it appears that because of radioactive releases from other facilities in the area the daily intake of radioactive material from all pathways of exposure by a suitable sample of an exposed population would exceed one-third of the daily intake resulting from exposure to concentration limits specified by Part 20. Savannah River reports annually the releases of radioactive materials from its facilities. Furthermore, the results of Applicants' preoperational monitoring program, reported in Applicants' OL-ER at §§ 6.4 et seq., necessarily include the contribution from the Savannah River Plant to the existing radiological burden in the vicinity of the plant. In addition, the DOE draft environmental impact statement prepared in connection with its proposal to reopen the L-reactor contains estimates of incremental releases from that facility. The Savannah River releases and estimates, which have also been ignored by Petitioners, are far below the levels which would invoke Section 20.106(e) for Vogtle.

In fine, GANE ignores both Applicants' estimated releases and the Savannah River Plant reported and estimated releases. It makes no attempt to address them. It identifies no radionuclide that will be produced by the two facilities in concentrations exceeding to 10 C.F.R. Part 20 limits (and none exists).

3. Conclusion

GANE-2 appears intended to raise as a NEPA issue the need to assess the cumulative effects of Vogtle and the L-Reactor. This contention lacks legal basis, since there is no requirement that the NRC duplicate the evaluation of cumulative effects performed by the Department of Energy. All other cumulative effects of the Savannah River Plant and Vogtle have been fully assessed, and GANE offers no basis to reconsider them.

It is also possible that GANE is attempting to raise a safety issue, but in this regard its pleadings are so vague and so lacking in basis that such an issue must be rejected.

C. GANE-3: Psychological Stress

GANE-3 states that Applicants have failed to show that "the fear caused by living adjacent to a nuclear facility will not threaten the security and well-being of the community, in violation of various laws and rules and regulations." GANE Supplement at 8. GANE provides no basis for its allegation

that operation of the Vogtle Plant in fact will induce fear and accompanying stress in the public. Nor does GANE identify the "laws and rules and regulations" to which it vaguely refers. Applicants are aware of no requirement that fear on the part of the public to operation of the Vogtle plant constitutes a basis for challenging plant operation. To the contrary, the Supreme Court recently has held that fear of the threat posed by a nuclear plant's operation is not cognizable under the National Environmental Policy Act. Metropolitan Edison Company v. People Against Nuclear Energy, 103 S.Ct. 1556, 1561 (1983); compare GANE-3 (alleging that "Childhood fears of nuclear catastrophe may last a lifetime and cause irreparable damage;" and "fears and anxieties will threaten the security and well-being of our community").

Moreover, in licensing nuclear power plants, absent evidence of a "unique and traumatic" nuclear accident in the vicinity of the plant, the Commission has instructed its licensing boards not to entertain psychological stress contentions. Consideration of Psychological Stress Issues; Policy Statement, 47 Fed. Reg. 31762 (1982). Cleveland Electric Illuminating Company, et al. (Perry Nuclear Power Plant, Units 1 & 2), LBP-82-69, 16 N.R.C. 751 (1982); Carolina Power & Light Company and North Carolina Eastern Municipal Power Agency (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 N.R.C. 2069, 2084, 2085, 2095, 2096 (1982). The Vogtle

operating license proceeding falls squarely within this rule.19/

D. GANE-4: Electromagnetic Radiation

GANE-4 alleges that Applicants have underestimated the risk of the electromagnetic radiation from transmission lines, in violation of 10 C.F.R. §§ 51.20, 51.21 and NEPA. As a basis, GANE argues that Applicants claim the effects of the transmission lines as a benefit, when in fact "growing scientific evidence" indicates a hazard.20/

GANE's assertion that Applicants claim the effects of transmission lines as a benefit is unfounded. GANE does not address Applicants' Environmental Reports, wherein Applicants considered the effects of the transmission lines. CP-ER, § 5.4.1.4; OL-ER, § 5.5 et seq. Instead, GANE refers to a company brochure and to other, unidentified Georgia Power Company

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19/ The Commission's Policy Statement was issued prior to the Supreme Court's reversal of the Court of Appeals' decision in the PANE case; consequently, the Policy Statement allows for consideration of psychological stress in the circumstances allowed by the Court of Appeals but subsequently reversed by the Supreme Court. See 47 Fed. Reg. at 31762-63.

20/ GANE also interjects, in its Supplement at 10, that Applicants provide "no assurance that [the] level [of the field intensity] will not be increased in the future." Aside from the fact that the transmission lines cannot be operated at voltages greater than that for which they were designed (and hence it would require the construction of a new transmission system to increase field intensity), Applicants submit that NEPA requires only an evaluation of the system as proposed.

environmental impact studies to support its assertion. GANE does not claim that such effects are claimed as a benefit in Applicants' Environmental Reports, and they are not. Therefore, this assertion is irrelevant.

GANE's reference to "growing scientific evidence" also lacks basis. GANE asserts that Applicants have not addressed "the most recent health risk data on effects of exposure to high voltage fields." GANE Supplement at 11. GANE refers to the testimony of a "Dr. Moreno" [actually "Marino"] in a proceeding before the New York Public Service Commission. The proceeding was held in 1976-1978 and was considered in Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (Susquehanna Steam Electric Station, Units 1 and 2), LBP-82-30, 15 N.R.C. 771 (1982). The Susquehanna Licensing Board considered the Marino testimony, but concluded that no evidence exists to date that the operation of 500kv power lines21/ will have an adverse biological health effect on humans. Id. at 790-93, 830-34. The Board noted that the New York Public Service Commission had also rejected the testimony. Id. at 792.

GANE also refers to an unpublished paper by Karl Z. Morgan, which he presented in Congressional hearings in 1978.

21/ Plant Vogtle's transmission lines are 500 kv or less. CP-ER, § 5.4.1.1.

(Attachment 1.) Professor Morgan propounds that low level ionizing radiation is dangerous. But transmission lines generate no ionizing radiation. With respect to non-ionizing radiation, Morgan merely cautions, as a passing note, that there may be effects of which we are unaware.22/ This paper is admittedly exclusively confined to Morgan's discussion of ionizing radiation and as such provides no basis for GANE's proposed contention on transmission lines.

Neither the Marino testimony nor the Morgan article presents significant new information necessary for reexamination of the environmental evaluations of transmission lines made at the construction permit stage,23/ and therefore they do

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22/ Professor Morgan's full and only statement on this point is as follows: "These non-ionizing radiations [sonic, ultrasonic, infrasonic, ultraviolet, infrared, microwave, radiofrequency, and long wave], like ionizing radiation, may be multiplying their presence so rapidly in the human environment that they too are resulting in health hazards in some areas before we are sufficiently aware of the magnitude of the problem."

23/ GANE also refers to a Colorado study, but does not identify it. Petitioners are probably referring to a 1976-1977 study in the Denver area, conducted by Nancy Werthheimer and Ed Leeper and published in the American Journal of Epidemiology in 1979. This study, however, related principally to high current configurations rather than to high voltage lines. While the study established no cause and effect relationship, Werthheimer and Leeper stated, "it was particularly homes close to . . . transforming points [substations and local transformers] that were over-represented among our cancer cases." Furthermore, the existing epidemiological evidence (including the Werthheimer/Leeper study) was considered by the Susquehanna Board. The Board found that "some tests do show results that

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not provide a sufficient basis for the contention.24/

Finally, GANE refers to a finding by Robert Helliwell of Stanford University that "electromagnetic radiation from the Canadian power system was being injected into the earth's magnetic ducts." GANE then hypothesizes that the interaction of this radiation with electrons will produce x-rays, which in turn will produce ultraviolet light, which in turn will produce skin cancer. GANE does not identify any source for Robert Helliwell's findings. Petitioners' allegation probably relates to a series of letters between Dr. Helliwell and Dr. Marino, appended to the Marino testimony in the New York Public Service Commission proceeding. (Attachment 2.) However, in these letters, Dr. Helliwell stated that he did not believe that

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could be interpreted as adverse, but these are so flawed that the results are inconclusive." Susquehanna, supra, LBP-82-30, 15 N.R.C. at 792. Accordingly, the Werthheimer/Leeper study is not significant new information that would warrant a supplemental evaluation of the effects of transmission lines.

Similarly, GANE refers to a conversation with H. Richard Payne of EPA, who purportedly stated that "evidence exists showing an effect" from exposure. GANE does not quantify this effect, does not indicate the duration of the exposure, does not reveal the evidence on which this conclusion was based, and does not even reveal the context in which the remark was made. Such vague and obscure references do not provide an adequate basis.

24/ Philadelphia Electric Company (Limerick Generating Station, Units 1 and 2), LBP-82-43A, 15 N.R.C. 1423, 1459, 1461 (1982); Detroit Edison Company, et al. (Enrico Fermi Atomic Power Plant, Unit 2), LBP-79-1, 9 N.R.C. 73, 86 (1979).

significant X-rays or ultraviolet radiation would be produced from power line radiation into the magnetosphere. Dr. Helliwell concluded "that the introduction of power lines capable of radiating high harmonics [necessary for power line radiation into the magnetosphere] should not alter the average intensity of any radiation that might reach the surface of the earth." Dr. Helliwell continued, "I see no reason to be concerned about the effects on our environment of power line radiation that enters the magnetosphere."

GANE makes no attempt to quantify and provides no basis for its absurd hypothesis that Vogtle transmission lines will increase ultraviolet radiation and increase the incidence of skin cancer. The environmental evaluation required by NEPA is subject to a rule of reason. Scientists' Institute for Public Information, Inc. v. AEC, 481 F.2d 1079, 1091-92 (D.C. Cir. 1973). It does not require an assessment of remote, speculative consequences. Life of the Land v. Brinegar, 485 F.2d 460, 472 (9th Cir. 1973), cert. denied, 416 U.S. 961 (1974); NRDC v. Morton, 458 F.2d 827, 837-38 (D.C. Cir. 1972), cited in Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 N.R.C. 2069, 2085 (1982).

For the reasons discussed above, GANE-4 should be rejected.

E. GANE-5/CPG-5: Seismicity

GANE-5/CPG-5 contends that Applicants have not properly assessed and considered the geology of the site, in light of new data made available by the U.S. Geological Survey. In their discussion of this contention, Petitioners refer to the "Millett fault," and each states, "Petitioner disputes applicant's claim that the fault is not capable." GANE Supplement at 12; CPG Supplement at 10. Petitioners also refer to the Charleston Earthquake and assert that "even if Applicant is correct that the Millett Earthquake fault is not capable--a premise Petitioner disputes--the area is of a similar geology to Charleston and therefore poses a risk of a devastating earthquake, perhaps as high as XII on the Mercalli scale." Id. at 13.

Petitioners make no attempt to address specifically Applicants' design and relevant analyses. Applicants have ignored neither seismic design and siting criteria nor the Charleston Earthquake. Applicants' extensive assessment of seismology and geology can be found in the FSAR at § 2.5 et seq. The information provided by Applicants, and ignored by Petitioners, is briefly summarized below.

1. The Millett Fault

The Millett fault was postulated by the U.S. Geological Survey in 1982 on the basis of groundwater and other indirect indications. Faye & Prowell, "Effects of Late Cretaceous and

Cenozoic Faulting on the Geology and Hydrology of the Coastal Plain Near the Savannah River, Georgia and South Carolina," USGS Open-File Report 82-156, at 73 (1982). See FSAR, § 2.5.1.2.3.1. Applicants promptly made further studies to determine the existence of this postulated fault, but found no evidence of its existence within the depths to which the investigation extended. FSAR, §§ 2.5.1.2.3.1, 2.5.2.3 citing Bechtel Power Corporation, "Studies of Postulated Millett Fault," Unpublished Report for Georgia Power Company (1982). Furthermore, even if the Millett fault did exist at a depth below which Applicants' investigation extended, it could not be capable by virtue of the age of the undisturbed overlying sediments. FSAR, § 2.5.1.2.3.1.

Petitioners state that they dispute Applicants' conclusion. Petitioners, however, give no reason for their dispute. Their unsupported skepticism does not constitute a basis. Nor is Petitioners' dispute with Applicants alone. The NRC endorsed Applicants' studies which concluded that the Millett Fault was not capable. NRC Staff Review of Vogtle Report "Studies of Postulated Millett Fault" (March 16, 1983). Moreover, a subsequent report by USGS (authored by Prowell, the geologist who was co-author of USGS open File Report 82-156), released after Applicants' studies were made available to USGS, does not list the Millett structure among all cretaceous and cenozoic age faults in the eastern United States. Prowell,

"Index of Faults of Cretaceous and Cenozoic Age in the Eastern United States," USGS Map MF-1289 (1983). In short, Petitioners have no basis for their bald assertion, and they simply present no information that would make a reasonable man inquire further as to the Millett Fault.

2. The Charleston Earthquake

Applicants' safe shutdown and operating basis earthquake were derived from the 1886 Intensity X Charleston earthquake. FSAR §§ 2.5.2.4 to 2.5.2.7. Studies, discussed in the FSAR, have defined a linear north-northwest trending zone of seismic activity, about 15 miles long and two miles wide, in the immediate Charleston-Summerville area. The closest approach of this Charleston-Summerville seismic zone to the site is 78 miles. Because of the historical clustering of earthquakes in this zone, because of the continuing microearthquakes there at levels higher than normal background in the southeastern United States, and because of the geologic features peculiar to that area, Applicants concluded that high activity earthquake clustering will remained confined to the Charleston-Summerville seismic zone. FSAR, §§ 2.5(c), 2.5.2.3.

Based on an earthquake of the same intensity as the Charleston earthquake but occurring only 78 miles from the plant, the maximum credible site intensity is VI-VII to VII on the Modified Mercalli Scale. FSAR, §§ 2.5.2.4, 2.5.2.6. However, for conservatism, a safe shutdown earthquake intensity of

VII-VIII was chosen for Plant Vogtle. Id. The seismic design bases were considered and approved at the construction permit stage by the Advisory Committee on Reactor Safeguards and by the U.S. Geological Survey. Safety Evaluation Report, Appendix, G, at G-1 (Supp. 1, May 1, 1974); id., Appendix H, at H-4. Seismic design was also considered and approved by the Licensing Board during the construction permit proceeding. Georgia Power Company (Alvin W. Vogtle Nuclear Plant, Units 1, 2, 3 and 4), LBP-74-39, 7 A.E.C. 895, 914 (1974).

The only support Petitioners offer in support of their assertion that there is a risk of an intensity XII earthquake25/ at the Vogtle site is a 1982 USGS letter stating in part that "no geologic structure or feature can be identified unequivocally as the source of the 1886 Charleston earthquake" (emphasis added). This statement is not new. In fact, USGS, in approving the seismic design criteria for the Vogtle plant at the construction permit stage, expressly recognized that because the structural features underlying the Coastal Plain are so poorly known, epicenters in the region cannot be related directly to appropriate geologic features. USGS went on to conclude:

25/ The Charleston earthquake was only of intensity X. See FSAR, § 2.5.2.1, at p. 2.5.2-4.

> The large earthquake of 1886 at Charleston, S.C., dominates the seismicity of the southeastern United States. The tectonic structures in the vicinity of this earthquake are not well defined, but there does appear to be a concentration of seismic activity in the basin south of the Cape Fear Arch and particularly in the Charleston, S.C., area. Available geological and seismological evidence indicates that the higher intensity earthquakes are localized along the deepest part of the axis of the basin. Therefore, it should be assumed that the Vogtle site could again experience ground vibration similar to or somewhat higher than that reported in 1886. It should be assumed, also, that moderate earthquakes, up to intensity VI (MM), similar to those which have occurred elsewhere in the Coastal Plain could occur in the general vicinity of the site.

CP-SER, Appendix H, at H-3 (Supp. 1, May 1, 1974).

The recent USGS statement is also consistent with Applicants' correlation of earthquake activity with geologic structures and tectonic provinces. <u>See</u> FSAR, § 2.5.2.3. It does not support Petitioners' assertion that the geology of the Vogtle area is of a similar geology to Charleston. The USGS was not addressing the geology of the Vogtle site (as it had done dispositively at the construction permit stage); rather, it was addressing the remote possibility that an earthquake similar to the Charleston earthquake could occur anywhere on the eastern seaboard. This is a generic issue relating to every nuclear power plant in the eastern United States. To the extent this observation relates to Vogtle, it is a generic issue being addressed by the NRC. Furthermore, until studies have been completed, it is the NRC's position that for the

licensing of nuclear power plants, the Charleston earthquake should be restricted to the immediate Charleston vicinity. Memorandum from Richard H. Vollmer, Director, Division of Engineering to Harold R. Denton, Director, Office of Nuclear Reactor Regulation, Enclosure 2 (March 2, 1983)(Division of Engineering Geoscience Plan to Address USGS Clarification Relating to Seismic Design Earthquakes in the Eastern Seaboard of the United States).

3. Conclusion

With respect to the postulated Millett fault, Petitioners baldy assert that they "dispute" Applicants' conclusion. With respect to the Charleston earthquake, Petitioners misapply a statement in a USGS letter. The statement does not support Petitioners' assertion that the Vogtle area is of similar geology to Charleston. Instead, it addresses a generic issue--an issue, however, that was recognized in Applicants' analyses and USGS's prior approval of the Vogtle seismic design criteria. Neither of these assertions is material; neither casts reasonable doubt on Applicants' proposal and analyses; neither is an adequate basis for the contention. Accordingly, GANE-5/CPG-5 should be rejected.

F. GANE-6/CPG-6: Pressurized Thermal Shock

Petitioners contend that Applicants cannot guarantee the safe operation of the reactor for the life of the plant due to unresolved questions of thermal shock effects on irradiated

reactor vessels. GANE Supplement at 13; CPG Supplement 11. This issue is a generic issue, Unresolved Safety Issue (USI) A-49, "Pressurized Thermal Shock." See NUREG-0606, "Unresolved Safety Issues Summary" (Vol. 6, No. 1, Feb. 17, 1984) at 42. It is the subject of an NRC Task Action Plan. See NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants" (Feb. 1980).

An Unresolved Safety Issue may be raised in a licensing proceeding. However, the mere identification of an Unresolved Safety Issue is not a sufficient contention. A petitioner who wishes to litigate an Unresolved Safety Issue must establish a nexus between the license application and the task action plan by showing both:

> (1) that the undertaken or contemplated project has safety significance insofar as the reactor under review is concerned, and
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> (2) that the fashion in which the application deals with the matter in question is unsatisfactory, that because of the failure to consider a particular item there has been insufficient assessment of a specific type of risk for the reactor, or that the short-term solution offered in the application to a problem under Staff study is inadequate.

Gulf States Utilities Company (River Bend Station, Units 1 and 2, ALAB-444, 6 N.R.C. 760, 773 (1977).

Petitioners make no attempt to demonstrate the significance of USI A-49 to Vogtle. Instead, they discuss the Rancho Seco Reactor. They do indicate that Applicants have described

the copper and phosphorous levels in the reactor vessel (FSAR, § 5.3.3), but do not allege that these would have an adverse effect; and as indicated in the FSAR, the effect of fluence and copper content will be factored into the Technical Specification pressure-temperature limits for the reactor vessel. FSAR, § 5.3.2. Applicants, in demonstrating compliance with 10 C.F.R. Part 50, Appendices G and H, considered the copper, nickel, and phospherous content of the reactor vessel components in their calculations of end-of-life reference nil ductility temperatures. The calculated end-of-life reference nil ductility temperatures are significantly less than that at which operation of the plant would not be permitted under NRC criteria. Special core designs were not used in the calculations and are not necessary. FSAR, Q251.1. Similarly, Petitioners do not address the design of the Vogtle reactor vessels, described in the FSAR, § 5.3.3.1; nor do they address Applicants' Material Surveillance program (FSAR, § 5.3.1.6).

Petitioners also make no attempt to address the second prong of the River Bend test. They do not address the Vogtle application, they do not specify a "type of risk" that has not been assessed, and they do not address short-term solutions. In this regard, the potential for a PTS event occurs only after the reactor has been operating a substantial period of time (i.e., after the vessel has lost its fracture toughness properties and is embrittled by neutron radiation).26/ There is,

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26/ The probability of such an event is also dependent upon a variety of other plant-specific factors, such as the particular

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therefore, no short-term hazard. See Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 N.R.C. 245, 248 n.6 (1978).

Accordingly, Petitioners provide no basis for their contention.

There is a second reason why this contention should not be raised in this proceeding. The Commission recently published a proposed rule which would require certain actions to be taken by applicants and licensees to identify any corrective actions that might be required to prevent or mitigate potential PTS events. Proposed Rule, "Analysis of Potential Pressurized Thermal Shock Events," 49 Fed. Reg. 4498 (1984). Accordingly, because this matter is being fully addressed by the Commission in a generic rulemaking proceeding, it is not an appropriate matter for litigation in this proceeding.

G. GANE-7/CPG-7: Groundwater

GANE-7/CPG-7 contends that Applicants have not adequately assessed "the value" of the groundwater below the Vogtle site. GANE-7/CPG-7 also asserts that Applicants fail "to provide adequate assurance that the groundwater will not be contaminated."

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material composition of the vessel and vessel welds and the presence of flaws which are capable of propagation. See NUREG-0606, supra, at 42.

Petitioners cite 10 C.F.R. §§ 51.20(a)-(c), 50.34(a)(1), and 100.10(c)(3). GANE Supplement at 14-15; CPG Supplement at 12-13.

This contention appears to raise two distinct issues. The first part of the contention, addressing Applicants' assessment of the value of groundwater, raises an environmental issue, and Petitioners' citation to 10 C.F.R. § 51.20(a)-(c) presumably relates to this issue. The second part of the contention, addressing Applicants' alleged failure to provide adequate assurance that groundwater will not be contaminated, appears to raise a health and safety issue, and presumably corresponds to Petitioners' citation to 10 C.F.R. §§ 50.34(a)(1) and 100.10(c)(3).

1. Environmental Consideration of Groundwater

Petitioners again ignore information provided in Applicants' Environmental Reports. Applicants conducted an extensive investigation to define the characteristics of the groundwater at the site and to determine the significance of accidental spillage. This assessment is documented in Applicants' Construction Permit and Operating License Stage Environmental Reports. CP-ER, § 2.5.4; OL-ER § 7A.4. Two aquifers exist at the site; a shallow water table aquifer and the deep, confined Tuscaloosa aquifer.27/ Furthermore, a relatively

27/ There is actually a third aquifer in the region, the principal artesian (or upper confined) aquifer. However, at the

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impermeable 60- to 70-foot thick marl forms an aquiclude between the two aquifers and makes contamination of the confined Tuscaloosa aquifer particularly unlikely. Id. at p. 2.5-9. See also OL-ER, § 2.1.3.8.1.2. Thus, only the water table aquifer likely would be affected by an accidental release, and this aquifer lies largely within the exclusion radius of the plant. Water movement indicates that spillage at the plant site would eventually find its way to Mathes Pond (also known as Mallard Pond), where it could be intercepted. CP-ER § 2.5.4, at p. 2.5-18. Moreover, the time of migration of a spill to Mathes Pond would be controlled by the permeabilities of the soil and is estimated to be on the order of 350 years. Id., § 5.4.3.28/ Based on this assessment, Applicants concluded that there will be no significant impact on groundwater even in the unlikely event that a spill or other accidental release to the groundwater occurred. Id., § 11.2, and Attachment A at p. 11-A-3. Because the environmental impact is insignificant, there is no need to assess the "value" of the groundwater.

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Vogtle site, the two confined aquifers (the Tuscaloosa and principal artesian aquifers) are not isolated from each other hydraulically, and are simply referred to as the Tuscaloosa aquifer in the CP-ER.

28/ The specific calculation is given in the FSAR at § 2.4.13.1.

Petitioners do not address Applicants' analysis or even acknowledge its existence. Petitioners do not specify how this analysis is inadequate. They merely assert that the groundwater aquifer could become contaminated and perhaps so too could the Tuscaloosa aquifer.29/ They proffer no significant new information necessary for a contention addressing environmental matters previously evaluated at the construction permit stage. See cases cited in note 24, supra. Furthermore, Applicants have assessed the remoteness of such contamination, and Petitioners provide no grounds to dispute this. Accordingly, there is no basis to doubt Applicants' evaluation of the environmental effect on groundwater; and there is no basis, legal or factual, for Petitioners' contention that Applicants have inadequately assessed the value of groundwater. GANE-7/CPG-7, as it relates to environmental considerations, lacks basis and specificity, and it should be rejected.

29/ GANE does state that, "It is known that just south of the plant site, this clay changes into a limeston . . . ." GANE does not explain the relevance of this remark, but perhaps attempts to imply that seepage from the water table aquifer at the plant can find its way to the confined aquifer. However, the water table aquifer is isolated on an interfluvial high and is intercepted by Beaverdam Creek. CP-ER, § 2.5.4. The limestone formation actually occurs several miles to the souteast (FSAR, § 2.4.12.1.1.2) and outside the area intercepted by Beaverdam Creek. GANE's vague implication is simply irrelevant. Similarly, Petitioners' unsupported assertion that the 50-foot hydraulic head on the water table aquifer will push it through fractures in the marl is unfounded. This pressure differential demonstrates the effectiveness of the marl as an aquiclude.

2. Safety Considerations with Respect to Groundwater

Petitioners refer to 10 C.F.R. § 50.34(a)(1) and 10 C.F.R. § 100.10(c)(3) in support of their assertion that Applicants fail "to provide adequate assurance that groundwater will not be contaminated." GANE Supplement at 15; CPG Supplement at 13. However, neither of these sections nor any other section explicitly requires Applicants to provide adequate assurance that groundwater will not be contaminated.30/ What is relevant is whether Vogtle has been properly built and whether there is reasonable assurance that it can be operated without endangering the health and safety of the public. See 10 C.F.R. § 2.104(c).

Petitioners do not allege any design or construction deficiency. They do not even allege that there is a real hazard. They do not say how Applicants' radwaste program is inadequate or indicate any mechanism whereby a significant spill might occur. Applicants have described their liquid waste management system (FSAR, § 11.2) and have demonstrated that routine releases will not exceed permissible levels. Applicants have also assessed the possibility of accidental releases. FSAR,

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30/ 10 C.F.R. § 50.34(a)(1) applies to an applicant's preliminary safety evaluation report and site evaluation factors. Similarly, 10 C.F.R. § 100.10(c)(3) governs site evaluation. But Applicants' construction permit is not at issue in this proceeding. Cf. Duquesne Light Company et al. (Beaver Valley Power Station, Unit 2), LBP-84-6, 19 N.R.C. ____, slip op. at 20 (Jan. 27, 1984).

§ 15.7. See also OL-ER, §§ 7.1, 7A. It is Applicants' demonstrated compliance with the Commission's regulations that provides reasonable assurance that Vogtle can be operated safely. Petitioners do not dispute this compliance or suggest any scenario, credible or otherwise, that might question the efficacy of such compliance. GANE-7/CPG-7, as it relates to the reasonable assurance that Vogtle can be operated safely, is devoid of specificity and basis, and it should be rejected.

H. GANE-8/CPG-8 Quality Assurance

GANE-8/CPG-8 contends that Applicants have failed to enforce an adequate construction quality assurance program, as required by 10 C.F.R. Part 50, Appendix B. In their discussion of this contention, Petitioners refer to a few NRC documents and a number of Applicants' files addressing several discrete topics. These references indicate that occasionally a minor problem in construction or quality assurance has been identified. But Petitioners do not contend that identified problems have not been promptly and adequately remedied. Instead, Petitioners unsoundly infer broad programmatic deficiencies and unfairly impugn Applicants' dedication to quality. Petitioners conclude that Vogtle "mirrors similar situations at the Zimmer Nuclear Plant and the Byron Nuclear Plant," and as a result, "Plant integrity cannot be assured." GANE Supplement at 21, CPG Supplement at 19.

In determining relevance and materiality, a Licensing Board must view QA allegations in the context of the enormity of nuclear power plant construction. As the Appeal Board has indicated:

> In any project even remotely approaching in magnitude and complexity the erection of a nuclear power plant, there inevitably will be some construction defects tied to quality assurance lapses. It would therefore be totally unreasonable to hinge the grant of an NRC operating license upon a demonstration of error-free construction. Nor is such a result mandated by either the Atomic Energy Act of 1954, as amended, or the Commission's implementing regulations. What they require is simply a finding of reasonable assurance that, as built, the facility can and will be operated without endangering the public health and safety. 42 U.S.C. §§ 2133(d), 2232(a); 10 C.F.R. § 50.57(a)(3)(i). Thus, in examining claims of quality assurance deficiencies, one must look to the implication of those deficiencies in terms of safe plant operation.

Union Electric Company (Callaway Plant, Unit 1), ALAB-740, 18 N.R.C. 343, 346 (1983) (footnote omitted). See also Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-756, 18 N.R.C. _____, slip op. at 7 (December 19, 1983). Beyond this customary scrutiny, there may remain a question whether there has been a breakdown of quality assurance "of such dimensions to raise legitimate doubt as to the overall integrity of the facility." Callaway, supra, ALAB-740, 18 N.R.C. at 346.

In this context, an average number of relatively innocuous deficiencies does not provide a basis for a contention alleging

a general breakdown in quality assurance. They do not reasonably imply a "pervasive failure to carry out the quality assurance program" which might stand in the way of the requisite safety finding. Id. at 346.31/

The illogic and utter lack of basis for Petitioners' conclusion that Vogtle is another Byron or Zimmer is evident from the very documents they cite.32/ The inferences Petitioners draw from each are unfounded. Applicants have attached the very documents to which Petitioners refer and as discussed below, these documents demonstrate that Petitioners' allegations lack factual basis.

31/ The materiality of allegations forming the basis of a QA contention is vital to observe the intent of the Commission's basis with specificity requirement. Because of the size and complexity of nuclear power plant construction, and the rigor of the NRC's and applicants' own inspection programs, no nuclear power plant can be constructed without a certain number of nonconformances. If these nonconformances automatically form the basis for litigation of an applicant's entire QA program, the integrity of the plant, and applicant's character, then a contention of unparalleled breadth could be raised in every licensing proceeding without regard to the purposes of the basis with specificity requirement, as enumerated in Peach Bottom, supra, ALAB-216, 8 A.E.C. at 20-21.

32/ The illogic of Petitioners' conclusion is also revealed by the Commission's April 4, 1984 Report to Congress, submitted pursuant to Section 13 of the NRC Authorization Act For Fiscal Years 1982 and 1983 (P.L. 97-415) (referred to as the Ford Amendment). The Commission cited Vogtle as an example of a nuclear power plant construction project with a successful quality assurance program -- a project which has experienced no major problems.

1. Welding.

Petitioners assert that "repeated violations of NRC regulations by Applicant in the constiuction methods applied to pipe fittings and welds must be interpreted as undermining confidence in the capability of the coolant and containment systems to perform their essential tasks." GANE Supplement at 17; CPG Supplement at 15. But Petitioners do not allege that there have been "repeated violations involving construction methods applied to welds and pipe fittings." Instead, they state that "potential deficiencies involving welds were raised as an issue at least as early as April 29, 1981" and they cite "I&E file #X7BG03-M18." The "I&E" prefix to this citation is incorrect and misleading; file X7BG03-M18 is not an NRC file, but one created by Applicants. (Attachment 3.) As Petitioners ignore or misunderstand, the contents of this file were actually the result of Applicants' inspections and endorse the comprehensiveness of Applicants' quality assurance program. Applicants conducted inspection of welds on containment liner penetrations -- work performed by Chicago Bridge and Iron (CB&I) in its facilities in Birmingham. These inspections revealed some problems associated with weld cleanup and some conditions (such as slag, porosity, arc strikes) that should have been eliminated prior to testing. Applicants therefore issued a nonconformance report and reported the matter to the Commission. The discrepancies were found not to affect adversely the plant's

structural integrity, and the work was subsequently approved for use-as-is. There was no NRC inspection involved and no violation. This file demonstrates the effectiveness of Applicants' QA program.

2. Inspection/Testing

Having failed to allege facts that would even remotely call into question Applicants' weld construction methods, Petitioners, in mid-sentence, change the subject. Petitioners continue, stating, "problems involving the appropriate inspection of welds have occurred as recently as September 1983." GANE Supplement at 17; CPG Supplement at 15. Petitioners refer to I&E Report 83-15, Applicants' file X7BG10, and "IR 50-424 and 50-425" (presumably I&E Report 83-18).

As recounted in I&E Report 83-15 (Attachment 4), an NRC inspection discovered that the process sheet for examination of one coolant pressure boundary weld on isometric drawing IKA-1201-119-02, Rev. 10, failed to specify performance of a required examination. Applicants promptly investigated the matter and found that the piping contractor's engineer had prepared the wrong process sheets for welds on the isometric drawing in question, but had prepared the correct sheets for the welds on all other primary loop isometrics (20 drawings in all). Applicants acknowledged the violation, which the NRC classified as severity level IV.33/ File X7BG10, Log GN-288,

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33/ For facility construction, 10 C.F.R. Part 2, Appendix C, Supplement II describes five severity levels. A severity level

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Letter from D. Foster to the Office of Inspection and Enforcement, USNRC (November 28, 1984). (Attachment 5.)

In I&E Report 83-18 (Attachment 6), the NRC issued a notice of severity level V violation ("of minor safety or environmental significance"). An NRC inspector had examined the Reactor Closure Heads Internal Cladding, had observed questionable surface conditions, and concluded that these conditions did not appear to meet the criteria for performing a meaningful liquid penetrant examination. Reexamination of two suspect areas of the cladding by liquid penetrant methods revealed linear and rounded indications greater than acceptable limits. Applicants determined, however, that the reactor vessel manufacturer had properly performed the examination, and that the indications were apparently caused by gritblasting, which had been used as a form of post-examination cleaning.34/ The

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I violation is one that involves a structure or system that is completed in such a manner that it would not have satisfied its intended safety related purpose. A severity level II violation is one involving a breakdown in the quality assurance program or the completion of a structure or component in such a manner that it could have adverse effect on the safety of operations. A severity level III violation involves a QA deficiency related to a single work activity. Severity level IV is a violation not amounting to levels I through III (and therefore not indicative of a programmatic or systemwide problem or a problem that could have an adverse effect on the safety of operations). A severity level V violation is one having minor safety or environmental significance. Applicants have never received a severity level I, II, or III violation.

34/ File X7BG10, Log GN-295, Letter from D. Foster to I&E (Dec. 14, 1983). (Attachment 7.) Gritblasting as a method of

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entire cladding surface of the closure heads of both units was subsequently cleaned, repaired where necessary, and liquid penetrant examined.

3. Implementation of Procedures

Petitioners next proclaim that "Any quality assurance program predicated on implementation of dictated procedures without due regard to the exercise of critical judgement and standards of professional practice must be considered woefully inadequate." GANE Supplement at 17-18; CPG Supplement at 15-16. With this peroration as predicate, Petitioners refer to "IR 52 50-424 Appendix A Report Details." They are probably attempting to refer to I&E Report 83-16.

In I&E Report 83-16 (Attachment 8), the NRC issued another notice of severity level V violation. CB&I's Radiographic Examination Procedure for welds did not specify that the heat-affected zone of a weld should be included in the area of interest for radiography of welds, to which film density requirements applied. Applicants disputed the violation. Applicants did not deny that the heat-affected zone of a weld should be given proper consideration during inspections, but

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post-examination cleaning was in accordance with paragraph NB-5113 of the ASME B&PV Code, Section III. It was not, as Petitioners allege, a departure from standard procedure, and the results were entirely unforeseen. Id.

with respect to radiography procedures, Applicants observed that the pertinent ASME Code does not require the heat-affected zone to be included in the area of interest. The radiography density requirement is restricted to the weld metal. File X7BG10, Log GN-263, Letter from D. Foster to Office of Inspection and Enforcement, USNRC (Sept. 30, 1983). (Attachment 9.)

The dispute was mooted, however, because the area in question was removed to allow installation of a penetration assembly, and the NRC withdrew the violation. Letter from R. Lewis to R. Kelly (Dec. 22, 1983). (Attachment 10.)

Petitioners criticize Applicants' defense of the CB&I Radiography procedures as "erroneously equat[ing] methods of quality assurance with simple compliance to written procedures." GANE Supplement at 18; CPG Supplement at 16. But it is this statement which evinces a misunderstanding of QA procedures. Testing procedures are intended to be explicit and complete; and the ASME Codes, many sections of which have been incorporated into the NRC's regulations (see 10 C.F.R. § 50.55), establish the standards of professional practice.

Petitioners also refer to a "related matter" in I&E Report 82-29 (Attachment 11). GANE Supplement at 18; CPG Supplement at 16. In this report, an NRC inspector suggested the need to clarify welding procedures. This suggestion stemmed from a discussion between the inspector, a welding supervisor, and a site QA supervisor as to whether welding could continue in a

"very light misting rain." (No violation was involved). The discussion was merely the result of imprecision in the written procedures, which left room for differing professional judgments.

4. Procurement

Petitioners next assert that "[in] addition to these procedural aspects of quality assurance [i.e., the matters discussed above], there are other questions involving the applicant's 'controlling the quality of the . . . component or system to predetermined requirements.'" GANE Supplement at 18-19; CPG Supplement at 16-17. Petitioners allege that the Applicants' diesel generators,35/ manufactured by Transamerica Delaval, have had several problems involving the governor lube oil cooler assembly, the air valve assembly, piston skirts, and certain cables. A potential problem involving the governor lube oil cooler assemblies was determined to be inapplicable to the Vogtle diesel generators, however, because of the way in which those assemblies were installed. With respect to the air valve assembly, Transamerica Delaval notified the NRC in May 1982 that the capscrew that secures this assembly might be 1/4" too long and should be replaced with a shorter screw. File X7BG03-M29: Letter from R. Boyer, Transamerica Delaval, to

35/ Petitioners incorrectly refer to the diesel generators as standby steam generators.

Director, Office of Inspection and Enforcement, USNRC (May 13, 1982); Log GN-186, Letter from D. Foster to Office of Inspection and Enforcement, USNRC (Aug. 6, 1982). (Attachment 12.) Transamerica Delaval also notified Applicants in October 1982 that potentially defective piston skirts might have been installed, but Transamerica Delaval committed to test, and repair or replace where necessary, all piston skirts. File X7BG03-M36: Letter from R. Boyer, Transamerica Delaval, to Director, Office of Inspection and Enforcement, USNRC (Oct. 28, 1982); Log GN-219, Letter from D. Foster to Office of Inspection and Enforcement, USNRC (March 1, 1983). (Attachment 13.) Finally, in November 1982, Transamerica Delaval reported that certain cable in the diesel generators had recently failed the IEEE-383 insulation flame test; accordingly: this cable is being replaced. File X7BG03-M49: Letter from R. Boyer, Transamerica Delaval, to Director, Office of Inspection and Enforcement (Sept. 27, 1983); Log GN-274, Letter from D. Foster to Office of Inspection and Enforcement, USNRC (Nov. 3, 1983). (Attachment 14.)

Petitioners conclude that "[by] failing to make a general assessment of the suitability of the TD diesel generator system for such an extremely important emergency function, the applicant has brought its own quality control capabilities into question, undermining confidence in the safe functioning of its operating plant in direct contradiction to NRC QA

requirements." GANE Supplement at 19; CPG Supplement at 17. However, Petitioners are not disputing that such an assessment was made, but are criticizing Applicants for lack of precognition. Petitioners' inference is unsupported and unreasonable.

Petitioners also address subcontractor quality assurance and refer to an August 22, 1983, meeting on this subject.36/ Petitioners state that a topic of this meeting was job intimidation of quality control workers. However, as the letter which Petitioners cite clearly indicates, this meeting was held at Applicants' request to discuss with the NRC the results of three separate subjects related to investigations conducted by Applicants. Applicants had investigated whether the salary administration practices and personnel policies of Pullman Power Products could have a detrimental effect on the attitude of quality control personnel with regard to the quality of workmanship. Applicants had also investigated allegations made to them by a Walsh Company boilermaker. Finally Applicants had reinspected over 35,000 welds in piping spool pieces manufactured by Pullman Power Products, and had issued nonconformance

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36/ In their supplements, Petitioners interject a reference to I&E Report 83-04, which has nothing to do with procurement. This report discusses an incident in which winds damaged weather protection for freshly poured concrete and responsible personnel were not promptly notified. The NRC issued a severity level V violation. Procedures were changed to make sure this would not happen again. File X7BG10, Log GN-222, Letter from D. Foster to Office of Inspection and Enforcement, USNRC (March 22, 1983). (Attachment 15.)

reports for defects found.37/ See Letter from J. O'Reilly, Regional Administrator, USNRC, to D. Foster (Sept. 28, 1983). (Attachment 16.) Again, these investigations are supportive indications of a comprehensive QA program, and do not support a contention which alleges fault in a QA program.

Petitioners then assert that "[p]rocurement failures continue after numerous I&E Bulletins from past QA/QC inaction." GANE Supplement at 20; CPG Supplement at 18. However, the only I&E Bulletin which Petitioners mention as being relevant to procurement is I&E Bulletin 83-06 (see Attachment 17), a Bulletin addressed to every nuclear power plant operating or under construction. This one Bulletin certainly does not constitute "numerous I&E Bulletins" and has nothing to do with QA/QC inaction on the part of Applicants and their subcontractors.

37/ With respect to Applicants' reinspection of the spool welds, the NRC's most recent Systematic Assessment of Licensee Performance (SALP) Report, which Petitioners cite, stated:

> Another activity relating to piping systems was the reinspection of approximately 15,000 piping spool pieces that had been fabricated by Pullman Power Products. The reinspection was needed to ascertain the acceptability of fabrication welds after code rejectable deficiencies had been found in a sampling of spool pieces stored at the plant site. The extensive reinspection program was handled in a thorough manner, and resulted in the satisfactory resolution of a generic quality problem.

I&E Report 84-01 (Attachment 28) at 30. The SALP Report is discussed further below.

Furthermore, Petitioners do not cite a single I&E Report even suggesting a deficiency with Applicants' procurement program. Instead, Petitioners list several letters in which Applicants reported to the NRC potential deficiencies in components: certain relay cards (Attachment 18), trip breakers (Attachment 19), valve operators (Attachment 20), and cable terminations (Attachment 21), and Boveri Electric Inc. circuit breakers (Attachment 22).38/ All these potential deficiencies were discovered by Applicants or the manufacturer and corrective action addressed. Petitioners' references to these matters are in fact testimony to an effective quality assurance program and prompt, full disclosure. There is simply no negative implication.

5. Completeness of the Vogtle QA Program

Petitioners next contend that "repeated questions" have been raised about changes in the Vogtle QA program and that "there are questions" about the completeness of the Vogtle QA program. GANE Supplement at 20; CPG Supplement at 18. They refer to a Staff request in 1982 for additional information about a change in Applicants' QA program. (See Attachment 24.) Applicants had proposed a Field Change Notice (FCN) procedure

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38/ Petitioners also refer to "AKR-30 and AKR-50" circuit breakers. Presumably, Petitioners intend to refer to General Electric type AK-2 circuit breakers (see Attachment 23), but these are not used in safety-related systems at Vogtle.

to supplement the customary Field Change Request (FCR) procedure. The FCN procedure permitted, under certain conditions, the commencement of work relating to a design change before final engineering approval of the change. Because this was a new procedure, the NRC asked several questions before approving the procedure.

Petitioners also refer to a November 29, 1983 NRC Memorandum For Assigned Reviews For the Vogtle Electric Generating Plant. (Attachment 25.) As a step in the NRC's review of Applicants' FSAR, assigned reviewers were asked to check Applicants' equipment qualification list and assure that all pertinent equipment within each reviewer's area of responsibility had been included.

These are simply examples of normal NRC review practices. Such absolutely routine correspondence certainly does not support the allegations of "repeated questions" about changes and existing questions concerning the completeness of the Vogtle QA program.

6. Systematic Assessment of Licensee Performance

Finally, and ironically, Petitioners refer to the contents of the Systematic Assessment of Licensee Performance (SALP) Board Assessment," an annual inspection report issued by the NRC Staff.39/ First, they state that "Four allegations made by

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39/ Petitioners also refer to a 1982 arrest report (the result of an investigation initiated by Applicants), and assert that

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a former employee pertain to inadequate concrete QC testing and falsification of concrete QC test records." GANE Supplement at 21; CPG Supplement at 19. Petitioners offer no citation, but it is clear they are quoting from I&E Report 83-06 (the July 1981-October 1982 SALP). (Attachment 27.) This SALP Report concluded that Applicants' program in this area was in category 1 and stated:40/

> [A]n investigation was performed, by a regional inspector and an investigator of four allegations made by a former employee pertaining to inadequate concrete QC testing and falsification of concrete QC records. Two allegations were not substantiated. The remaining two allegations were partially substantiated. However, the licensee's QA program had detected and corrected the problems prior to the investigation. During the investigation, one violation was identified. . . . This violation was not associated with any of the allegations but was identified during review of concrete records. The licensee was cooperative with NRC investigators.

I&E Report 83-06, at 29.

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drug and alcohol use continues to take place on the site. Petitioners provide no basis for this charge, nor do they suggest how it related to any of the other assorted topics addressed in this contention. (Attachment 26.)

40/ Category 1 is the highest performance level and is defined as follows: "Reduced NRC attention may be appropriate Licensee management attention are aggressive and oriented toward nuclear safety; licensee resources are ample and effectively used such that a high level of performance with respect to operationsl safety or construction is being achieved." I&E Report 83-06, at 1.

Petitioners state that four other violations are described in the SALP. The first mentioned by Petitioners is a $40,000 civil penalty for "valve mispositioning." However, no such violation and penalty is described in the Vogtle SALP. No such violation has ever occurred at Vogtle (or could occur at a plant under construction). In fact, no civil penalty has ever been assessed against Vogtle. Second, Petitioners state "testing procedures have identified discrepancies involving cadwell [sic] operators." Again, no citation is given, but in I&E Report 83-06, the Staff identified a "deviation [not a violation] from the PSAR commitments regarding the failure to sample and test cadwelds in accordance with the recommendations of Regulatory Guide 1.10." I&E Report 83-06, at 28. The Staff noted that Applicants were responsive in correcting the deviation (id.), and took the deviation into account in determining Applicants' Category 1 performance ranking. Id. at 29.

Petitioners state that "[P]rotection of equipment procedures have been neglected." GANE Supplement at 21; CPG Supplement at 19. This was a severity level V violation noted in I&E Report 83-06 at 34. Petitioners also refer to "failure to establish adequate radiography procedures." Applicants know of no description in either of the two Vogtle SALPs of a failure to establish adequate radiography procedures, although a severity level V violation for improperly storing radiographic records was discussed in I&E Report 84-01, the November

1982-October 1983 SALP (Attachment 28). I&E Report 84-01, at 39. The NRC's discussion of Applicants' radiography program and practices belies Petitioners' reliance on the SALP. I&E Report 84-01 stated:

> The licensee has demonstrated sound technical decision making commensurate with quality assurance concerns. This was best exemplified by the licensee response to IE Bulletin 82-01. This bulletin pertained to problems where two vendors knowingly supplied altered radiographs. The licensee expanded the scope of the bulletin by performing an independent review of the radiographs for all of the shop fabricated welds for the components furnished by six other vendors. This inspection has identified numerous nonconformances with these radiographs. The licensee was working with the vendors to resolve the issue. This action was typical for the Vogtle project, where the specifics of a problem were expanded in a generic fashion to assure that a problem did not exist in related areas.

Id.

The irony in Petitioners' reference to the SALP is particularly apparent by review of the overall conclusions of the NRC. The NRC examined all I&E Reports relevant to Vogtle and issued since October 1981 (which includes all I&E Reports cited by Petitioners). The NRC, in the principal source proffered by Petitioners as the basis for their contention, found no indication of a programmatic breakdown. I&E Report 83-06, at 3; I&E Report 84-01, at 3. In I&E Report 83-06, the Staff concluded:

> GPC has developed, and is implementing, a vigorous construction project management effort with well qualified and experienced personnel. Major strengths were noted in the areas of containment and other safety related structures, licensing activities, and quality assurance program.

I&E Report 83-06, at 3. And in I&E Report 84-01, the Staff concluded:

> The licensee continues to implement a vigorous construction project management effort with well qualified and experienced personnel. Major strengths were noted in the areas of safety related components and the quality assurance program. No major weaknesses were identified.

I&E Report 84-01, at 3.

Conclusion

Petitioners' purported synopsis of Applicants' QA program is at very best hyperbole, and their inferences and conclusions are absolutely unfounded. Applicants attach the very documents and files Petitioners refer to as a basis for their contention. Even a cursory review of these documents reveals that they provide no basis whatsoever for a contention attacking the sufficiency of Applicants' entire QA program.

I. GANE-9/CPG-9: Design Description

In GANE-9/CPG-9, Petitioners contend that Applicants have failed to include in the PSAR and FSAR an adequate discussion of novel design features. The only specific advanced by Petitioners is Applicants' alleged failure to adequately explain a recent proposal to eliminate the need to postulate longitudinal and circumferential pipe breaks in the Vogtle reactor coolant systems primary loop. See GANE Supplement at 22; CPG Supplement at 20.

This proposal does not constitute a basis for the contention. The requirement to provide special attention to novel design features applies only to the PSAR, which is not at issue in this proceeding. The FSAR need only provide a description of design sufficient to permit understanding. 10 C.F.R. § 50.34(b). Applicants' proposal has not yet been included in the FSAR because it is just that -- a proposal. On October 5, 1983, Applicants proposed eliminating the need to postulate longitudinal and circumferential pipe breaks in the reactor coolant system primary loop (hot leg, cold leg, and cross-over pipes.) This proposal would only affect the need for RCS pipe whip restraints and jet barriers. The proposal was based on a generic Westinghouse fracture mechanics analysis which addressed Unresolved Safety Issue A-2 and which supported the leak-before-break concept, on a specific plant applicability report, and on the availability of the installed reactor coolant pressure boundary leak detection system which satisfied Regulatory Guide 1.45. See Letter from D. O. Foster to Harold Denton (October 25, 1983). Petitioners provide no reason to suppose that Applicants' proposal raises a safety issue.

Subsequently, the NRC issued generic Letter 84-04 which summarized the results of Staff's review of the Westinghouse fracture mechanics analysis. The NRC concluded that the analysis provided an acceptable basis for the application of alternative pipe break criteria to Westinghouse PWR's, such as

Vogtle; however, to obtain NRC concurrence with alternative pipe break criteria, an applicant must request an exemption to General Design Criteria 4. Applicants applied for an exemption on April 12, 1984. See Letter from D. Foster to H. Denton (April 12, 1984). The application for an exemption has not yet been approved, and no design change has yet been made.

Even if Applicants' proposal had been approved, making the FSAR momentarily incomplete, Petitioners would still have no basis for their contention. Applicants' proposal and letters to the Commission contain the same information as would an amendment to the FSAR; and Petitioners' assertion that they have insufficient information is frivolous as a matter of law. Applicants' letters clearly indicate the proposed design change -- the elimination of RCS Pipe Whip Restraints and Jet Barriers. Letter from D. Foster to H. Denton (October 25, 1983), Enclosure C. As is evident from their reference to this letter, Petitioners are aware of this information. Petitioners know exactly what the design change would be; they know the design basis; and they have considerable information on the leak-before-break concept and the Staff's review of Westinghouse's generic fracture mechanics analysis. This information is certainly sufficient to permit "understanding of the systems designs and their relationship to safety evaluations." 10 C.F.R. § 50.34(b).41/

41/ Petitioners quote 10 C.F.R. § 50.34(b) and emphasize the phrase "the basis with technical justification therefor." GANE

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Accordingly, GANE-9/CPG-9 lacks legal and factual basis and should be denied.

J. GANE-10/CPG-10: Environmental Qualification

GANE-10/CPG-10 is an extremely broad contention -- far broader than the corresponding discussion of the contention, which presumably is meant to provide basis with specificity. GANE-10/CPG-10 broadly alleges that Applicants have not shown that safety-related electrical and mechanical equipment and components will be environmentally qualified. Hundreds of pieces of equipment are being or have been qualified (see FSAR, Table 3.11.N.1-1), and the documentation will be tens of thousands of pages long. This contention would put every item at issue and would deprive Applicants of reasonable notice of what they must defend. The contention, as worded, is overly-broad and devoid of the requisite specificity.

In the discussion accompanying this contention, on the other hand, Petitioners raise several particular concerns with certain testing methods and pieces of equipment.42/ Applicants

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Supplement at 22; CPG Supplement at 20. However, this phrase clearly applies to the description of performance requirements, and the performance requirements of Applicants' RCS will remain unchanged.

42/ Without exception, these concerns are taken from the Union of Concerned Scientists' Supplemental Petition for Emergency

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address these concerns below. To the extent this contention is admitted at all, it must be limited specifically to Petitioners' identified particular concerns. Unless the contention is so limited, the reasons for requiring specificity, and the intent of the Commission in promulgating 10 C.F.R. § 2.714, will have been effectively abrogated.

1. Integrated Dose v. Dose Rate

Petitioners allege that Applicants' testing methods are inadequate because Applicants consider only integrated radiation dose. Petitioners assert that cable insulation and jackets, seals, rings, and gaskets, that contain polymers must be environmentally qualified on a dose-rate basis. Applicants do not object to this allegation as it relates to the polymer bearing components enumerated.

2. Synergism

Petitioners refer to a SANDIA study demonstrating synergism in cable, presumably to imply that Applicants' testing methods are inadequate. 10 C.F.R. § 50.49(e)(7) requires consideration of synergistic effects "when these effects are believed to have a significant effect on equipment performance." Applicants' test methods comply with this directive.

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and Remedial Action (February 7, 1984), which is presently before the Commission. See Statement of Policy on Environmental Qualification, 49 Fed. Reg. 8422, 8425 (1984).

See FSAR, Table 3.11.B.3-1 (Sheet 11). Applicants' FSAR states that synergistic effects have been identified for cable and that consideration of synergism for this item is therefore appropriate. Id. Petitioners ignore this. Petitioners therefore do not address Applicants' testing methods. Petitioners fail to identify any equipment for which synergistic effects should but will not be considered. Accordingly, this allegation lacks both basis and specificity. To the extent Petitioners might be asserting that synergism must be considered for all equipment qualification, the allegation is an attack on the Commission's environmental qualification rule.

3. Cable in Multiconductor Configurations

Petitioners allege that the qualification of cable in multiconductor configurations may have been based on tests of the cable in a single conductor configuration, and such tests may not be representative. Applicants do not object to this allegation.

4. Terminal Blocks

Petitioners allege that terminal blocks must be tested under accident conditions. Applicants do not object to this allegation.

5. Solenoids Valves

Petitioners allege that ASCO solenoid valves are not adequately qualified against temperature. Applicants do not object to this allegation.

6. Limitorque Motor Operators

Petitioners allege that Limitorque Motor Operators are not adequately qualified against steam spray. Applicants do not object to this allegation.

7. Hydrogen Recombiners

Petitioners allege that Rockwell International post-LOCA hydrogen recombiners fail environmental qualification tests against radiation. Applicants, however, do not use the Rockwell (catalytic-type) recombiner, but rather use a Westinghouse (electric) recombiner (as is apparent by the reference to the Westinghouse development program which Petitioners noted). This allegation is irrelevant, and there is no basis for a contention addressing Applicants' post-LOCA hydrogen recombiners.

8. Qualification Against Fire

Petitioners contend that Applicants have not satisfied 10 C.F.R. § 50.48. Petitioners assert that this provision "requires a showing that safety equipment is capable of surviving a fire in order to shut the plant down." GANE Supplement at 26; CPG Supplement at 24. Petitioners are in effect alleging that 10 C.F.R. § 50.48 imposes additional environmental qualification criteria.

Petitioners misstate and are clearly attacking the Commission's rules. 10 C.F.R. § 50.48 does not require that all safety related equipment be capable of surviving a fire (an

impossible requirement), but instead requires a number of other, specified measures to ensure the capability to safely shut down the plant in the event of a fire. 10 C.F.R. § 50.48(a).43/ These measures include fire detection and suppression systems, separation of redundant systems, the use of fire barriers, and the use of noncombustible and heat resistant materials wherever practical.

Petitioners leave no doubt that their real dispute is with the Commission. They state, "Since the NRC has no testing program to establish that the necessary safety equipment is qualified to withstand the fire environment, there is no assurance that Applicants' equipment can withstand such conditions . . ." and they cite the recent UCS Petition in which this challenge to the adequacy of the Commission's fire protection rule is being raised. GANE Supplement at 26; CPG Supplement at 24 (emphasis added).

Accordingly, this allegation must be rejected as lacking legal basis and as an attack on the Commission's environmental qualification and fire protection rules.

9. Seismic Qualification

Petitioners assert that Applicants have not used suitable seismic qualifications of safety related equipment, because

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43/ 10 C.F.R. § 50.48 n.3 incorporates by reference certain NRC documents providing basic fire protection guidance.

margins of safety have been reduced by changes in design criteria and qualification methods. Petitioners refer to NUREG-0606, "Unresolved Safety Issues Summary" (August 20, 1982). Although they do not identify which unresolved safety issue is the purported basis for this contention, it is clear that they must be referring to USI A-46, "Seismic Qualification of Equipment in Operating Plants." USI A-46 requires an assessment of operating plants in order to determine whether backfitting is necessary. Vogtle, on the other hand, is still under construction, and current standards apply. Petitioners do not allege, much less provide a basis for concluding, that Vogtle does not meet current standards. This allegation lacks basis and must be rejected.

10. "Shortcomings" in Qualification Methodologies

Petitioners next refer to a number of general questions which SANDIA proposes to examine. Petitioners apparently contend that topics for research to improve qualification procedures "raise fundamental doubts" about present equipment qualification. But with respect to its proposed topics of research, SANDIA did not say that present methodology was inadequate; and there is no allegation that Applicants are not using state of the art methods of environmental qualification. Petitioners do not allege that SANDIA's questions are likely to be answered soon. Furthermore, even if future research resulted in improved methods of qualification, it does not follow that

present methods would be invalid, Finally, Petitioners make no attempt to identify particular equipment whose qualification might be in doubt. This allegation lacks specificity, and vague references to subjects of research do not provide an adequate basis. The allegation should be rejected.

11. Accident Parameters

Petitioners refer to statements made by Steven Hanauer (former NRC Assistant Director for Plant Systems) and Robert Pollard (a UCS engineer) in support of an assertion that accident parameters for purposes of environmental qualification need to be redefined. Petitioners, however, do not specify what parameters are inadequate. Moreover, both these statements were made in 1979 and were based on considerations stemming from the TMI accident. The Commission's environmental qualification rule, however, was promulgated in 1983 and substantially upgraded the environmental qualification requirements (48 Fed. Reg. 2729 (1983)); and it requires qualification against the most severe design basis accident. To the extent that the general concerns in the statements might not have been resolved, they would constitute attacks on the standards which the Commission has defined.

Accordingly, this allegation lacks specificity by failing to allege any accident parameter that is inadequate, and it lacks basis. It should be rejected.

K. GANE-11/CPG-11: Unresolved Safety Issues

GANE-11/CPG-11 is yet another contention which, as worded, is far broader than its basis. GANE-11/CPG-11 vaguely alleges inadequate consideration of "generic defects in the Westinghouse PWR." However, Petitioners' discussion of this contention merely addresses several generic Unresolved Safety Issues: Unresolved Safety Issue (USI) A-1, which addresses water hammer; USIs A-3, A-4, and A-5, which address Steam Generator Tube Integrity;44/ and USI A-49, which addresses pressurized thermal shock. GANE Supplement at 28; CPG Supplement at 26; compare NUREG-0606, "Unresolved Safety Issues Summary," (Vol. 6 No. 1, Feb. 17, 1984) at 11, 14, and 42. Again, as with GANE-10/CPG-10 in particular, if admitted at all, this contention should be limited to the specific concerns identified and particularized by Petitioners. In any event, each of these Unresolved Safety Issues is the subject of an NRC Task Action Plan. See NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants" (Feb. 1980). As previously discussed with respect to GANE-6/CPG-6, a petitioner who wishes to litigate an Unresolved Safety Issue must

44/ In the first sentence of Petitioners' discussion of this contention, the terms "corrosion-induced wastage," "cracking," "reduction in tube diameter," and "vibration induced fatigue cracks" are drawn from the description of USIs A-3, A-4, and A-5. The term "degradation due to bubble collapse water hammer" presumably refers to USI A-1.

establish a nexus between the license application and the task action plan by showing both:

> (1) that the undertaken or contemplated project has safety significance insofar as the reactor under review is concerned, and
>
> (2) that the fashion in which the application deals with the matter in question is unsatisfactory, that because of the failure to consider a particular item there has been insufficient assessment of a specific type of risk for the reactor, or that the short-term solution offered in the application to a problem under Staff study is inadequate.

Gulf States Utilities Company (River Bend Station, Units 1 and 2), ALAB-444, 6 N.R.C. 760, 773 (1977). Applicants now address the particular concerns identified by Petitioners.

1. USI A-1: Water Hammer

Petitioners raise this Unresolved Safety Issue only as it relates to steam generators. See GANE Supplement at 28; CPG Supplement at 26. Petitioners make no attempt to show that this Unresolved Safety Issue has any significance for Vogtle. They also make no attempt to address the fashion in which the application deals with the matter. Applicants describe the measures taken to protect against water hammer in their FSAR at § 5.4.2.2, 10.4.7.2.1, and 10.4.7.2.2.1. This includes the use of J tubes and a feed ring in Applicants' topfeeding steam generators--a proven design concept. See NUREG-0927, "Evaluation of Water Hammer Experience in Nuclear Power Plants" (May 1983) at 2-38. A separate auxiliary feedwater line and nozzle also minimizes the potential for water hammer. FSAR, § 5.4.2.2.

Petitioners totally fail to satisfy the pleading standards for this unresolved safety issue.

2. USI A-3, A-4, and A-5: Steam Generator Tube Integrity

Applicants provide an extensive description of their steam generators and water chemistry program in the FSAR at § 5.4.2 and § 10.3.5 respectively. Petitioners, however, make no attempt to address Applicants' proposals or show that USI A-3 through A-5 have any significance with respect to Vogtle. Instead, Petitioners merely paraphrase the description of these USIs. See NUREG-0606, supra, at 14. This certainly does not satisfy the pleading requirements for Unresolved Safety Issues.

Applicants have addressed the forms of tube degradation that have been experienced in other Westinghouse steam generators: wastage, intergranular attack, and cracking. Wastage is characterized by a general loss of metal from the tube wall as a result of a chemical corrosive reaction. Intergranular attack is a chemical reaction in which the grain boundaries of the Inconel 600 tube are attacked by acidic or caustic solutions. Both of these forms of tube degration occur only in steam generators using sodium phosphate as a chemical additive to control the pH of the steam generator secondary water. Vogtle, however, uses all volatile treatment (AVT)--hydrazine and ammonium hydroxide. See FSAR, § 10.3.5.2.

Stress corrosion cracking -- other than due to denting -- has been experienced only in the Westinghouse Model 51 steam

generators constructed in the early 1970s. The Vogtle steam generators, however, are Westinghouse Model F, and increased margin against stress corrosion cracking has been obtained by the use of thermally treated Inconel-600 tubing, particularly effective in resisting caustic corrosion. FSAR, § 5.4.2.4.3.

Stress corrosion cracking may also occur due to denting. Denting is caused by rapid corrosion of carbon-steel tube support plates at cylindrical holes where the tubes pass through support plates. Vogtle's design, however, reflects technical advances to avoid denting, which Petitioners ignore. At Vogtle, the tube support plates used in the Model F are ferritic stainless steel, which is resistant to corrosion in the AVT environment. Furthermore, the support plates are designed with quatrefoil tube holes rather than cylindrical holes, and this design promotes high velocity flow along the tube and should minimize the accumulation of impurities at the support plate locations. FSAR, § 5.4.2.4.3.

Instead of addressing Applicants' proposals, Petitioners assert that there is not "sufficient technical information currently available to deal with a steam generator tube rupture (SGTR) accident as occurred in a Westinghouse Plant at the Borselle [sic] Nuclear Power Station." GANE Supplement at 28; CPG Supplement at 26. Petitioners refer to Board Notification 83-151 and a Memorandum from Roger Mattson to Darrell Eisenhut (September 26, 1983) concerning this Board Notification.

Borssele, however, is not a Westinghouse plant, and as is readily apparent from these sources, no SGTR accident occurred at Borssele. Rather, the operators of the Borssele Nuclear Power Plant modified their SGTR procedures and revised the ECCS actuation logic to avoid ECCS actuation in the event of an SGTR accident. This revision was intended to improve the manageability of SGTR events, and it intrigued the NRC Staff. The Staff therefore requested the Westinghouse Owners Group to evaluate the potential benefits and detriments of the revision.45/ Accordingly, Petitioners' reference to an accident at Borssele lacks factual basis. Moreover, Petitioners do not attempt to demonstrate the relevance to Vogtle of the Borssele revision to ECCS actuation logic. They do not allege that Vogtle cannot be safely operated without this revision.

Petitioners have failed to satisfy the pleading standards for this unresolved safety issue (Steam Generator Tube Integrity); they have failed to provide any degree of specificity; and they have failed to provide any basis in support of their contention. These infirmities in contention dictate its rejection.

45/ The Westinghouse Owners Group responded on February 17, 1984. Letter from J. J. Sheppard to Darrell G. Eisenhut (Feb. 17, 1984). The Group concluded that because of inherent differences in design, the arguments for reinstating the coincident signal logic in the Borssele reactor was not directly relevant to Westinghouse reactors, and such a change would degrade protection against a stuck-open PORV.

3. USI A-49: Pressurized Thermal Shock

Again, Petitioners do not address Applicants' treatment of this issue and fail to show its significance.46/ Again, they merely paraphrase the description of the unresolved issue. They thus fail to satisfy the pleading requirements for unresolved safety issues.

Furthermore, pressurized thermal shock is the subject of GANE-6/CPG-6. This issue in GANE-11/CPG-11 is therefore redundant and should be rejected pursuant to 10 C.F.R. § 2.714(e).

L. GANE-12/CPG-12: Acid Releases

Petitioners allege that Applicants have not properly assessed the amount of salt and hydrochloric acid release from the cooling towers and the extent of consequent adverse agricultural and environmental damage.

Petitioners provide absolutely no basis for their assertion that "thousands of pounds per day" of hydrochloric acid would be emitted from the cooling towers;47/ and the hypothesis that any hydrochloric acid could be emitted from the cooling

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46/ Applicants' treatment of this unresolved issue is discussed with respect to GANE-6/CPG-6, above.

47/ Petitioners assert that hydrochloric acid was brought up by the NRC Staff at the Construction Permit hearing, and they refer to a personal communication with N. Herring, who made a limited appearance during the construction permit proceeding. Applicants, however, have reviewed the transcripts of the construction permit proceeding and have found no discussion at all of possible hydrochloric acid releases from the cooling towers.

towers defies the fundamental laws of chemistry. The cooling tower water is alkaline. OL-ER, Table 3.6-2 (sheet 2).48/ Any hydrochloric acid produced by chlorination would be rapidly neutralized.49/

Petitioners also state absolutely no basis for their assertion that the impact from salt releases has not been adequately assessed. The Staff assessed the effect of salt emissions at the construction permit stage and concluded they would be negligible. CP-FES, § 5.5.1.1.50/ Petitioners provide no basis or new information to dispute this conclusion.51/ Furthermore, the 305 lb/acre/year maximum salt deposition estimate which the Staff considered was revised in Applicants' Operating License State Environmental Report to reflect current design parameters and expected operating conditions. Petitioners cited NRC Staff question E290.3, which addressed the previously calculated emission, and were therefore certainly aware of the revision. Applicants' Operating License Stage Environmental Report estimates the maximum salt deposition to be 31

48/ In fact, sulphuric acid is added to the cooling tower water to reduce alkalinity.

49/ Fundamental physical laws may be considered by a Board in rejecting a contention. Philadelphia Electric Company (Limerick Generating Station, Units 1 and 2), ALAB-765, 19 N.R.C. _____, slip op. at 13 n.13 (March 30, 1984).

50/ Petitioners incorrectly cite the FSAR instead of the FES.

51/ See cases cited in note 24, supra.

lbs/acre/year on-site, and 21 lbs/acre/year offsite. See OL Supp. ER, § 3.6.4.2 and responses to questions E290.3 and E451.17 (at the back of the Operating License Stage Environmental Report). To the extent Petitioners would argue that this is a change since the construction permit stage environmental review, it would be ludicrous to assert as a basis a reduction in the effect by an order of magnitude.

Accordingly, Petitioners have failed to address Applicants' proposal and have failed to provide any basis for their concerns. GANE-12/CPG-12 should be rejected.

M. GANE-13/CPG-13: Emergency Planning

GANE-13/CPG-13 addresses emergency planning. The emergency plan, however, is being extensively revised. Applicants therefore submit that ruling on this contention should be deferred until the revised plan has been submitted to the Commission. Applicants will ensure that Petitioners are provided a copy, and Applicants will not object on timeliness grounds to new or revised emergency planning contentions which are filed within a reasonable time thereafter.

N. CPG-1: Timely Completion of the Plant

CPG 1 contends that there is no reasonable assurance that Plant Vogtle will be completed on a timely basis, as required by 10 C.F.R. Part 2, Appendix A, Section VIII(b)(1).

CPG does not state what a timely basis for completed construction is. Applicants submit, however, that construction is timely if it is completed prior to the expiration of the construction permit. Pursuant to the Vogtle construction permit, Applicants' construction completion dates are March 31, 1988 for Unit 1 and September 30, 1989 for Unit 2. Furthermore, in Washington Public Power Supply System, et al. (WPPSS Nuclear Power Project No. 1), LBP-83-66, 18 N.R.C. 780, 781 (1983), a Licensing Board ruled that the issue whether a facility is being completed on a timely basis can only be properly raised in the context of an applicant's application for an extension of its construction completion date. No such extension is pending before the Board in this proceeding.

Even if timely completion of construction were cognizable in this proceeding, CPG raises no litigable issue. CPG merely asserts that economic considerations could force the cancellation of the units. But if the Vogtle units were cancelled, the CPG contention would be moot. Moreover, it is readily apparent from its discussion of CPG-1 that CPG is in reality seeking to relitigate whether the construction permit should have been granted and whether Applicants are financially sound. Neither of these issues is proper for adjudication in this proceeding. In an operating license proceeding, a Licensing Board has no general jurisdiction over the construction of the plant, and it cannot suspend a previously issued permit. Consumers Power

Company (Midland Plant Units 1 and 2), ALAB-674, 15 N.R.C. 1101 (1982). And financial qualifications are not litigble in this proceeding because they are the subject of general rulemaking. Proposed Rule: Elimination of Review of Financial Qualifications of Electric Utilities in Operating License Reviews and Hearings for Nuclear Power Plants, 49 Fed. Reg. 13044 (1984). See Applicants' response to CPG-3, below.

O. CPG-2: Need for Power

CPG-2 contends that there is no reasonable assurance that the production capacity of Plant Vogtle will be needed, as required by NEPA and 10 C.F.R § 50.42.

The Commission has determined, however, that experience has consistently demonstrated that a constructed nuclear power plant will always be needed, either to meet incremental energy needs or to replace older, less economical generating capacity. 47 Fed. Reg. 12940, 12942 (1982). The Commission has therefore directed that in operating license proceedings "Presiding officers shall not admit contentions proffered by any party concerning the need for power. . . ." 10 C.F.R. § 51.53(c). Furthermore, CPG's reference to 10 C.F.R. § 50.42 is misplaced. 10 C.F.R. § 50.42 applies to the allocation of nuclear fuel and to antitrust considerations, neither of which is an appropriate issue in this proceeding. See 10 C.F.R. § 2.104(c).

P. CPG-3: Financial Qualifications

CPG-3 contends that there is no reasonable assurance that Applicants are financially qualified to operate and subsequently decommission the facility. CPG Supplement at 6.

Since the Commission's rule barring consideration of financial qualifications in an operating license proceeding was recently remanded by the U.S. Court of Appeals for the District of Columbia Circuit,52/ the Commission has commenced a rulemaking proceeding to revalidate its previous proscription. 49 Fed. Reg. 8445 (1984). Because financial qualifications are the subject of this rulemaking, they are an inappropriate subject for a contention in this proceeding.

Even if this rulemaking were not underway (or did not bar litigation of financial qualifications in this proceeding), Applicants submit that the contention should, at the very least, be deferred. The Commission has already held several meetings to consider the financial qualifications issue and is likely to provide the Licensing Boards with further guidance. There is no reason not to await such guidance, and deferral would avoid imposing an unnecessary and undue burden on Applicants, particularly with respect to discovery obligations.

52/ New England Coalition on Nuclear Pollution v. NRC, No. 82-1581 (D.C. Cir. February 7, 1984).

Q. CPG-4: Alternative Energy Sources

CPG-4 contends that Applicants have not reasonably explored alternatives to the plant as required by NEPA and "other laws, rules, and regulations." CPG asserts that conservation may be less costly.

CPG does not identify the "other laws, rules, and regulations" that require assessment of alternatives in the operating license stage, and in fact the Commission's regulations specifically proscribe consideration of alternatives. 10 C.F.R. § 51.53(c) states "Presiding officers shall not admit contentions proffered by any party concerning . . . alternative energy sources for the proposed plant in operating license proceedings." This proscription is not inconsistent with NEPA. It merely reflects the fact that the Commission has determined generically that no viable alternatives to a completed plant are likely to exist which could tip the NEPA cost benefit balance against issuance of the operating license. 47 Fed. Reg. 12940 (1982).

Respectfully submitted,

SHAW, PITTMAN, POTTS & TROWBRIDGE

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Counsel for Applicants

Dated: May 7, 1984

May 7, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
GEORGIA POWER COMPANY, ET AL.	)	Docket Nos. 50-424
	)	50-425
(Vogtle Electric Generating Plant,	)	
Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "Applicants' Response to GANE and CPG Supplements to Petitions for Leave to Intervene," dated May 7, 1984, were served upon the persons on the attached Service List by deposit in the United States mail, postage prepaid, this 7th day of May, 1984.

Ernest L. Blake, Jr.

Ernest L. Blake, Jr., P.C.

Dated: May 7, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
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(Vogtle Electric Generating Plant,	)	
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Attachment 1

Significance of Human Exposure to Low-Level Radiation*

by

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Although in this presentation I will speak only about the effects of ionizing radiation (x, γ, β, n) on man, I purposely have not qualified the word Radiation in the above title in order to remind us that there are other forms of radiation (sonic, ultrasonic, infrasonic, ultra violet, infrared, microwave, radio frequency and long wave). These non-ionizing radiations, like ionizing radiation, may be multiplying their presence so rapidly in the human environment that they too are resulting in health hazards in some areas before we are sufficiently aware of the magnitude of the problem. Let us resolve not to repeat with them the mistakes we have made with ionizing radiation by underestimating their risks of chronic damage.

When I began as one of the five first health physicists at the Metallurgical Laboratory at Chicago under Dr. A. H. Compton early in 1943, I was reminded of the suffering and death that had been a retribution to many persons who through ignorance and carelessness had not shown proper respect for the danger of chronic damage (cancer) associated with excessive exposure to x-rays and radium. As health physicists, and as forerunners of a new profession (e.g. 3400 members of the U.S.-Canadian Health Physics Society in 1977), we were charged with the responsibility of devoting our new profession to the task of minimizing radiation exposure and making this new industry of atomic energy one of the safest of modern industries in spite of the fact that we would be building nuclear reactors (called piles in those days) that would produce levels of radiation behind thick walled containment that were many billions of times greater than man had here-to-fore conceived of. Also, although only one ten millionth of a gram (0.1μg) of radium was the

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*Presented before the Congressional Hearings in Washington, D.C., January 24, 1978

maximum permissible body burden for a radiation worker, there would be hundreds of pounds of plutonium in a reactor (e.g. 3 metric tons in a 1000 MWe LMFBR) and we even then had good reason to believe plutonium would be as radiotoxic (or moreso) as radium. Looking back over these past 35 years I believe in many ways we have made a remarkable safety record. By contrast many other industries (chemical, coal burning power plants, ground transportation, etc.) from their beginning gave very little attention to minimizing associated occupational hazards and environmental problems. This, however, does not mean we have a perfect record in the nuclear industry for we have made many mistakes and only by comparisons with the bad record of other industries does atomic energy have a good record. It should be emphasized also that in so far as unnecessary exposure to ionizing radiation is concerned the finger of guilt should be pointed first toward the medical professions from whence more than 90% of our exposure from man made sources derives. I[1] have shown that this medical exposure could be reduced to 10% of its present amount while at the same time improving medical radiography. This is a long neglected area that deserves more attention form all of us.

The problem of control of radiation exposure during the past 35 years has been made especially difficult because each year new evidence has shown that the risks of radiation induced malignancies is greater than we had thought the year before and from time to time there have been quantum drops in the levels of maximum permissible exposure.

During the first half of this 35 years of the atomic age (1942-1960) a large number of scientists, perhaps most who were knowledgeable in health physics and radiobiology, accepted the threshold bypothesis or the theory that there is a safe level of exposure to ionizing radiation and so long as a person does not exceed this threshold or safe level no harm will result or the radiation damage on the average will be repaired as fast as it is produced. During the last half of this period (1960-present) an overwhelming amount of data have been accumulated that show there is <u>no safe</u> level of exposure and no dose of radiation can be so low that the risk of it causing a malignancy is zero. The question before us at this meeting today, therefore, is not, Is there a risk from low level exposure?

or, What is a safe level of exposure? Rather it is, How great is this risk? or, How large may a particular radiation risk be before it exceeds the expected benefits?

One of the problems we face today is that many scientists had accepted the threshold hypothesis as a cardinal law and had lived with this hypothesis so long that they became staid or petrified in their thinking and now they cannot believe or accept the fact that the threshold hypothesis was wrong and it cannot be applied to the induction of malignancies by radiation. It is obvious to all scientists in this field as well as to the diehards for the threshold hypothesis that at least for some types of radiation damage and for some kinds of radiation exposure there is some repair going on in the body of the radiation damage, but the diehards do not seem willing or able to accept the preponderance of evidence that there is never complete repair of radiation damage in the practical case since even at very low exposure there are many thousands of interactions of the radiation with cells of the human body (for example one rad of x-rays of 1 MeV energy corresponds to 2.2 billion photons per cm^2 acting on the body). It is inconceivable that all the billions of irradiated and damaged cells would be repaired. There are undoubtedly many mechanisms of radiation damage that contribute to the development of a malignancy following an exposure to radiation; perhaps the most significant damage from a low exposure would be to the nucleus of one of the billions of cells that are irradiated. There are 46 chromosoms in the nucleus of each normal somatic cell of the human body and along each chromosom is coded millions of bits of information like an immense library which enables or instructs the cell to function properly and to divide at the appropriate time. When radiation passes through the human body, four principal things can happen: 1) the radiation passes through the cell without producing any damage, 2) the radiation kills the cell or renders it incapable of cell division, 3) the radiation damages the cell such that the damage is repaired adequately and 4) the cell is damaged such that it survives and multiplies in its perturbated form over a period of years (5 to 70 years in forming a clone of cells that eventually is diagnosed as a malignancy. It is only this last event that concerns us here with somatic damage from low level exposure. It seems

obvious that if this etiology or a similar series of events leads to the development of a malignancy, there can be no dose so low that the risk is zero. Thus the risk of induction of cancer from radiation exposure increases more or less with the increase or accumulation of radiation exposure and the risk is simply one of chance, just the same as the risk or chance of an accident each time we take an automobile trip.

It is very evident also that all persons do not run the same risk of developing a malignancy from a given radiation exposure and that the risk of some types of cancer are greater for some people than are the risks of other types of cancer. Dr. Burch[2] has shown for example that the final onset of a malignancy may require a series of events and a certain type of leukemia may require as many as three successive events (like throwing three electrical switches which are connected in series). Thus, for example, one switch may be thrown genetically so that if one twin dies of a particular type of leukemia (1 switch thrown genetically), the other twin has a high probability of eventually suffering similar consequences. Some of these switches may be thrown by viruses, bacteria, chemicals, mechanical insults or by radiation and when one of these agents throws its switch in a cell of the body, it seems to pave the way or make it easier for radiation to throw its switch in this cell. Thus the studies of Dr. Bross[3,4] have shown that children (age 1-4) with allergic diseases such as asthma or hives have a 300-400% increased risk of dying of leukemia compared with other children (i.e. allergic diseases throw one switch). Children who received in utero diagnostic x-ray exposure have a 40 to 50% increase in risk of dying of leukemia (in confirmation of the extensive studies in the United Kingdom conducted by Drs. Alice Stewart and George Kneal[5]) while children with two switches thrown (i.e. in utero exposure and later developing a virus disease) have 5000% increase in risk of dying of leukemia. Studies of Stewart and Kneal,[5] B. MacMahon,[6] BEIR Committee,[7] and Bross[3,4] as well as those of many other researchers show that children have a higher risk of dying of radiation induced leukemia than do middle aged persons. Also, it

has been shown by others, for example Hempelmann et al.,[8] Albert et al.,[9] Silverman and Hoffman,[10,11] etc., that radiation induced thyroid carcinoma presents a higher risk in children than in an adult population. There are studies also which indicate women have a much higher risk of certain types of radiation induced malignancies than do men and that older and younger men (see report of Mancuso, Stewart and Kneal[12] on increased malignancies among Hanford radiation workers) have a higher risk of radiation induced malignancies than do men of middle age. After publication of the studies of E. Saenger and E. A. Tompkins[13] supported by the Bureau of Radiological Health which was interpreted to show that there is no risk of thyroid carcinoma from exposure of young people to ^{131}I, the BEIR Committee[7] wisely pointed out that the conclusions of Saenger and Tompkins that there is no increased cancer risk cannot be justified by their failure to find a clear cut increase in thyroid carcinoma among hyperthyroid patients treated with ^{131}I. I believe their failure to observe an increased cancer risk can be explained in several ways, two of which are the followup period was too short and high doses produce cell sterilization (or destruction of the thyroid cells that are likely targets to develop a malignancy without seriously damaging extrathyroid tissue. Hempelmann,[8] Modan et al.[10] and many others have shown that radiation exposure to the thyroid does increase the risk of thyroid carcinoma at low doses (at least down to 6.5 rad) and that the risk seems to increase linearly with the magnitude of the dose. Incidentally, E. B. Lewis[14] pointed out that after examining this data of Saenger and Tompkins[13] he observed that they failed to note there was a significant increase in leukemia among persons between ages 50 and 79 who received the ^{131}I treatments. He[14] noted also that the statistical method used by Tompkins "lacks the power to detect even large differences between the two groups." Parenthetically, I might remark that we hear the Hanford Study of Mancuso, Stewart and Kneal was moved recently in-house to Oak Ridge where Dr. Tompkins will be responsible for the statistical method.

As stated above the cancer risk from exposure to ionizing radiation is certainly much greater than was thought to be the case some years ago. During the 29 years I was director of the Health Physics Division at Oak Ridge National Laboratory, my Division worked closely with the Atomic Bomb Casulty Commission (ABCC) and was responsible for determining the radiation dose received by the survivors of the bombings at Heroshima and Nagasaki, Japan. Thus I have a great interest in the success of these studies. In the early period, following the deaths from radiation sickness that occurred shortly after the atomic explosions, it appeared to many scientists the principal chronic risk from the radiation exposure was only an excess of cases of leukemia. The number of excess leukemias reached a peak about six years after the bombing and since then it has slowly declined. During this period many persons jumped to the conclusion that the only chronic risk among these survivors was that of developing leukemia. Unfortunately, however, as the study of these survivors has continued and extended further into the incubation periods of the various malignancies, other forms of cancer (bone, breast, lung) have shown a significant increase above the controls. Probably with the passage of time we will find that this exposure has resulted in an increase of statistical significance in many or most kinds of malignancies that are common among human populations.

During the first half of this atomic age (1942-1960) almost everyone assumed that the genetic risk from low level radiation exposure far exceeded the risks of chronic somatic damage such as cancer or life shortening. During the last half of the atomic age, however, it has become increasingly clear that this assumption also is unwarranted and untenable. The BEIR[7] committee pointed out "until recently, it has been taken for granted that genetic risks from exposure of populations to ionizing radiation near background levels (~ 100 mrem/y) were of much greater import than were somatic risks. However, this assumption can no longer be made if linear non-threshold relationships are accepted as a basis for estimating cancer risks." The committee then went on to supply many pages of data most of which lend strong support of the linear hypothesis. In 1971 the International Commission on Radiological Protection (ICRP)[20] made a similar observation, "It could be concluded that the ratio of

somatic to genetic effects after a given exposure is 60 times greater than was thought 15 years ago." This discussion, however, is not to depreciate the seriousness of genetic risks from exposure to ionizing radiation but rather to emphasize that the scientific community was rather smug 15 years ago (as some scientist still are today) in its belief that somatic risk is far less than genetic risk and somatic risk is almost negligible at low doses but now most of us realize the risk of inducing cancer at low doses of radiation is far greater than we once thought it to be and it may be as great or greater for the human race than genetic risk. Never-the-less I wish here to pause and sound a warning that I'm sure my long time friend and associate the genetist, Dr. H. J. Muller, would urge me to make were he alive today, namely the BEIR report only treated the long term recessive mutation question in a superficial way and it may well be that many and perhaps most of our human diseases are related to a genetic factor and especially to the 10,000 non-visible or "small" mutations that result per visible mutation that we can observe. It may be that in the long run Muller's small mutations that result in such things as lack of vigor, susceptibility to disease, a slight reduction in mentality and physique, etc. will be a greater burden to society than the easily identifiable dominant mutations because small mutations are eliminated so slowly from the gene pool.

As stated above, there has been a number of reductions in the permissible exposure levels for occupational workers and for members of the public during the past 35 years even though many of those in organizations which set the standards at national National Council on Radiation Protection, (NCRP) and international (ICRP) levels (radiologists, nuclear energy workers, etc.) would never admit there was any real need for greater conservatism. Table 1 indicates some of the quantum drops in permissible exposure levels during the past 35 years. The occupational maximum permissible exposure level has dropped by a factor of 10 and the level for members of the public by a factor of 300. Our concern for the environment and the concentrations of radionuclides in air, water and food likewise have undergone evolutionary jumps. For example, in 1943-44 I was faced with setting the maximum permissible concentration of radioactive material in White Oak Lake which impounds the liquid radioactive waste discharged from Oak Ridge National Laboratory. The only standard I had to go by was the NCRP value of 0.1R/d

TABLE 1. CHANGES IN LEVELS OF PERMISSIBLE EXPOSURE TO IONIZING RADIATION

FOR RADIATION WORKERS:

Recommended Values		Comments
0.1 erythema dose/y (~1R/wk for 200 kV x-ray)	52 R/y	Recommended by A. Mutscheller and R. M. Sievent in 1925. This was recommended by ICRP in 1934 and used world-wide until 1950.
0.1 R/day (or 0.5 R/wk)	36 R/y	Recommended by NCRP on March 17, 1934.
0.3 rem/wk	15 rem/y	Recommended by NCRP on March 7, 1949 and ICRP in July, 1950 for total body exposure.
5 rem/y	5 rem/y	Recommended by ICRP in April, 1956 and NCRP on January 8, 1957 for total body exposure.

FOR MEMBERS OF THE PUBLIC:

0.03 rem/wk	1.5 rem/y	Suggested by NCRP in September, 1952 for any body organ.
0.5 rem/y	0.5 rem/y	Suggested by NCRP on April 15, 1958 and by ICRP in July, 1959 for gonads or total body.
5 rem/30 y	0.17 rem/y	Suggested by ICRP on September 9, 1958 for gonads or total body.
25 mrem/y	0.025 rem/y	Suggested by USEPA on January 13, 1977 for any body organ except thyroid
5 mrem/y	0.005 rem/y	Suggested by USERDA in 1974 for persons living near a nuclear power plant.

NOTE: (1) 1 R = 0.88 rem.
(2) See Reference 16 for additional information.

because this was before lower levels were set for the population at large. Some of my engineering associates at ORNL were very much provoked at my conservatism and they insisted I set the level at 100 R/d because of the rather unlikely event persons would drink or swim in this water (signs and fences as well as the turbidity of the water discouraged fishing and swimming in White Oak Lake). Today we recognize this higher level would have been over 6 million times the value we probably would use today (5 mrem/y = 0.0000 15 R/d). I stuck to my guns, however, and used the 0.1 R/d instead of 100 R/d but in the context of today's standards I deserve no praise because this value is 6000 what some would insist on our using today even though shortly below where White Oak Lake empties into the Clinch River and leaves the reservation it is diluted by a factor of over one hundred thousand.

Much of what has been said about the risks of exposure to low levels of ionizing radiation would have considerably less weight if it could be shown that although the linear hypothesis holds at intermediate to high levels of exposure it provides a very large element of conservatism at low doses and dose rates. I am amazed and appalled at the large number of scientists (mostly associated in some way with ERDA - now DOE) and radiologists who in spite of an overwhelming amount of data supporting the linear hypothesis at low doses, are still saying we have no human exposure data at low doses and that there is a large factor of conservatism in this hypothesis when it is applied to low doses. I take the opposite position, namely that we have a large amount of such data. Much of it is human exposure data showing a statistically significant increase in a number of types of malignancies as a consequence of exposure to low doses of radiation and the number of malignancies increases progressively as the dose accumulates. These doses in some cases are considerably lower than the present levels of maximum permissible occupational exposure. In fact, I have gone further; I(17) and many others(25,26,27) have shown that in some cases (e.g. internal exposure to plutonium) the linear hypothesis is non-conservative and the present MPC values for plutonium and transplutonic elements(21) should be reduced considerably, (see reference 17 attached as an appendix to this paper).

Table 2 indicates the magnitude of the cancer risk and that this risk has been shown to increase linearly with the accumulated dose down to very low values, i.e. down to less than 1 rad for leukemia or other forms of cancer resulting from pelvimetries, and to 6.5 rad for thyroid carcinoma resulting from x-ray therapy of the scalp for ringworm (tinea capatis). It must be pointed out that these doses (0.8 and 6.5 rad) are not the doses below which the linear hypothesis breaks down but the lowest points on the human exposure curves for these two malignancies and we have every reason to believe the linearity of these curves continues on down to zero dose and that there is a similar linearity for other types of cancer that simply have a longer incubation period. It should be emphasized also that this 0.8 rad is only 5% of the 15 rad permitted each year to the active bone marrow of the radiation worker and that the 6.5 rad is only 22% of the 30 rad permitted each year to his thyroids. (The MPC values given by ICRP and NCRP for members of the public are calculated on the basis of 10% of these dose rates, i.e. 1.5 rem/y for bone marrow and 3 rem/y for thyroid). Also, from this table we note that if a million children each received 1 rad from in utero exposure we would expect from 300 to 3000 leukemias, depending upon whether or not the child had certain respiratory diseases, some of which, as indicated by Bross,[3,4] act synergistically with radiation exposure. There is not as much data on adults as for children but as seen from Table 2 their risk may not be less by more than a factor of three. Furthermore studies of Stewart,[5,22] MacMahon,[6,23] and many others indicate that following in utero exposure the incidence of focal cancers (such as central nervous system tumors) is about that of leukemia so the number of fatal malignancies might be twice the numbers given above (i.e. 600 to 6000 cancers for a million children exposed to only 1 rad).

In 1970 Jablon and Kato[24] pointed out that their data on the survivors of the atomic bombings who were exposed in utero do not support the findings of Stewart,[5,22] MacMahon[6,23] and others. They indicated that on the basis of findings of Stewart and Kneale and upon the corresponding linear hypothesis they should expect 36.9 excess cancers in this group during 10 years after exposure and there had been only one (a case of

TABLE 2. CANCER RISK AND KNOWN RANGE OF LINEARITY

Linearity of Dose Down To:	Risk Per Person Per Rad	Comments	References
<10 Rad	$0.3 - 1.0 \times 10^{-4} \ell$ $0.5 - 1.7 \times 10^{-4} C$	Hiroshima & Nagasaki atom bomb survivors	7, 18, 19, 20
Av. 370 Rad	$0.2 - 0.3 \times 10^{-4} \ell$	Ankylosing spondylites patients	7, 20
0.2 - 0.8 Rad	$3 \times 10^{-4} \ell$ $6 \times 10^{-4} C$	Pelvimetry Exposures - Stewart & Kneale	5, 22
- 1.0 Rad	$3 - 30 \times 10^{-4} \ell$	Pelvimetry Exposures - Bross et al	3, 4
20 Rad	$0.5 - 1.1 \times 10^{-4} T$	X-Ray Therapy - Hempelmann	8
6.5 Rad	$1.2 \times 10^{-4} T$	X-Ray for Tinea Capitis - Modan et al	10

ℓ = Leukemia risk/person. rad
C = Total cancer risk/person. rad
T = Thyroid cancer risk/person. rad

liver cancer). As a consequence many persons (who were desperately trying to disprove the linear hypothesis and show low level exposure is harmless) were quick to proclaim that there was something wrong with the retrospective studies of cancer induction by diagnostic in utero x-ray as reported by Stewart, MacMahon and others and that now we could relax. Unfortunately (for in utero exposed children), this was not the case. Dr. Stewart and a number of other writers[1,17,19,28,29,30] have published reports which give strong support of the studies of Stewart, MacMahon, etc. of cancer induction by diagnostic in utero x-ray such that there is now little doubt the Japanese studies greatly underestimate this cancer risk. In fact Jablon and Kato[24,7] in their original publication gave an explanation that now seems to be one of the principle reasons they observed such an unusually low cancer rate among children who had received in utero exposure at the time of the bombing; they said "Conceivably such a result might follow if there were an excessive spontaneous abortion-rate for fetuses by large doses." Thus the fetuses which were most likely to have developed into cases of radiation induced leukemia received such high doses that they did not survive to become statistics. In fact the record indicates there was an unusually high incidence of abortions and rate of infant mortality following the atomic bombings. Many studies[31] have shown that during periods of stress and community disasters it is the infants and young children that suffer the most. It is known that during such periods of suffering and unrest incipient cancers can easily be mistaken for acute infections. Also, it seems likely that the Japanese control group may have had a greater cancer risk than normal.

In conclusion, I have given a very simplified picture of events that probably lead to the production of cancer in humans and from such a theory as well as a large amount of data on low level human exposure to ionizing radiation it seems evident that all forms of cancer can be caused by low level exposure, that the linear hypothesis when applied to low level exposure is not conservative and thus there can be no dose so low that the risk of radiation induced cancer is zero. In the case of radiation exposure the risk is simply an increase in risk that the person will develop cancer

but this cancer may not develop and be recognized until 5, 20 or 50 years later. In comparison with other industries, I believe the nuclear energy has a remarkable safety record. However, I believe it has many black spots on this record such as, for example: 1- The West Valley, N. Y. reprocessing plant and the Kerr-McGee fabrication plant near Oklahoma City which broke every conceivable health physics rule of good radiation safety, 2- The practice of "burning-out" temporary employees on some of the "hot" operations is inexcusable. The steam generators of the PWR have continued to account for high occupational exposures in the nuclear power plants and the solution is not to hire temporary employees to divide up the dose but rather to correct the source of the trouble. Other industries are not innocent. For example we recall exposures from shoefitting machines, radium dial watches, smoke detectors containing ^{241}Am and ^{226}Ra, color TV and industrial radiography. The shadow of improper concern or understanding of radiation risks also falls on the military with its exposure of young men on Test Smoky. The worst offender, however, has been the medical profession. For example, after years of effort we finally outlawed mass chest x-ray programs only to have the profession get underway a mass mammography program. Like the mass chest x-ray program this was much harder to stop than it was to get it underway. The dentists are better only in that the dose they deliver to the population is less than that in medical diagnosis, but their failure to take radiation protection measures[1] has without question increased significantly the number of CNS tumors, thyroid carcinomas and leukemias among our population. Is it a wonder that the cancer rate continues to rise in the U.S.? When will our government agencies that are responsible for the recognition, prevention and control of radiation hazards from low exposure to ionizing radiation awaken to their tasks of serving the people?

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STATE OF NEW YORK
PUBLIC SERVICE COMMISSION

OPINION NO. 78-13

CASE 26529 - POWER AUTHORITY OF THE STATE OF NEW YORK - Moses-Massena 230 kV Transmission Line, Massena-Moses 765 kV Transmission Line, and Massena-Quebec 765 kV Transmission Line

CASES 26529 and 26559 - Common Record Hearings on Health and Safety of Extra-High Voltage Transmission Lines.

OPINION AND ORDER DETERMINING HEALTH AND SAFETY ISSUES, IMPOSING OPERATING CONDITIONS, AND AUTHORIZING, IN CASE 26529, OPERATION PURSUANT TO THOSE CONDITIONS

Issued: June 19, 1978

CASE 26529/26559 - Common Record Hearings on the Health and Safety of 765 kV Transmission Lines

Amicus Curiae Brief

Submitted By

ANDREW A. MARINO

TABLE OF CONTENTS

6

VETERANS ADMINISTRATION
HOSPITAL
IRVING AVENUE AND UNIVERSITY PLACE
SYRACUSE, NEW YORK 13210

IN REPLY
REFER TO:

January 16, 1976

Dr. R. A. Helliwell
Radioscience Laboratory
Stanford University
Stanford, California 94305

Dear Dr. Helliwell:

I have read with much interest your recent article in the Journal of Geophysical Research (volume 80, page 4249). Would you kindly send me a reprint?

The New York Public Service Commission is currently holding a hearing to determine the environmental compatibility of high voltage transmission lines (765 kv). My collegue, Dr. Robert Becker, and myself, are serving as expert witnesses for the State in the area of the biological impact of the proposed lines. For those of us who are unfamilar with your field, your report appears to describe an effect which ought to be studied further from the environmental impact viewpoint. For instance, if the total radiated power were given (at each harmonic) of the 70 mile transmission line being proposed, would it be possible to compute the magnitude of the electron and energy shower on the ionosphere that would result? Also, how does the magnitude of the stimulated emission compare to that normally present? If you could provide me with some rough answere to these questions, it would help greatly in our preliminary evaluation of the environmental significance of the effect which you have described.

There is one further point on which I would like to seek information. I would expect that the largest power radiated from a power system would be at 60 hertz. Would the effect which you have observed therefore be stronger at this frequency, as compared with that seen at the various harmonics?

Sincerely,

Andrew A. Marino, Ph.D.
Biophysicist

STANFORD ELECTRONICS LABORATORIES

STANFORD, CALIFORNIA 94305

20 January 1976

Andrew A. Marino
Veterans Administration Hospital
Irving Avenue and University Place
Syracuse, New York 13210

Dear Dr. Marino:

Enclosed is a reprint of our recent article in the Journal of Geophysical Research.

We have had great difficulty modeling the Canadian power system for purposes of computing the radiated power. One of the problems is that we do not know the ground currents, which determine the effective area of the antenna (assumed to be a loop). On the other hand observations of the power line radiation in the magnetosphere suggests that it is no stronger than the natural emissions which occur there. One of the observed functions of the power line radiation is to phase lock the oscillations which are started by other sources.

We would not expect that a new transmission line would produce a noticeable change in the electron precipitation from the magnetosphere. Although we indicate in our paper that power line radiation exercises control of the radiation belt, I think that it is fair to say that in the absence of power line radiation the belt intensity would simply build up to the point where more spontaneous emissions would occur and we would observe about the same amount of total precipitation.

We expect the power that is radiated at 60 Hz to be relatively small because of the very long wavelength involved. The efficiency of the power system as a radiator increases rapidly with frequency and hence we would expect the peak in radiation to occur at some fairly high harmonic, depending upon the wave form on the system. We have noticed in our own studies that the principal power line activity in the magnetosphere seems to be concentrated in the range 2000 to 6000 Hz.

Very truly yours,

R. A. Helliwell
Professor

RAH:ms
Enc.

STANFORD ELECTRONICS LABORATORIES

STANFORD, CALIFORNIA 94305

March 1, 1977

Mrs. Louise B. Young
755 Sheridan Road
Winnetka, Illinois 60093

Dear Mrs. Young:

Thank you for your letter of February 6, 1977. Although power line radiation may cause X-rays, I believe it is not a significant factor, on the average, for a simple reason. If man-made inputs to the magnetosphere were diminished, the concentration of radiation belt particles would simply increase until natural wave activity produced the same average precipitation as before. However a detailed understanding of this question must await further research.

Very truly yours,

RAHelliwell

R. A. Helliwell
Professor

RAH:ms

STANFORD ELECTRONICS LABORATORIES

STANFORD, CALIFORNIA 94305

1 March 1977

STATE OF NEW YORK
SERVICE COMMISSION
DATE 3-24-77
CASE NO. 26529/59
EX. R-5

Mr. Henry J. Nowak
Niagara Mohawk Power Corp.
300 Erie Blvd. West
Syracuse, N.Y. 13202

Dear Mr. Nowak:

In response to your letter of January 27, 1977, I have examined the reference testimony by Andrew A. Marino. Although his description of my experiments is generally correct, his conclusions regarding the biological effects of transmission line radiation are, in my opinion, not supported by data. My specific comments are as follows.

On page 47, lines 5, 6, and 7 is the statement "Electrons which have surrendered energy to the wave drop out of the magnetosphere and rain down on the ionosphere." This statement is not strictly correct since only a small fraction of the electrons that exchange energy with the wave are actually scattered into the loss cone and hence reach the ionosphere.

On page 47, it is correctly stated that there is a normal background count of electrons on the ionosphere due to galactic sources. However there is also another source, namely natural electromagnetic noise in the magnetosphere. This source produces a more or less continuous drizzle of electrons into the ionosphere. Whistlers excited by lightning discharges add additional waves in whistler ducts which produce enhanced counts of electrons and hence Bremsstrahlung X-rays. A recent paper in <u>Science</u> summarizes some of the pertinent results on precipitation (Energetic Radiation Belt Precipitation: A Natural Depletion Mechanism for Stratospheric Ozone, by Richard M. Thorne, <u>Science</u>, 21 January 1977, pp. 287-89). Figure 2 of that paper shows, for example, that the galactic cosmic ray flux produces most of the electrons below an altitude of about 25 km. Above that altitude Bremsstrahlung X-rays, intense polar cap absorption events and electron precipitation become important. However the electron precipitation events from all sources are confined mainly to altitudes above 60 km. The X-rays produced by these precipitation events reach lower altitudes but do not penetrate significantly below 20 km altitude.

On pages 48-49 is a statement that the X-rays would produce UV radiation that might cause biological effects. I do not have an answer as to how much UV is produced by the X-rays. However I would be surprised if it were of any significance compared to solar UV.

The critical factor, however, is the following. Disturbances created by radio waves are not expected in any way to alter the average release of energy from the radiation belts. Their role is simply one of control. If all of the transmitters on the earth's surface, including the power transmission lines, were turned off, then I would expect that the radiation belt would increase in intensity until various natural wave generation mechanisms acted to return the precipitation rate to its previous value. Therefore I conclude that the introduction of power lines capable of radiating high harmonics should not alter the average intensity of any radiation that might reach the surface of the earth.

The experiments we are performing are basically perturbation experiments which are aimed at obtaining new knowledge of the physical processes going on in the magnetosphere. We do not expect that these experiments will have any significant effect on the average behavior of the radiation belts or on the outputs therefrom. With regard to weather modification, there is too little knowledge about any of the physics to speculate on that matter. However, we hope that our experiments may contribute to an improved understanding of the relationship of climate to the radiation belts. I see no reason to be concerned about the effects on our environment of power line radiation that enters the magnetosphere.

Published research papers dealing with power line radiation in the magnetosphere are listed in the attached bibliography.

Very truly yours,

RAHelliwell

R. A. Helliwell
Professor

RAH:ms
Enc.

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Attachment 3

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-6526

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

W. E. Ehrensperger
Senior Vice President & Group Executive-
Power Supply

the southern electric system

April 29, 1981

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M18
Log: GN-129

Attention: Mr. James P. O'Rielly

Reference: Vogtle Electric Generating Plant-Units 1 and 2
50-424 and 50-425; Containment Liner Penetrations

Gentlemen:

Georgia Power Company verbally notified Mr. Paul Kellogg of the Nuclear Regulatory Commission on March 26, 1981, of a potentially significant deficiency concerning welds on containment liner penetrations.

Georgia Power Company has conducted weld inspections of the penetrations in question and has begun an engineering evaluation to determine the significance of this problem. Additionally, Georgia Power Company is reviewing the need to conduct a quality assurance audit at the vendor manufacturing facility. It is expected these evaluations will be concluded in June, 1981. Georgia Power Company expects to submit a final report to the Commission concerning the reportability of this potential deficiency on or before July 17, 1981.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

W. E. Ehrensperger

W. E. Ehrensperger
Senior Vice President & Group Executive-
Power Supply

CWH:skf

xc: See Page 2

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
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Post Office Box 4545
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Vogtle Project

July 21, 1981

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG02-M18
Log: GN-139

Ref.: Vogtle Electric Generating Plant-Units 1 and 2, 50-424 and 50-425;
Containment Liner Penetrations; letter GN-129, dated April 20, 1981.

Attention: Mr. James P. O'Reilly

Gentlemen:

In our previous letter, Georgia Power Company indicated a final response concerning the reportability of the above-referenced item would be submitted to the Commission on or before July 17, 1981. Georgia Power Company has recently received an engineering evaluation concerning these penetrations, which preliminarily indicated that the above item was not reportable under 10 CFR 50.55e or Part 10 CFR 21. Currently this evaluation is being reviewed by Georgia Power Company.

Georgia Power Company now expects to submit a final report to the Commission by August 7, 1981, concerning the reportability of the above item.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

Doug Dutton
Project General Manager

RCW:sew

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D.C. 20555

J. H. Miller, Jr.
R. J. Kelly
R. E. Conway
G. F. Head
R. H. Pinson
C. F. Whitmer
D. L. McCrary
R. A. Thomas
J. A. Bailey
O. Batum
D. E. Dutton
K. M. Gillespie
E. D. Groover
L. T. Gucwa
B. L. Lex
C. R. Miles, Jr.
H. C. Nix
R. W. Staffa
J. L. Vota

MF

Georgia Power Company
Post Office Box 4545
Atlanta, Georgia 30302
Telephone 404 522-6060

Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202
Telephone 205 870-6011

Vogtle Project

August 7, 1981

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II, Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M18
Log: GN-141

Ref.: Vogtle Electric Generating Plant - Units 1 and 2
50-424, 50-425; Containment Liner Penetrations - Unit 1

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has concluded its evaluation concerning the investigation regarding the reportability of the above-referenced item. It has been concluded that the welding defects associated with the containment liner penetrations would not adversely affect the safety of operations and are not reportable under 10CFR50.55(e) or Part 10CFR21.

This response contains no proprietary information and may be placed in the NRC public document room upon receipt.

Yours truly,

Doug Dutton
Project General Manager

CWH:sew

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D.C. 20555

J. H. Miller, Jr.
R. J. Kelly
R. E. Conway
G. F. Head
R. H. Pinson
C. F. Whitmer
D. L. McCrary
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E. D. Groover
L. T. Gucwa
B. L. Lex
C. R. Miles, Jr.
H. C. Nix
R. W. Staffa
J. L. Vota

Attachment 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

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DOF

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-424/83-15 AND 50-425/83-15

This refers to the routine safety inspection conducted by Mr. E. H. Girard, of this office on September 27 - 30, 1983, of activities authorized by NRC Construction Permit Nos. CPPR-108 and CPPR-109 for the Vogtle facility. Our preliminary findings were discussed with Mr. H. H. Gregory, III, Project Construction Manager, at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to violate NRC requirements. This item and references to pertinent requirements are listed in the Notice of Violation enclosed herewith as Appendix A. Elements to be included in your response are delineated in Appendix A.

We have examined actions you have taken with regard to previously identified enforcement matters and unresolved items. The status of these items is discussed in the enclosed report.

One new unresolved item is identified in the enclosed inspection report. This item will be examined during subsequent inspections.

In accordance with 10 CFR 2.790(a), a copy of this letter, its enclosures, and your reply will be placed in NRC's Public Document Room upon completion of our evaluation of the reply. If you wish to withhold information contained in the inspection report, please notify this office by telephone and include a written application, to withhold information contained therein, in your response. Such application must be consistant with the requirements of 2.790(b)(1).

The responses directed by this letter and the enclosures are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

for Richard C. Lewis, Director
Division of Project and
Resident Programs

Enclosures:
1. Appendix A, Notice of Violation
2. Inspection Report Nos. 50-424/83-15 and 50-425/83-15

cc w/encls:
H. H. Gregory, III, Construction Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Vice President and General Manager
G. Bockhold, Jr., Plant Manager

APPENDIX A

NOTICE OF VIOLATION

Georgia Power Company
Vogtle

Docket No. 50-424
License No. CPPR-108

As a result of the inspection conducted on September 27 - 30, 1983, and in accordance with the NRC Enforcement Policy, 47 FR 9987 (March 9, 1982), the following violation was identified.

> 10 CFR 50, Appendix B, Criterion V, as implemented by 17.1.5 of the PSAR, requires that activities affecting quality be prescribed by documented instructions, procedures or drawings assuring that the activities are accomplished. 10 CFR 50.55(a) requires inspection of reactor coolant pressure boundary piping welds in accordance with the ASME Boiler and Pressure Vessel Code, Section III (hereafter "the Code"). The Code requires penetrant examination (an activity affecting quality) of reactor coolant pressure boundary piping welds.
>
> Contrary to the above, the licensee's instruction sheet for specifying examinations of reactor coolant pressure boundary weld 119-W-05 (on Isometric Drawing 1K4-1201-119-02 R10) failed to specify performance of the Code required penetrant examination.
>
> This is a Severity Level IV Violation (Supplement II).

Pursuant to the provisions of 10 CFR 2.201, you are hereby required to submit to this office within thirty days of the date of this Notice, a written statement or explanation in reply, including: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Date: OCT 24 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-424/83-15 and 50-425/83-15

Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Docket Nos.: 50-424 and 50-425

License Nos.: CPPR-108 and CPPR-109

Facility Name: Vogtle 1 and 2

Inspection at Vogtle site near Waynesboro, Georgia

Inspector: for E. H. Girard — 10/19/83 Date Signed

Approved by: J. J. Blake, Section Chief — 10/19/83 Date Signed
Engineering Program Branch
Division of Engineering and Operational Programs

SUMMARY

Inspection on September 27-30, 1983

Areas Inspected

This routine, unannounced inspection involved 30 inspector-hours on site in the areas of licensee action on previous enforcement matters and reactor coolant loop piping welding.

Results

Of the two areas inspected, no violations or deviations were identified in one area; one apparent violation was found in one area (Violation - Failure to Provide for Code Required Penetrant Examination - paragraph 5.d).

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *H. H. Gregory, III, Project Construction Manager
   *M. Googe, Assistant Project Construction Manager
   *G. A. McCarley, Project Compliance Coordinator
   *E. D. Groover, QA Site Manager
   C. Sarver, Senior QA Engineer
   *L. T. Ellgass, Associate QA Field Representative

   Other licensee employees contacted included QC inspectors, construction craftsmen, and office personnel.

   Other Organization

   J. P. Runyan, QA Manager, Pullman Power Products

   NRC Resident Inspector

   W. F. Sanders

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on September 30, 1983, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the violatin listed below. Although the concern addressed by item 424, 425/83-15-02 was mentioned during the exit interview, the item was formally identified as an unresolved item in a telephone call to the licensee on October 14, 1983. No dissenting comments were received from the licensee.

   Violation 424/83-15-01, Failure to Provide for Code Required Penetrant Examination, paragraph 5.d.

   Unresolved Item 424, 425/83-15-02, Erroneous Response Statement, paragraph 3.a.

3. Licensee Action on Previous Enforcement Matters

   a. (Open) Violation (424, 425/81-08-02): Undersize Structural Welds. The inspector examined examples of structural welds similar to those addressed in the subject violation to verify their compliance with size requirements; and examined the licensee's specification X2AP01, Section 5.1, to determine whether the welds originally questioned were within

the specification as stated in the licensee's letter of response dated September 2, 1981. Structural welds examined by the inspector were as follows:

- Drawing AX2D11F009R10

  Connections 23, 24, 25, 30, 59, and 60 in Control Building Room A23

- Drawing AX2D11F011R7

  Connections 325 and 326 in Control Building Room A18

The inspector's review of the specification reference described in the licensee's response letter found that the response was apparently in error in the statement (relative to the welds that were the subject of the original violation) - "These welds were found to be within the specification requirements of X2AP01 - Section C5.1.F.8.d, Revision 4," which states the following: "Undercut (underfill) not exceeding 3/32 inch shall be acceptable for the full length of the weld". It did not appear that the specification contained such a statement. The inspector indicated his concern regarding the erroneous response statements and its source. The licensee was unable to immediately provide any explanation. The licensee was informed that the source of the subject response statement and its significance would be examined further in subsequent inspection as an unresolved item, identified 424, 425/83-15-02, Erroneous Response Statement. The licensee was also informed that the original violation would remain open pending their submittal of a clarifying response and its acceptable review by Region II in a subsequent inspection.

b. (Open) Unresolved Item (424, 425/82-23-01): Article 3 Basis. This item documented an NRC inspector's concern as to the licensee's basis for electing to use Article 3 of ASME Section V (74S75) as the source of examination requirements for radiography of containment liner welds. This item was last addressed in inspection report 424, 425/83-02. During that inspection the NRC inspector further examined and discussed concerns relative to this item with the licensee. As stated in the report for that inspection, it was the inspector's understanding that the licensee would research this area further and provide additional information to indicate whether applicable Code requirements had been violated. When questioned regarding the item during the current inspection the licensee was unable to provide any further information. This was discussed with the Plant Project Manager, who stated that they would assure that the information was obtained for review and discussion in a subsequent NRC inspection. The item will remain open pending the Region II's review of the information to be provided.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. A new unresolved item identified during this is discussed in paragraph 3.a.

5. Reactor Coolant Loop Piping Welding - Unit 1

The inspector examined welding and welding related activities for reactor coolant loop, (RCL), piping activities to determine their compliance with SAR commitments and Regulatory requirements, including the requirements of the applicable code, ASME Section III (77W77). The inspector's examinations and findings are described below.

a. Welding Procedure Specifications (55171)

The inspector reviewed examples of the welding procedure specifications, (WPS's), for RCL piping and their qualification records to verify their compliance with procedural content requirements of the code and code qualification test requirements (including proper testing and results). The following procedures and their qualification records were reviewed:

250-III/I-8-KI-A1
24-III/I-8-KI-12
32-III/I-8-12
29-III/I-8-13-1

b. Observation of Welding Activities (55173)

The inspector observed welding related activities in progress on the following piping welds:

WELD NO.	ISO	PIPE SIZE
119-W-05	1X4-1201-119-02R10	3" Dia. x .438 wall
005-W-03	1X4DL4A17-R10	31" Dia. x 2.5" wall
005-W-04	1X4DL4A17-R10	31" Dia. x 3.25" wall
005-W-02	1X4DL4A17-R10	31" Dia. x 2.5" wall
006-W-02	1X4DL4A17-R10	31" Dia. x 2.5" wall
006-W-03	1X4DL4A17-R10	31" Dia. x 3.5" wall

As applicable to the work in progress the welding was examined to determine whether:

(1) Work was conducted in accordance with a document which coordinates and sequences operations, references procedure, establishes hold points, and provides for production and inspection approval.

(2) Weld identification and location were as specified.

(3) Applicable drawings were at the work station and readily available.

(4) Welding procedure essential variables were complied with.

(5) Welding technique and sequence requirements were adhered to.

(6) Base and welding materials were properly identified, verified, and traceable.

(7) Weld joint geometry was as specified.

(8) Alignment was in accordance with requirements.

(9) Temporary attachments were attached by qualified welders.

(10) Purge gas was in accordance with procedure requirements.

(11) Preheat and interpass temperatures were in accordance with procedures.

(12) Welding equipment was in good condition.

(13) Interpass cleaning was in accordance with applicable procedures.

(14) Weld history records were adequate.

(15) Inspection personnel were properly qualified.

In observing welding-related activities the inspector noted the following items of concern relative to control of welding:

- The welding amperage set for weld C05-W-04 on one automatic welding machine was noted generally at the upper limit of the procedure and on one occasion it exceeded the procedure limit by 1 amp. (300 versus the 299 limit). The inspector informed licensee and the welding operators were cautioned.

- The welding machine for weld 119-W-05 was located up many flights of stairs and several hundred feet from where the welding was performed. In questioning, the welder stated he had never seen amperage checks made with tong meters or similar devices. The inspector informed the licensee that he was concerned that allowing such great distances between machines and welds, and the lack of amperage checks on welding, could lead to unsatisfactory welds. The licensee stated they did perform surveillances on welding amperage and indicated they would check further into the matter and take action if necessary. The inspector did not consider the above concerns significant enough, by themselves, to warrant issuance of a violation.

c. Welder Qualifications (55177)

The inspector checked the licensee's procedures and reviewed the qualification records and qualification status list relative to the welders and welding operators observed welding on the welds identified in 5.b. above to:

(1) Determine whether procedures had been established to qualify welders and welding operators in accordance with the code.

(2) Determine whether the welders and welding operators were currently qualified.

(3) Determine the accuracy of qualification status records and their compliance with the code.

d. Nondestructive Examination (55073)

The inspector asked to observe penetrant examination of completed weld 119-W-05 on ISO 1K4-1201-119-02R10. The licensee's examiner agreed but noted that the weld process sheet (used to specify weld inspections) did not require the examination. The inspector observed the examination to determine whether the examiner complied with applicable code and procedural requirements.

The inspector noted that the applicable code required the penetrant examination and that the failure to specify the examination in the process sheet was considered a nonconformance with the requirements of 10 CFR 50, Appendix B, Criterion V. The inspector informed the licensee that this nonconformance would be identified as violation 424/83-15-01, Failure to Provide for Code Required Penetrant Examination.

Within the areas examined, one violation was identified, as described in d above.

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-7726

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

Georgia Power
the southern electric system

D. O. Foster
Vice President and General Manager
Vogtle Project

November 28, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG10
Log: GN-288

Reference: 50-424/83-15, 50-425/83-15

Attention: Mr. R. C. Lewis

Gentlemen:

The Georgia Power Company wishes to submit the following information concerning the violation discussed in your inspection report 50-424/83-15 and 50-425/83-15:

Violation 50-424/83-15-01, "Failure to Provide for Code Required Penetrant Examination" - Severity Level IV.

(1) Georgia Power Company acknowledges the discrepancy identified in this violation.

(2) The reasons for the violation are as follows:

  a. The piping contractor's (Pullman Power Products) engineer erroneously prepared the wrong process sheets for the welds on isometric drawing 1KA-1201-119-02, Rev.10.

  b. The piping contractor's QA document reviewer failed to detect the error in the process sheets.

(3) The lack of penetrant examination requirements in process sheets associated with isometric drawing 1KA-1201-119-02, Rev. 10, is considered to be an isolated case. All other primary loop piping isometrics (twenty drawings in all) and all associated process sheets were examined and found to provide for penetrant examinations. Isometric drawing 1KA-1201-119-02, Rev. 10, and all associated process sheets, including the one for reactor coolant pressure boundary weld 119-W-05, have been revised to assure that required penetrant examinations are performed.

(4) Pullman Power Products engineering and QA personnel have been admonished to review process sheets closely to assure compliance with all applicable requirements.

(5) Full compliance with applicable regulatory and code requirements was achieved on November 16, 1983.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

R J Kelly
for D. O. Foster

REF/DOF/cc

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D. C. 20555

xc: R. J. Kelly
R. E. Conway
G. F. Head
J. T. Beckham
D. N. MacLemore
D. E. Dutton
W. F. Sanders
R. H. Pinson
B. M. Guthrie
R. A. Thomas
J. A. Bailey
O. Batum
H. H. Gregory
C. W. Hayes
E. D. Groover
L. T. Gucwa
M. Malcom
G. Bockhold
P. D. Rice
J. L. Vota

Attachment 6

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

NOV 07 1983

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Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-424/83-18 AND 50-425/83-18

This refers to the routine safety inspection conducted by Mr. W. F. Sanders of this office on August 16 - September 30, 1983, of activities authorized by NRC Construction Permit Nos. CPPR-108 and CPPR-109 for the A. W. Vogtle facility. Our preliminary findings were discussed with Mr. H. H. Gregory, III, Construction Project Manager, at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to violate NRC requirements. This item and references to pertinent requirements are listed in the Notice of Violation enclosed herewith as Appendix A. Elements to be included in your response are delineated in Appendix A.

In accordance with 10 CFR 2.790(a), a copy of this letter, its enclosures, and your reply will be placed in NRC's Public Document Room upon completion of our evaluation of the reply. If you wish to withhold information contained in the inspection report, please notify this office by telephone and include a written application, to withhold information contained therein, in your response. Such application must be consistent with the requirements of 2.790(b)(1).

The responses directed by this letter and the enclosures are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

Hugh C. Dance

Hugh C. Dance, Chief
Project Branch 2
Division of Project and
Resident Programs

Enclosures:
1. Appendix A, Notice of Violation
2. Inspection Report Nos. 50-424/83-18
and 50-425/83-18

cc w/encls:
H. H. Gregory, III, Construction
Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Vice President
and General Manager
G. Bockhold, Jr., Plant Manager

APPENDIX A

NOTICE OF VIOLATION

Georgia Power Company
A. W. Vogtle Units 1 and 2

Docket Nos. 50-424 and 50-425
License Nos. CPPR-108 and CPPR-109

As a result of the inspection conducted on August 16 - September 30, 1983, and in accordance with the NRC Enforcement Policy, 47 FR 9987 (March 9, 1982), the following violation was identified.

10 CFR 50, Appendix B, Criterion IX as implemented by paragraph 17 of the PSAR requires measures be established to assure that special processes, including nondestructive testing, be controlled and accomplished in accordance with applicable codes. ASME B&PV Code, 1971 Edition with Summer 1971 Addenda has been identified as the applicable code for the Closure Head Weld Metal Cladding. ASME B&PV Code Section III, Paragraph NB 5352 describes acceptance standards for relevant indications.

Contrary to the above, adequate measures had not been established to assure that special processes, including nondestructive testing, were controlled and accomplished in accordance with applicable codes in that the re-examination of the cladding by Liquid Penetrant methods revealed linear indications and rounded indications greater than the acceptable limits of the applicable code. This nondestructive examiniation was previously performed and certified by the nuclear steam system supplier manufacturer.

This is a Severity Level V Violation (Supplement II).

Pursuant to the provisions of 10 CFR 2.201, you are hereby required to submit to this office within thirty days of the date of this Notice, a written statement or explanation in reply, including: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Date: NOV 07 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-424/83-18 and 50-425/83-18

Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Docket Nos.: 50-424 and 50-425

License Nos.: CPPR-108 and CPPR-109

Facility Name: Alvin Vogtle 1 and 2

Inspection at Vogtle Nuclear Station, Waynesboro, Georgia

Inspector: John F. Rogge for — 11/1/83
W. F. Sanders — Date Signed

Approved by: V. Panciera — 11/4/83
V. Panciera, Chief, Projects Section 2B — Date Signed
Division of Project and Resident Programs

SUMMARY

Inspection on August 16 - September 30, 1983

Areas Inspected

This routine unannounced inspection involved 216 inspector-hours on site in the areas of closure head control rod drive mechanism welding; inspection of piping; closure head cladding welding; construction progress in primary containment Units 1 and 2, auxiliary building and control building.

Results

Of the four areas inspected, no violations or deviations were identified in three areas; one violation was found in one area (closure head cladding welding - paragraph 6).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*H. H. Gregory III, Project Construction Manager
*M. H. Googe, Assistant Project Construction Manager
*E. D. Groover, Quality Assurance Site Manager
*R. W. McManus, Manager of Quality Control
*W. C. Lyon, Inspection Supervisor
*G. A. McCarley, Project Compliance Coordinator
*J. R. Petro, Sr., Quality Assurance Field Representative
*R. H. Robinson, Engineering Supervisor

Other licensee employees contacted included construction craftsmen, technicians, mechanics, and office personnel.

Other Organizations

*J. B. McLachlan, Project Field Engineer, Bechtel Power Corporation
*J. Rudd, Site Quality Assurance Supervisor, Bechtel Power Corporation
*D. Wieland, Site Manager, Westinghouse Electric Corporation
*W. Reed, Site Engineer, Westinghouse Electric Corporation

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on September 29, 1983, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings.

(Open) Violation 424, 425/83-18-01 "Failure to Control a Special Process in the Nondestructive Examination (NDE) of the Closure Head Cladding" - Paragraph 6.

3. Licensee Action on Previous Enforcement Matters

Not inspected.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Independent Inspection Effort

Periodic inspections were made throughout this reporting period in the form of general type inspections in different areas of both facilities. The areas were selected on the basis of the scheduled activities and varied to provide wide coverage. Observations were made of activities in progress to note defective items or items of noncompliance with the required codes and regulatory requirements. On these inspections, particular note was made of the presence of quality control evidence in the form of available process sheets, drawings, material identification, material protection, performance of tests, and housekeeping.

Interviews were made with craft personnel, supervisors, coordinators, quality control inspectors, and others and they were available in the work areas. Observations were made in the following areas: Primary Containment Structures 1 and 2, Auxiliary Building, Control Building, and Electrical Storage.

No violations or deviations were identified.

6. Closure Head Cladding

A visual inspection of Unit 1 and Unit 2 Reactor Closure Heads Internal Cladding disclosed questionable surface conditions, which did not appear to meet the criteria for performing a meaningful Liquid Penetrant Examination (LPE). The following surface conditions were apparent:

a. Streaks of weld flux/slag were observed in the valleys between some of the weld beads. The edges of the weld beads in these areas did not merge smoothly with each other.

b. Arc strikes were observed in several areas.

c. Foreign material in the form of very small droplets of molten metal were observed in several areas.

These observations and concerns were described to the Licensee's Quality Control Department, which subsequently inspected the cladding and reported the results to the Nuclear Steam System Supplier (NSSS). This was followed later with representative sample areas selected and reinspected by a NDE Specialist representing the NSSS manufacturer. The inspector witnessed the examination and results which are described below:

Unit 1 Two areas approximately one foot square each were examined by the Solvent Liquid Penetrant Examination (LPE) method in accordance with ASME Section V. This revealed a circular indication of 7/32" in one area and ¼" circular in the other. (ASME Section III allows 3/16" round maximum).

Unit 2 One area approximately 18" x 2' was examined by the LPE method. This revealed several indications which exceeded the Code allowable. The most severe indication was approximately ½" long linear.

These conditions indicate a failure to establish adequate measures to assure that special processes including non-destructive testing are controlled and accomplished in accordance with applicable codes and is in violation of 10 CFR 50 Appendix B, Criterion IX. This violation is identified as 424, 425/83-18-01.

7. Investigation of Civil Work by Licensee

On August 5, 1983, the licensee notified the Regional Office that several employees of the Civil Contractor had informed them that they had concerns regarding the quality of certain work at the Vogtle Plant. They also stated that these concerns would be investigated by a special team and appropriate corrective action taken. A four-member team consisting of a team leader, two team members, and General Counsel were formed at the direction of the Senior Vice-President, Engineering and Construction, and given the principal duties as follows:

a. Obtain formal statements from the concerned employees.

b. Identify the work in question.

c. Evaluate the concerns utilizing document reviews, interviews, engineering studies, non-destructive examinations and destructive examinations as appropriate.

d. Recommend corrective action.

e. Prepare complete documentation of the investigation.

This was considered a licensee identified matter and was monitored throughout the investigation period by the NRC Resident Inspector. This was followed by a review of the documentation to evaluate the depth and scope of the inquiry and review corrective actions relative to safety implications.

The records show that twenty-three (23) individual quality concerns were reported by eleven (11) of the thirty-four (34) crew members. The investigation revealed that in thirteen (13) of the twenty-three (23) cases, the concern was either allowed by design documents or had been previously reported and investigated through the Quality Control System and deviation reports. The remaining concerns were reviewed for safety significance, additional testing, verification of design requirements and corrective actions. This review was supplemented by a review of randomly selected transcripts of interviews conducted under oath and recorded by court reporters.

No violations or deviations were identified.

8. Control Rod Drive Mechanism Welding

An inspection was made of the activities related to the welding of the mechanism Lower Canopy Seals on the Unit 1 closure head. The welding procedure employs an automatic gas shielded tungsten arc that moves 360° around the fixed position weld joint by means of a motor-driven carriage. A special weld joint configuration employs a preplaced "Y" type consumable insert. The following documents were reviewed:

Weld Procedure Specification (WPS)-80.5.25, Rev. E

Procedure Qualification Record (PQR) 151

Liquid Penetration Report Per Nisco ES-100-2

Visual Inspection Report of Macro Examination

Weld Procedure Specification for Repair (SWPS)-132-2, Rev. J

Procedure Qualification Record PQR-152, Rev. A

It was noted that the procedure qualification was made by welding six successful test assemblies representing the geometric configurations and using the appropriate material combination in accordance with ASME Section III, NB 4366 and NB 4367. Visual examinations of the Qualification cross-section samples and visual examination of a small repair was made by the inspector.

No violations or deviations were identified.

9. IE Bulletin Followup

79-BU-15 (Closed) Deep Draft Pump Deficiencies - On September 11, 1979 the licensee submitted a response to IE Bulletin 79-15. Vollmer, NRR, memo dated June 22, 1981, stated that this subject will be included in the normal licensing reviews. This item is closed.

Attachment 7

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-7726

Mailing Address
Post Office Box 4545
Atlanta, Georgia 30302

Georgia Power

D. O. Foster
Vice President and General Manager
Vogtle Project

December 14, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG10
Log: GN-295

Reference: 50-424/83-18, 50-425/83-18

Attention: Mr. R. C. Lewis

Gentlemen:

The Georgia Power Company wishes to submit the following information concerning the violation discussed in your inspection report 50-424/83-18 and 50-425/83-18:

Violation 50-424,425/83-18-01, "Failure to Control a Special Process in the Nondestructive Examination (NDE) of the Closure Head Cladding" - Severity Level IV.

(1) Georgia Power Company acknowledges the discrepancies identified in this violation.

(2) The reasons for the violation are as follows:

   a. The contract with the reactor vessel manufacturer, Combustion Engineering, called for liquid penetrant examination (PT) of the Unit 1 closure head cladding surface prior to gritblasting. PT is normally done in the finished condition but is not recommended on gritblasted surfaces. Combustion Engineering maintains that the three unacceptable rounded indications found in the Unit 1 closure head by PT at the site were not present at the surface during the ASME Boiler and Pressure Vessel Code acceptance PT at the Combustion Engineering shop. These indications were found, upon excavation, to be slag inclusions which apparently opened due to the gritblasting operation after the liquid penetrant examination. The gritblasting was considered post-examination cleaning by the Authorized Nuclear Inspector, Combustion Engineering, and Westinghouse Power Corporation, the NSSS supplier, in accordance with Paragraph NB-5113, Section III, of the ASME Code. The Unit 2 closure head was not gritblasted as Combustion Engineering discontinued this practice in 1978. No unacceptable indications were found in the Unit 2 closure head.

   b. Georgia Power Company cannot determine how or exactly when surface defects which appear to be arc strikes were inflicted on the Unit 2 closure head. They apparently occurred at some time between the PT examination at the Combustion Engineering shop and the inspection at the jobsite.

(3) The closure head manufacturer, Combustion Engineering, was contracted to perform the following corrective actions:
   a. The Unit 1 closure head cladding was thoroughly cleaned and buffed to minimize minor surface imperfections. 100% of the surface was then liquid penetrant (PT) examined. Three unacceptable rounded indications, identified as slag inclusions, were removed by grinding. Two of the indications were on a slag line in the same plane and extended to the base metal but did not violate the pressure boundary. The cladding was then weld repaired using approved procedures, heat treated, liquid penetrant examined (repaired areas only), and stress relieved.
   b. The Unit 2 closure head was not gritblasted as was the Unit 1 head. Some discoloration of the surface, however, raised some questions on the acceptability of some areas of the cladding. Therefore, the Unit 2 cladding surface was also thoroughly cleaned, buffed, and liquid penetrant examined over 100% of its area. No unacceptable indications were found. The apparent arc strikes and some slag pockets were effectively removed during the surface cleaning and buffing.
   c. All of the cleaning, buffing, repairing, and PT operations performed on the Unit 1 and Unit 2 closure heads have been properly documented. The documentation was reviewed and approved by the Authorized Nuclear Inspector. Any questions which arose regarding procedures and materials were closely scrutinized and resolved. All corrective actions performed by Combustion Engineering were witnessed by Georgia Power.

(4) Action to prevent further violations is not applicable in this case. Combustion Engineering has discontinued the practice of gritblasting closure heads following PT, but they have not manufactured a reactor vessel since 1978.

(5) Full compliance with applicable regulatory and code requirements was achieved on December 5, 1983.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D O Foster

D. O. Foster

REF/DOF/cc

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D. C. 20555

xc:

R. J. Kelly	D. E. Dutton	J. A. Bailey	L. T. Gucwa
R. E. Conway	W. F. Sanders	O. Batum	M. Malcom
G. F. Head	R. H. Pinson	H. H. Gregory, III	G. Bockhold
J. T. Beckham	B. M. Guthrie	C. W. Hayes	P. D. Rice
D. N. MacLemore	R. A. Thomas	E. D. Groover	J. L. Vota

Attachment 8

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

AUG 3 1 1983

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-424/83-16 AND 50-425/83-16

This refers to the routine safety inspection conducted by Mr. W. P. Kleinsorge of this office on August 9 - 12, 1983, of activities authorized by NRC Construction Permit Nos. CPPR-108 and CPPR-109 for the Vogtle facility. Our preliminary findings were discussed with Mr. H. H. Gregory, III, Project Construction Manager, at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to violate NRC requirements. This item and references to pertinent requirements are listed in the Notice of Violation enclosed herewith as Appendix A. Elements to be included in your response are delineated in Appendix A.

In accordance with 10 CFR 2.790(a), a copy of this letter, its enclosures, and your reply will be placed in NRC's Public Document Room upon completion of our evaluation of the reply. If you wish to withhold information contained in the inspection report, please notify this office by telephone or include a written application, to withhold information contained therein, in your response. Such application must be consistent with the requirements of 2.790(b)(1).

The responses directed by this letter and the enclosures are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

SEP 6 1983

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

H C Dance

H. C. Dance, Chief
Project Branch 2
Division of Project and
Resident Programs

Enclosures:
1. Appendix A, Notice of Violation
2. Inspection Report Nos. 50-424/83-16 and 50-425/83-16

cc w/encls:
H. H. Gregory, III, Construction Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Vice President and General Manager
G. Bockhold, Jr., Plant Manager

APPENDIX A

NOTICE OF VIOLATION

Georgia Power Company
Vogtle

Docket Nos. 50-424 and 50-425
License Nos. CPPR-108 and CPPR-109

As a result of the inspection conducted on August 9 - 12, 1983, and in accordance with the NRC Enforcement Policy, 47 FR 9987 (March 9, 1982), the following violation was identified.

10 CFR 50, Appendix B, Criterion IX as implemented by paragraph 17.1.9 of the PSAR requires measures be established to assure that special processes including nondestructive testing be controlled and accomplished in accordance with applicable codes. ASME B&PV Code Sections V and VIII 1974 edition with addenda through summer 1975 was identified as the applicable code for the containment liner plate welding. ASME B&PV Code Section V paragraph T-233, specifies a maximum radiographic film density of 3.80 in the area of interest. ASME B&PV Code, Section VIII paragraph UW-58 requires spot radiography of butt welded joints. The heat affected zone has been interpreted to be included in the area of interest for radiography of welds.

Contrary to the above, adequate measures had not been established to assure that special processes including nondestructive testing were controlled and accomplished in accordance with applicable codes in that:

1. CB&I Procedure RTIN, Revision 4, "Radiograph Examination Procedure for Welds" does not specify that the heat affected zone is included in the area of interest.

2. As the result of the above lack of specificity, radiographic film density on accepted film SRT-8 R-2, the final acceptance radiograph for Unit 1 458A, Seam No. 2 in the containment dome, in some locations of the heat affected zone was 4.11 to 4.38.

This is a Severity Level V Violation (Supplement II).

Pursuant to the provisions of 10 CFR 2.201, you are hereby required to submit to this office within thirty days of the date of this Notice, a written statement or explanation in reply, including: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Date: AUG 3 1 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-424/83-16 and 50-425/83-16

Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Docket Nos.: 50-424 and 50-425

License Nos.: CPPR-108 and CPPR-109

Facility Name: Vogtle 1 and 2

Inspection at Vogtle site near Waynesboro, Georgia

Inspector: W. P. Kleinsorge — Aug 29 1983 Date Signed

Approved by: J. J. Blake, Section Chief — 8/29/83 Date Signed
Engineering Program Branch
Division of Engineering and Operational Programs

SUMMARY

Inspection on August 9 - 12, 1983

Areas Inspected

This routine, unannounced inspection involved 33 inspector-hours on site in the areas of construction progress (Units 1 and 2) and, steel structures and supports (Units 1 and 2).

Results

Of the two areas inspected, no violations or deviations were identified in one area; one apparent violation was found in one area (Violation - "Failure to Establish Adequate Radiography Procedure" - paragraph 6b). No deviations were found.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*H. H. Gregory, III, Project Construction Manager
*E. D. Groover, QA Site Manager
*H. W. Swain, Mechanical QC SS
*W. E. Mundy, Sr., QA Field Representative

Other licensee employees contacted included construction craftsmen, technicians, mechanics, and office personnel.

Other Organizations

*J. Mamon, PFE-QE, Bechtel Power Company Corp. (BPC)
*S. K. Thomas, PFE-Civil BPC
*C. L. Fields, Superintendent, Chicago Bridge and Iron, (CB&I)
*L. S. Savage, QA Technician CB&I
J. P. Runyan, QA Manager Pullman Power Products (PPP)

NRC Resident Inspector

W. F. Sanders

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on August 12, 1983, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings listed below. No dissenting comments were received from the licensee.

(Open) Violation 424,425/83-16-01: "Failure to Provide Adequate Radiography Procedure" - paragraph 6b.

3. Licensee Action on Previous Enforcement Matters

Not inspected.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Independent Inspection Effort

Construction Progress

The inspector conducted a general inspection of power block site to observe construction progress and construction activities such as welding, material handling and control, housekeeping, and storage.

Within the areas examined, no violations or deviations were identified.

6. Steel Structures and Supports (Units 1 and 2)

The inspector observed welding work activities and reviewed records for steel structures and supports as described below to determine whether applicable code and procedure requirements were being met. The applicable code for containment fabrication welded by CB&I is the ASME B and PV Code Section III, 1974 edition with addenda through summer 1975, and Section VIII, 1974 edition with addenda through summer 1975. The applicable code for electrical and civil structures described herein welded by Cleveland and Ingalls is AWS D1.1-75. The applicable code for HVAC supports described herein, welded by Pullman/Kenith - Fortsom Co. (P/K-F), is AWS D1.1-77. The applicable code for pipe supports described herein, welded by PPP, is AWS D1.1-79.

a. Review of Quality Program

The inspector reviewed the below listed documents to ascertain whether the structural welding program had been approved by the licensee and whether adequate plans and procedures had been established to assure that structural welding would be controlled and accomplished consistent with commitments and regulatory requirements.

Document No.	Title
GPC-GD-T-01, Rev. 9	"Nonconformance Control"
GPC-ED-T-17, Rev. 0	"Electrical Rework Request"
GPC-CD-T-16, Rev. 4	"Structural Steel and Q-Decking"
GPC-QC-T-05, Rev. 4	"Visual Inspection"
PPP-IX-59, dtd 12/21/81	"Installation and Removal of Temporary Welded Attachments To Structures Embeds and Steel Members (Specification X2P01)"
PPP-IX-50, dtd 6/15/83	"Pipe Support Field Installation and Fabrication Procedure"
P/K-F-V500, Rev. 7	"Repair/Adjustment"

b. Special Welding Activities (55158B)(Units 1 & 2)

The inspector examined special welding activities including weld repair as described below to determine whether applicable code and procedure requirements were being met.

Records of the following repair welds were examined relative to the following: welding procedure used; repair welding procedure includes all pertinent requirements; repair welding procedure qualification; repair welder performance qualification; ANI witnessed performance qualification; repair area does not exceed limits; base and filler material as specified; base material repairs documented; NDE performed, and records complete.

Ingalls

Connection No.	Unit	Drawing No.
88	1	AX2D11F018, R-5
163	2	AX2D11F017, R-7
277	2	AX2D11F017, R-7
230	2	AX2D11F017, R-7
2	1	AX2D11F014, R-5

Cleveland

Deviation No.	Unit	Drawing No.
ED-2110	1	AX2011N110, R-7
ED-1975	1	AX2011N110, R-7
ED-1973	1	AX2011N110, R-7
ED-1783	1	AX2011N110, R-7

PPP

Hanger No.	Unit
V1-1202-003-H005, R-6	1
VA-1210-047-H003, R-4	1 & 2
VA-1210-093-H028, R-4	1 & 2
V1-1202-005-H008, R-3	1
V1-1202-124-H001, R-2	1

P/K-F

Hanger No.	Unit
DS-2093101-5, R-0	2
DS-211B109-41, R-0	2
108A131-25, R-1	1
1081143-14, R-0	1
211B109-42, R-1	2

CB&I

Film No.	Unit	
SRT-163, R-1	1	
SRT-299, R-2	1	
SRT-105, R-1	1	
SRT-149, R-1	1	
SRT-79, R-1	1	
SRT-83, R-1	1	
SRT-52, R-1	1	
SRT-36, R-1	1	
SRT-8, R-2	1	
SRT-231, R-1	1	
SRT-111, R-2	2	
SRT-49, R-1	2	
SRT-0-1, R-1	2	Seam 1, 458A
SRT-9-10, R-1	2	Seam 1, 458A
SRT-254, R-1	2	
SRT-245, R-1	2	
SRT-225, R-1	2	
SRT-210, R-1	2	
SRT-176, R-1	2	
SRT-178, R-1	2	

With regard to the inspection above, the inspector noted that the radiographic film density, on accepted Film SRT-8, R-2, the final acceptance radiograph for Unit 1, 458A, Seam No. 2 in the containment dome in some locations of the Heat Affected Zone (HAZ) was 4.11 to 4.38. This is contrary to ASME B&PV Code Section V, paragraph T-233, which specifies a maximum radiographic film density of 3.80 in the area of interest.

ASME B&PV Code, Section VIII, paragraph UW-58, requires spot radiography of butt welded joints. The HAZ has been interpreted by NRC Region II to be included as part of a weld and therefore included in the area of interest. CB&I Procedure RTIN, Revision 4, "Radiograph Examination Procedure for Welds", does not specify that the HAZ is part of a weld. The lack of the above specificity resulted in the radiographic film density in the HAZ of the radiograph in question to exceed

the maximum allowed by the ASME Code. Therefore, CBI Procedure RTIN Rev. 4, is inadequate in that it does not assure examination of the total area of interest as specified in ASME Section VIII. Failure to establish adequate measure to assure that special processes including nondestructive testing are controlled and accomplished in accordance with applicable codes is a violation of 10 CFR 50 Appendix B, Criterion IX. This violation will be identified as 424, 425/83-16-01: "Failure to Establish Adequate Radiography Procedure".

c. Welder Qualification (55157B)(Units 1 and 2)

The inspector reviewed the GPC, PPP, CB&I and P/K-F programs for qualification of welders and welding operators for compliance with QA procedures and ASME Code requirements. The applicable Code for welder qualification is ASME B&PV Code, Section IX as invoked by GPC Specification X2AG06, Rev. 4 and X4AZ01, Section P.1, Rev. 8.

The following welder qualification status records and "Records of Performance Qualification Test" were reviewed relative to the repair welds listed in paragraph 6b.

WELDER SYMBOL	ORGANIZATION
FT	Ingalls
HA	Ingalls
JR	Ingalls
JN	Ingalls
ABB	Ingalls
BYY	Ingalls
GD	Cleveland
RFF	Cleveland
PI	Cleveland
RY	Cleveland
KGI	PPP
VT	PPP
DM1	PPP
FY1	PPP
RP1	PPP
CE	PPP
99	P/K-F
633	P/K-F
622	P/K-F
366	P/K-F

d. Welding Filler Material Control (55152B)(Units 1 & 2)

The inspector reviewed the GPC program for control of welding materials to determine whether materials are being purchased, accepted, stored,

and handled in accordance with QA procedures and applicable code requirements. The following specific areas were examined:

- Welding material purchasing and receiving records for the following materials were reviewed for conformance with applicable procedures and code requirements:

Type	Size	Heat, Lot/Batch No.
E-7018	3/32"	GG-074
E-7018	1/8"	095 AAA
E-7018	3/32	084 AAA

e. Visual Inspection of Welds (55155)(Unit 1)

The inspector visually examined completed and accepted welds as described below to determine whether applicable code and procedure requirements were being met.

(1) The below listed welds were examined relative to the following: location, length, size and shape; weld surface finish and appearance (including inside diameter of pipe welds when accessible); transitions between different wall thicknesses; weld reinforcement -- height and appearance; joint configuration of permanent attachments and structural supports; removal of temporary attachments; arc strikes and weld spatter; finish-grinding or machining of weld surface -- surface finish and absence of wall thinning; surface defects -- cracks, laps, and lack of penetration, lack of fusion, porosity, slag, oxide film and undercut exceeding prescribed limits.

Three welds on the north brace for TS-110247 were examined.

(2) Quality records for the above welds were examined relative to the following: records covering visual and dimensional inspections indicate that the specified inspections were completed; the records reflect adequate quality; history records are adequate.

Within the areas examined, no violations or deviations were identified except as noted in paragraph No. 6b.

333 Piedmont Avenue
Atlanta, Georgia
Telephone 404 526-7706

Mailing Address
Post Office Box 4545
Atlanta, Georgia 30302

Georgia Power

the southern electric system

D. O. Foster
Vice President and General Manager
Vogtle Project

September 30, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG10
Log: GN-263

Reference: 50-424/83-16
50-425/83-16

Attention: R. C. Lewis

Gentlemen:

The Georgia Power Company wishes to submit the following information concerning the violation discussed in your inspection report 50-424/83-16 and 50-425/83-16.

We do not concur with the finding identified concerning "Failure to Establish Adequate Radiography Procedure" nor do we concur with the interpretation of the ASME Boiler and Pressure Vessel Code, 1974 Edition through Summer 1975 Addenda (hereafter referred to as the "Code"), that the "area of interest", used primarily in Section V of the Code to define the area of a weld to be shown on a radiograph, includes the heat-affected zone. Our position is based on the following information gathered by our code experts:

1. Paragraph T-282 of Article 2, Section V of the Code gives the minimum requirements for radiography procedures. This paragraph contains no requirement for including the heat affected zone in the area of interest and specifying this in the procedure.

2. Paragraph UW-52 of Section VIII of the Code requires spot radiography of vessels that have butt-welded joints that are not radiographed for their full length. UW-52 refers to paragraph UW-51 for standards relative to radiographic examination of welded joints. Neither of these paragraphs require that the heat-affected zone be included in the area of interest. Paragraph UW-51 requires that radiographs be examined in accordance with Article 2, Section V of the Code.

3. Appendix A of Section V of the Code gives the following definition: Area of Interest - This includes the portion of the object in the radiograph that is to be interpreted.

4. Code Interpretation III-82-27 dated April 9, 1983, File No. NI-81-162, reads:

   Question: What extent of area beyond the weld metal is to be included in the radiographing of welds when required by NX-5200?

   Reply: The radiography requirement of NX-5200 is limited to the width of the weld metal.

   Since Section III of the Code is more restrictive than Sections V or VIII, we hold that this interpretation is also valid for those sections.

The Georgia Power Company is committed to meeting the requirements of the Code in the examination of welds at Plant Vogtle. While we do not deny that the heat-affected zone of a weld should be given proper consideration during welding inspections, it remains our position that failure to designate the heat-affected zone as part of the area of interest in radiography procedures and failure to maintain film density of 3.80 over the entire heat-affected zone of a radiograph do not constitute violations of the Code. To further clarify this question, we have addressed the following question to the ASME Boiler and Pressure Vessel Code Committee for Section VIII:

> To what extent of area beyond weld metal is to be included in the radiographing of welds when required by UW-52 and required for interpretation by UW-51(b), (1) through (4)?

When a reply to this question is received, the Georgia Power Company will take whatever actions that are necessary to ensure we are in full compliance with Code requirements.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

REF/DOF/cc

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D. C. 20555

xc:

R. J. Kelly	D. E. Dutton	J. A. Bailey	L. T. Gucwa
R. E. Conway	W. F. Sanders	O. Batum	M. Malcom
G. F. Head	R. H. Pinson	H. H. Gregory III	G. Bockhold
J. T. Beckham, Jr.	B. M. Guthrie	~~R.~~ E. Folker	P. D. Rice
D. N. MacLemore	R. A. Thomas	E. D. Groover	J. L. Vota

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

DEC 22 1983

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-424/83-16 AND 50-425/83-16

Thank you for your response of September 30, 1983, to our Notice of Violation issued on August 31, 1983, concerning activities conducted under NRC Construction Permit Nos. CPPR-108 and CPPR-109. We have evaluated your response and found that it meets the requirements of 10 CFR 2.201. We have elected to withdraw the violation and correct our records accordingly, but not for the reasons cited in your response.

The violation is withdrawn. We have subsequently learned that the area in question has been removed to allow for the installation of a penetration assembly, therefore, there is no longer a safety concern about the uninspected heat affected zone (HAZ). This coupled with the fact that it took a particular repair sequence to cause the thickness conditions which resulted in the unacceptable radiograph, gives us reasonable assurance that the finding was an isolated instance.

About your response - Interpretations of the code by "Code Experts" make your response appear to set aside engineering reason when you consider that, based on failure analysis experience, the technical world realizes that the heat affected zone of a weld is the most critical area of the weldment. After you have received a response to your question regarding the extent of radiographic area coverage of welds from the ASME Boiler and Pressure Vessel Code Committee for Section VIII, we would appreciate receiving a letter giving your reassessment of your technical position.

We appreciate your cooperation in this matter.

Sincerely,

R. C. Lewis

Richard C. Lewis, Director
Division of Project and
Resident Programs

cc: H. H. Gregory, III, Construction
Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Vice President
and General Manager
G. Bockhold, Jr., Plant Manager

Attachment 11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

JAN 11 1983

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-424/82-29 AND 50-425/82-29

This refers to the routine safety inspection conducted by Mr. W. F. Sanders of this office on November 15 - December 13, 1982, of activities authorized by NRC Construction Permit Nos. CPPR-108 and CPPR-109 for the A. W. Votgle facility. Our preliminary findings were discussed with Mr. H. H. Gregory, III, Construction Manager, at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to violate NRC requirements. This item and references to pertinent requirements are listed in the Notice of Violation enclosed herewith as Appendix A. Elements to be included in your response are delineated in Appendix A.

One new unresolved item is identified in the enclosed inspection report. This item will be examined during subsequent inspections.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC's Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

The responses directed by this letter and the enclosures are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

R. C. Lewis

R. C. Lewis, Director
Division of Project and
Resident Programs

Enclosures:
1. Appendix A, Notice of Violation
2. Inspection Report Nos. 50-424/82-29 and 50-425/82-29

cc w/encl:
H. H. Gregory, III, Construction Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Project General Manager
M. Manry, Plant Manager

APPENDIX A

NOTICE OF VIOLATION

Georgia Power Company
A. W. Vogtle 1 and 2

Docket Nos. 50-424 & 425
License Nos. CPPR-108 & 109

As a result of the inspection conducted on November 15 - December 13, 1982, and in accordance with the NRC Enforcement Policy, 47 FR 9987 (March 9, 1982), the following violation was identified.

> 10 CFR 50, Appendix B, Criterion V and the accepted QA program (PSAR Section 17), requires in part:, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures. Procedures shall include appropriate quantitative or qualitative acceptance criteria.
>
> Contrary to the above, activities affecting quality were not prescribed by documented procedures in all aspects, and were not accomplished in accordance with the existing procedures in that on December 1, 1982, twenty-seven electrical equipment cabinets were found stored in place on the "B" level of the control building without the required protection for damage from dust, dirt, moisture, vandalism and rodents.
>
> This is a Severity Level IV Violation (Supplement II).

Pursuant to the provisions of 10 CFR 2.201, you are hereby required to submit to this office within thirty days of the date of this Notice, a written statement or explanation in reply, including: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Date: JAN 11 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report Nos. 50-424/82-29 and 50-425/82-29

Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Facility Name: A. W. Vogtle Nuclear Plant Units 1 and 2

Docket Nos. 50-424 and 50-425

License Nos. CPPR-108 and CPPR-109

Inspection at Vogtle nuclear plant site near Waynesboro, Georgia

Inspector: John F. Rogge, Jr. W. F. Sanders	10 Jan 83 Date Signed
Approved by: V. L. Brownlee, Chief, Section 2B, Division of Projects and Resident Programs	1/11/83 Date Signed

SUMMARY

Inspection on November 15 - December 13, 1982

Areas Inspected

This routine inspection involved 136 resident inspector-hours on site in the areas of primary containment welding, storage of electrical equipment, concrete placement, heavy lifting equipment, and general observation of construction activities in progress.

Results

Of the five areas inspected, no violations or deviations were identified in four areas; one violation was found in one area (electrical equipment storage - paragraph 5.a.)

DETAILS

1. Persons Contacted

   Licensee Employees

   *D. O. Foster, Vice President and General Manager
   *W. T. Nickerson, Manager, General Plant Construction, Nuclear
   *H. H. Gregory, III, Construction Project Manager
   *M. H. Googe, Assistant Construction Project Manager
   *E. D. Groover, QA Site Supervisor
   *R. W. McManus, Manager of Quality Control
   *J. E. Sanders, Project Section Supervisor
   *H. W. Swain, Mechical Quality Control Section Supervisor

   Other licensee employees contacted included construction craftsmen, technicians and office personnel.

   Other Organizations

   *J. S. McLachlan, Bechtel Resident Engineer
   *W. C. Uhouse, Bechtel Engineer "N" Stamp Engineer
   *L. Fields, Chicago Bridge and Iron, Site QA Supervisor

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on December 13, 1982, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings listed below.

   (Open) Violation 424, 425/82-29-01 "Failure to control the Storage and preservation of Electrical equipment." Paragraph 5.a.

   (Open) Unresolved Item 424, 425/82-29-02 "Ambient conditions for welding." Paragraph 5.b.

3. Licensee Action on Previous Inspection Findings

   Not inspected.

4. Unresolved Items

   Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. A new unresolved item was identified during this inspection and is discussed in paragraph 5.b.

5. Independent Inspection Effort

Periodic inspections were made throughout this reporting period in the form of general type inspections in different areas of both facilities. The areas were selected on the basis of the scheduled activities and were varied to provide wide coverage. Observations were made of activities in progress to note defective items or items of noncompliance with the required codes and regulatory requirements. On these inspections, particular note was made of the presence of quality control inspectors, supervisors and quality control evidence in the form of available process sheets, drawings, material identification, material protection, performance of tests and housekeeping.

Interviews were made with craft personnel, supervisors, coordinators, quality control inspectors, and others as they were available in the work areas. Observations were made in the following areas: primary containment structures units 1 and 2, auxiliary building, control building, fuel handling building, fabrication of containment dome, storage of electrical equipment and heavy lift equipment. Within these areas inspected one violations was identified in the storage of electrical equipment and one unresolved item was identified in the #2 containment dome welding.

a) On December 1, 1982 an inspection was made of the electrical control cabinets "stored in place" for compliance to the requirements of Vogtle Field Project Manual procedure, GD-T-09 Rev. 3, and ANSI 45.2.2 The first items inspected were in room 71 of the control building "B" level, which contained (8) cabinets of Rod Control Equipment.

1-1606-U3-RPS-001-Rod Power Supply Control Cabinets

1-1606-S6-002-Reactor Trip Switchgear Breakers

1-1606-U3-FLR-001-Power Cabinets

1-1606-U3-FLR-002-Power Cabinets

1-1606-U3-FLR-004-Power Cabinets

1-1606-U3-FLR-005-Power Cabinets

1-1606-U3-FLR-006-Power Cabinets

1-1606-U3-FLR-007-Full Length Logic Cabinet

The inspector observed the following conditions on the side and bottoms of the cabinets: The plastic coverings are loose, torn and/or usecured at the bottoms, providing minimal protection for dust, moisture, and entry of rodents. An inspection was also made of the tops of the Rod Power Supply Control Cabinets and the Reactor Trip Switch Gear Breaker Cabinets where it was noted that the cabinets were being used as

platforms to work on electrical cable trays and supports. The detrimental conditions noted here consisted of worn and torn plastic covering which exposed the cabinets and the top ventilating grills to the entry of dirt, moisture, and foreign material. A layer of dust, accumulation of trash and an open can of paint primer was observed. Immediate action was taken by the licensee to remove the primer.

Additional inspections were made in other areas on the same level with similar deficient conditions apparent, however, none were observed to be as severe as the conditions in Room 71. Other equipment inspected included:

Rm 76-1-1805-53-1305-480V Switch Gear.

Rm 76-1-1805-53-1304-480v Switch Gear.

Rm 76-1-1805-53-1305-480V Motor Control Center.

Rm 68-1-1805-53-NBS-480V Motor Control Center.

Rm 50-1-1805-53-BL1-480V Switch Gear.

Rm 50-1-1805-53-NBR-480V Motor Control Center.

Rm 50-1-1805-53-1308-480V Switch Gear.

Rm 53-1-1807-Y3-12-Essential AC Inverter.

Rm 53-1-1806-Q3-DP2-125VDC Distribution Panel

Rm 53-1-1807-Q3-VN1-120VAC Essential Panel

Rm 53-1-1807-Q3-VN2-120VAC Essential Panel

Rm 53-1-1807-Q3-RNI-120VAC Reg Instrument Panel

Rm 53-1-1807-Q3-SNI-120VAC Reg Instrument Panel

Rm 53-1-1806-Q3-DP5-125VDC Distribution Panel

Rm 52-1-1806-53-DCA-125VDC Motor Control Center

Rm 52-1-1806-53-DSA-125VDC Switch gear

Rm 52-1-1807-Y3-IAI Vital AC inverter

Rm 48-1-1806-53-DSD 125VDC Switchgear

Rm 48-1-1807-Y3-1D4 Vital AC inverter

Rm 47-1-1806-53-DCB-125VDC Motor Control Center

Rm 47-1-1807-Y31-B12-Vital AC inverter

Rm 61-1-1805-53-B07-480VAC Switchgear

Upon the completion of the inspection, immediate actions were taken by the licensee in the form of removing the old coverings, opening the cabinets, removal of any dirt inside by the use of brushing and vacuuming, and all openings were covered with screen to prevent the possible entry of rodents. The inspector was informed that additional measures to preclude recurrance were bring developed in the areas of inspection criteria, inspection frequency and appropriate revisions to the Project Engineering Instructions and Field Project Manual. These items will be inspected after the program is in final form. This item is a violation (50-424/82-29-01 and 50-425/82-29-01).

b) On November 18, the inspector observed welding being performed on the No. 2 primary containment dome sections after a very light misting rain had commenced. Although the rain was apparent and could be felt with the open hand it was not enough at this time to completely cover a flat surface, however, if this condition were to continue it would have become detrimental to the quality of the weld. After discussing the condition with the welding supervisor and the site QA supervisor, work was stopped for the day although they were of the opinion that this condition was acceptable for welding and that this position could be substantiated by the "General Welding Procedure Specification for the Shielded Metal Arc Process" CBI-CWPS-SMAW(WPS-800) Rev. 10, paragraph 11.1 which states in part: "Welding shall not be performed when the surfaces of the pieces to be joined are wet from rain, snow, or ice, in the welding area." Further discussions with the Licensee, Architect Engineer and contractor have emphasized the need for clarification of the requirements to control the ambient conditions necessary for quailty welds in these structures. This item is unresolved pending clarification by the licensee (50-424/82-27-02 and 50-425/82-29-02).

c) An inspection of the activities relative to the load testing of the Transi-lift Series II (Lampson Crane) and the load testing for the placement of the primary containment dome and NSSS equipment. The equipment tested consisted of a Lampson transi-lift mobile lift crane series II equiped with 340 feet of boom, 190 feet of mast, 120 feet of stinger with 2160 kips of counter weights, the main falls (265'), load block, the reactor load links and the steam generator links.

The planning and requirements are detailed in the Bechtel Procedure HRP-42, Rev. 1, which incorporated the requirements of ANSI N-45.2.2 and Regulatory Guide 1.38. The inspector noted that the required test load for the steam generator was based on 1.1 x the lift load of 834.7

kips on a test load of 918.2 kips. An additional test load was applied for the Primary Containment Dome with 1.1 x lift load of 1061.7 kips which equaled 1168.0 kips.

No items of noncompliance were identified.

6. Concrete Placement

An inspection was made of the activities related to the concrete placement No. A-113-002 in the control building at the 260' EL level 3. Observations were made to ensure the slump tests and air tests are within the specification allowances. A review was made of the prepour sign off documentation on the pour card. Observations were made of the concrete drop, and vibrating.

The following documents were used for the inspections and the acceptance criteria.

a. Specification X2AP01

b. Field Procedure CD-T-02, Concrete Quality Control

c. PSAR Section 3 and 17

No items of noncompliance were identified.

7. Construction Quality Assurance Case Studies

On December 6, 1982, a special study of nuclear Quality Assurance was conducted by the Division of Engineering and Quality Assurance IE:HQD and was performed by a six member team over a 5 day period. The study was concluded and summarized to the licensee by a description of the findings in the following 4 categories:

a. Root Causes - Reviews for the presence of underlying indicators of problems similar to those sites which have experienced severe quality problems.

b. Observation of good practices which were apparent.

c. Prior to Operating License - QA/QC planned programs and actions.

d. Potential improvements - Observation of areas where reviews should be made for improvement.

No items of noncompliance were identified.

Transamerica
Delaval

Transamerica Delaval Inc.
Engine and Compressor Divisio
550 85th Avenue
P.O. Box 2161
Oakland California 94621
(415) 577-7400

May 13, 1982

Director, Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

In accordance with the requirements of Title 10, Chapter 1, Code of Federal Regulations, Part 21, Transamerica Delaval, Inc., hereby notifies the Commision of a potential defect in a component of a DSRV or DSR Standby Diesel Generator. There exists a potential problem with the starting air valve assembly which could result in engine non-availability.

Transamerica Delaval has supplied the DSR and DSRV engines, with the potential defect, to the following sites:

Long Island Lighting	Shoreham	74010/12
Middle South Energy	Grand Gulf	74033/36
Gulf States Utilities	River Bend	74039/40
Carolina Power & Light	Shearon Harris	74046/49
Duke Power Company	Catawba	75017/20
Southern California Edison	San Onofre	75041/42
Cleveland Electric Illum.	Perry	75051/54
Tennessee Valley Authority	Bellefonte	75080/83
Washington Public Power	Unit 1	75084/85
Texas Utilities Services	Comanche Peak	76001/04
Georgia Power	Vogtle	76021/24
Washington Public Power	Unit 4	76031/32
Consumers Power	Midland	77001/04
Tennessee Valley Authority	Hartsville/Phipps Bend	77024/35

Only one of these units has been placed in commercial operation - Southern California Edison, San Onofre. This station has been operating in excess of five years without any failure of the starting air valve assembly reported.

The starting air valve assembly was manufactured and installed in the cylinder head by Transamerica Delaval.

The potential defect is related to the length of the capscrew which holds the starting air valve assembly in the cylinder head. If this capscrew bottoms in the tapped hole in the cylinder head before the assembly is properly seated, the torque wrench reading would be misleading and the assembly could fail.

Transamerica
Delaval

Transamerica Delaval Inc
Engine and Compressor Divisio
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(415) 577-7400

Our review of this potential defect was completed on May 11, 1982.

The starting air valve assembly has been reviewed by Engineering. If all dimension tolerance is stacked in one direction, the installed clearance is less than desirable. A capscrew, 1/4" shorter still gives adequate thread engagement and eliminates any possibility that the capscrew will bottom in the tapped hole. We have changed the capscrew to 2-3/4" length.

It is impossible to determine if this problem exists in an installed assembly. The capscrews which are installed are 3" long. This potential problem can be eliminated by removing the capscrews and cutting 1/4" off and re-installing or by replacing the existing capscrews with a new 2-3/4" long capscrew.

A copy of this letter will be sent to each of the cognizant parties as listed in paragraph 2, no later than June 1, 1982. Transamerica Delaval will furnish parts and technical services as required by request and in accordance with each individual contract.

Since the correction of this potential defect depends on action by others, we can not estimate when the action will be completed.

Very truly yours,

REBoyer

R. E. Boyer, Manager
Quality Assurance

REB:cjb

M1

Georgia Power Company
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Telephone 404 522-6060

Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202
Telephone 205 870-6011

Vogtle Project

August 6, 1982

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M29
Log: GN-186

Attention: Mr. James P. O'Reilly

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424, 50-425; Starting Air Valves, Diesel Generators

Gentlemen:

On July 7, 1982, Georgia Power Company reported a potential deficiency concerning the starting air valve assembly on the standby diesel generators to Mr. Virgil Brownlee. Georgia Power Company has concluded this condition is reportable as a significant deficiency and a substantial safety hazard. Enclosed is our report for this item.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Very truly yours,

D. O. Foster

D. O. Foster
Project General Manager

DOF/CWH/tlp

Enclosure

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D.C. 20555

R. J. Kelly	J. A. Bailey	R. W. Staffa
R. E. Conway	O. Batum	J. L. Vota
G. F. Head	H. H. Gregory, III	W. F. Sanders
J. T. Beckham, Jr.	E. D. Groover	
J. H. Boykin	L. T. Gucwa	
D. E. Dutton	M. Malcom	
R. H. Pinson	C. R. Miles, Jr.	
D. L. McCrary	M. Manry	
R. A. Thomas	P. D. Rice	

Evaluation for a Significant Deficiency

Evaluation for a Substantial Safety Hazard

Introduction: On July 7, 1982, Mr. C. W. Hayes, Vogtle Project QA Manager, informed Mr. Virgil Brownlee on the NRC of a potential deficiency concerning DeLaval diesel generators.

Background: On May 13, 1982, Transamerica DeLaval notified the NRC of a defect in components of DSRV or DSR diesel generators. A potential problem exists with the starting air valve assembly that could result in engine non-availability. Plant Vogtle was identified as a site where this defect could exist.

The potential defect is related to the length of the capscrew which holds the starting air valve assembly in the cylinder head. If the capscrew "bottoms" in the tapped hole in the cylinder head before the assembly is properly seated, the torque wrench reading would be misleading and the assembly could fail.

The starting air valve assembly was manufactured and installed in the cylinder head by Transamerica DeLaval. Field inspection at the Vogtle site has confirmed the use of deficient capscrews.

Analysis of Safety Implications: The standby diesel generator provides an emergency source of onsite power to safety-related equipment to ensure its' continued operation following an accident occurring coincident with a loss of offsite power. Because the design and fabrication of the diesel generators for each unit is the same, a common made failure could be postulated in the starting air valve assembly of both engines due to improperly sized capscrews. The result of the common made failure is a loss of power to both trains of the emergency core cooling system (ECCS) and most of the emergency safety features (ESF) equipment.

Conclusions: This condition represents a deficiency found in design and construction, which, were it to have remained uncorrected could have affected adversely the safety of operations of the nuclear power plant and any time throughout the expected lifetime of the plant.

Due to the problems presented by the potential failure of diesel generators, this condition also represents a significant deficiency in the final design such that the design of the deisel generators does not conform to the criteria and bases stated in the safety analysis report.

Additionally, this condition has been reported to the NRC as a Part 10CFR21 by Transamerica DeLaval since it could cause a loss of redundancy and required safety functions may not be able to be performed.

Corrective Action: In their report to the NRC, Transamerica DeLaval describes two possible corrective actions. These are:

(1) Remove the existing capscrews; cut ¼" off the screw, reinstall the shortened screw.

(2) Remove the existing capscrew and replace with 2-3/4" capscrew available through Transamerica DeLaval.

The use of shortened screws provides adequate thread engagement while eliminating the possibility of bottoming in the tapped hole.

Corrective action will be verified at the Vogtle site by E. D. Groover and will be completed by 11/30/82.

Conclusion: This condition represents a reportable significant deficiency and a Part 10CFR21. Because Transamerica DeLaval has previously reported this condition to the NRC as a Part 10CFR21,

Georgia Power Company is reporting this condition as a Part 10CFR50.55(e) (significant deficiency.)

Transamerica Delaval

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(415) 577-7400

October 28, 1982

Director, Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

In accordance with the requirements of Title 10, Chapter 1, Code of Federal Regulations, Part 21, Transamerica Delaval, Inc. hereby notifies the Commission of a potential defect in a component of the DSRV or DSR Standby Diesel Generator. There exists a potential problem with the engine piston skirt casting which could result in engine non-availability.

Transamerica Delaval has supplied the DSR and DSRV engines to the following utilities:

Duke Power Company	Catawba		S/N	75017
			S/N	75018
			S/N	75019
			S/N	75020
Tennessee Valley Authority	Stride	S/N	77026	77031
		S/N	77027	77032
		S/N	77028	77033
		S/N	77029	77034
		S/N	77030	77035
Texas Utilities	Commanche Peak		S/N	76003
Gulf States Utilities	River Bend		S/N	74039
			S/N	74040
Carolina Power & Light	Shearon Harris		S/N	74046
			S/N	74047
Georgia Power Co.	Vogtle		S/N	76021
			S/N	76022
			S/N	76023
			S/N	76024

At present none of these units have been placed in commercial operation.

The Piston Skirts were manufactured and installed in the engine by Transamerica Delaval.

Transamerica
Delaval

The potential problem is the possibility of residual stress caused by the method of heat treating of Piston Skirts used between December 1978 and October 1981.

This residual stress in combination with operating stress could cause cracking of the Piston Skirt during operation which could result in engine failure if undetected.

We recommend that the Pistons be removed from the engines and returned to Transamerica Delaval's plant in Oakland, Calif. Transamerica Delaval will inspect, stress relieve, and return the Pistons for reinstallation. Any Piston Skirts which are found defective will be replaced.

A copy of this letter will be sent to each of the cognizant parties listed in paragraph 2 no later than November 18, 1982. Transamerica Delaval has in stock sufficient Piston Skirts and related parts to handle this change. The engine Instruction Manual contains instructions on how to replace Pistons, and each site has a copy of the Instruction Manual. Transamerica Delaval will furnish related parts and technical services as required upon request and in accordance with each individual contract.

All of the sites involved are in various states of construction, so we cannot estimate when the change out will be completed. Removal of Pistons requires the ability to bar the engine over. This can only be done after the engine is installed on its foundation and the flywheel has been installed.

Our evaluation of this matter was concluded on October 27, 1982.

Sincerely,

R E Boyer

R. E. Boyer, Manager
Quality Assurance

REB:hw

MLF

Georgia Power Company
Post Office Box 4545
Atlanta, Georgia 30302
Telephone 404 526-6526

Vogtle Project

March 1, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M36
Log: GN-219

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424; 50-425; Delaval Diesel Generator -
Piston Skirt Cracking

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has concluded its evaluation concerning the above referenced subject and has determined a significant deficiency and a substantial safety hazard could exist. However, since Transamerica Delaval has already reported this defect to the NRC in their letter of October 28, 1982, Georgia Power Company is reporting this concern as a significant deficiency.

Enclosed is our evaluation concerning the piston skirts.

This report contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

E. J. Turner
for D. O. Foster

CWH/DOF/tlp
Enclosure

xc: U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

R. J. Kelly
R. E. Conway
G. F. Head
J. T. Beckham, Jr.
J. H. Boykin
D. E. Dutton
R. H. Pinson
D. L. McCrary
R. A. Thomas
J. A. Bailey
O. Batum
H. H. Gregory, III
E. D. Groover
L. T. Gucwa
M. Malcom
M. Manry
P. D. Rice
J. L. Vota
W. F. Sanders

EVALUATION FOR A SUBSTANTIAL SAFETY HAZARD
EVALUATION FOR A SIGNIFICANT DEFICIENCY

Transamerica Delaval Diesel Generators
Piston Skirt Cracking

Initial Report:

On December 23, 1982, Mr. E. D. Groover, Georgia Power Company, QA site supervisor reported a potential deficiency to Mr. Virgil Brownlee of the USNRC Region II concerning the potential problem of piston skirt cracking in Delaval diesel generators.

Background:

On October 28, 1982, Transamerica Delaval notified the Nuclear Regulatory Commission of a potential defect in components furnished for DSRV or DSR Standby Diesel Generators. The defect concerned engine piston skirt castings which could result in engine nonavailability. The potential defect concerns the possibility of existence of residual stresses in the piston skirts manufactured between December 1978 and October 1981. The residual stresses result from the method of heat treating the piston skirts. This residual stress, in combination with operating stress, could cause cracking of the piston skirt during operation which, could result in engine failure if undetected.

The Vogtle Project architect/engineer was notified of this potential defect by Transamerica Delaval in a letter dated November 18, 1982. Georgia Power Company received notification from the architect/engineer in December 1982.

Analysis of Safety Implications:

The standby diesel generators provide an emergency source of onsite power to the safety-related equipment to ensure its continued operation following an accident occurring coincident with loss of offsite power. Because the design and fabrication of the piston skirts for the diesel generators furnished for the Vogtle Project is essentially the same, it is reasonable to postulate a common mode failure of a piston skirt on both engines. This would cause the failure of both engines, resulting in a loss of power to both trains of the emergency core cooling system (ECCS), and most of the emergency safety features (ESF) equipment. Since the piston skirts furnished for Vogtle can be assumed to contain high residual stresses due to the method of heat treatment, it is reasonable to assume a failure of the emergency onsite power supply and the consequential failure of the ECCS and ESF systems.

Review of Part 10 CFR 50.55(e) and Part 21:

The holder of a construction permit is to notify the Commission of each deficiency found in design and construction which, were it to have

remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant. For this deficiency, failure of the diesel generators results in the unavailability of the ECCS and ESF systems, which could affect safety of operations.

Georgia Power Company has determined that this concern does not represent a significant breakdown in any portion of the quality assurance program at Transamerica Delaval. This concern does represent a significant deviation from performance specifications that will require an extensive repair to establish the adequacy of the diesel generators to perform their intended safety function.

Guidance supplied by the NRC for Part 10 CFR 21 states that "a deviation that causes or could cause a failure in a redundant basic component is a reportable defect under Part 10 CFR 21 since the loss of safety function of a basic component is considered a major reduction in the degree of protection provided to the public health and safety. It is also possible that the defect exists in the redundant basic component which could result in a loss of safety function. The existence of a defective basic component, considering a single failure of its counterpart redundant basic component, could result in a loss of safety function." Based upon this guidance from NUREG-0302, Revision 1, page 21.3(k).2, it has been concluded a defect does exist and that this concern is reportable under Part 10 CFR 21.

Conclusion:

Georgia Power Company has concluded that the improper heat treatment of the piston skirt castings represents a reportable deficiency and substantial safety hazard. Since the NRC has been adequately informed of the existence of the defect by a previous letter from Transamerica Delaval, Georgia Power Company is reporting this concern as a significant deficiency. NRC guidance indicates that duplicate reporting is not required if the Commission has been adequately informed and if the required information is provided should a notification be required.

Corrective Action:

In their report to the Nuclear Regulatory Commission, Transamerica Delaval describes the following corrective action:

> "After the engines have been installed on their foundations and flywheels installed so the engine can be barred over, remove all pistons from the engines and return them to the Transamerica Delaval plant in Oakland, California. Transamerica Delaval will then inspect, stress relieve and return the pistons for reinstallation. Any piston skirts found defective will be replaced."

Because of the magnitude of the effort associated with this corrective action, Georgia Power Company is investigating with Transamerica Delaval, Inc. a number of questions regarding the logistics of the piston replace-

ment. Also, based upon the current construction schedule, the diesel generator building will not be completed until 1985. Since the engines must be installed on their foundations and barred over, Georgia Power Company cannot begin corrective action until 1985, and currently estimates corrective action will be completed by December 1986.

Mr. W. R. Evans, Georgia Power Company's mechanical Project Section Supervisor, is responsible for the corrective action.

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-7720

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

Georgia Power

the southern electric system

D. O. Foster
Vice President and General Manager
Vogtle Project

November 3, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II-Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M49
Log: GN-274

Reference: Vogtle Electric Generating Plant-Units 1 and 2
50-424, 50-425; Transamerica Delaval-Class 1E Cable

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has completed an evaluation of the above referenced subject and, based upon the results of the evaluation, has concluded that it constitutes a significant deficiency pursuant to the requirements of Part 10 CFR 50.55(e). Since Transamerica Delaval reported this problem as a substantial safety hazard in their September 27, 1983, letter to the NRC, Georgia Power Company is not reporting this event under Part 10 CFR 21. Enclosed is a copy of our evaluation and a copy of the letter from Transamerica Delaval to the NRC.

This response contains no proprietary information and may be placed in the NRC public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/REF/cc

enclosure

xc: U. S. Regulatory Commission
Document Control Desk
Washington, D. C. 20555

xc:

R. J. Kelly	R. H. Pinson	E. D. Groover
R. E. Conway	B. M. Guthrie	L. T. Gucwa
G. F. Head	R. A. Thomas	M. Malcom
J. T. Beckham, Jr.	J. A. Bailey	G. Bockhold
D. N. MacLemore	O. Batum	P. D. Rice
D. E. Dutton	H. H. Gregory, III	J. L. Vota
W. F. Sanders	C. W. Hayes	

EVALUATION FOR A SIGNIFICANT DEFICIENCY
TRANSAMERICA DELAVAL-CLASS 1E ELECTRICAL CABLES

Initial Report:

On October 3, 1983, Mr. C. W. Hayes of Georgia Power Company reported a potential deficiency to Mr. J. Rogge of the NRC concerning the qualification of certain class 1E electrical cables on the Transamerica Delaval diesel generators.

Background Information:

On November 4, 1983, Transamerica Delaval reported the failure of some 1E electrical cable to pass the IEEE-383 flame test. The November 4, 1982, letter referenced a Service Information Memo (SIM) No. 361 dated October 21, 1982. At this time, Transamerica Delaval stated that the loss of these cables would not adversely affect the engine's ability to carry out its specified function. Replacement of the following cables with class 1E cable was recommended.

(a) Shielded cable from the terminal block to the Airpax tachometer relay in the engine control panel.

(b) Shielded cable from the Airpax magnetic pickups to the junction boxes installed on the side of the engine.

(c) The multiconductor cable from the engine side mounted junction box to the Woodward governor actuator.

Transamerica Delaval conducted another review in September, 1983, and on September 27, 1983, they reported to the NRC a substantial safety hazard concerning the engine mounted electrical cables which could result in engine performance deterioration. The cables in question are:

(a) Shielded cables which run from the magnetic pick-ups to the engine junction boxes.

(b) The multiconductor cable which runs from the engine terminal box to the Woodward governor actuator.

For item (a), Transamerica Delaval concluded a problem could exist only if the ambient temperature exceeded 129° F.

Engineering Evaluation:

The multiconductor cable to the Woodward governor actuator provides automatic engine control for the proper loading of the diesels. Failure of this cable would result in the governor operating as a hydraulic speed sensing governor. Operation in this manner would allow the engine to run and carry load, but would provide a slightly slower response to load change or load pick-up. Load pick-up may not be achieved within the 9.5 second time limit in the Vogtle Electric Generating Plant (VEGP) FSAR, Section 8.3.1.1.3F.

The multiconductor cable is rated for 75°C (167°F). Evaluation has shown that if the ambient temperature is 98.5°F and if the diesel generator is required to be operated, the heat from the diesel and the ambient temperature will result in the exceeding of the 75°C rating. Section 9.4.7.1.1 of the VEGP FSAR concerning the Diesel Generator Building Ventilation System indicates the maximum ambient temperature of the building will be 120°F with the diesel generator operating. Therefore operation of the diesels would result in the exceeding of the temperature rating.

The standby diesel generators provide an emergency source of onsite power to safety-related equipment to ensure continued operation following an accident occurring coincident with a loss of offsite power. Failure of the multiconductor cable could result in the failure of the diesel generator to pick-up loads properly.

A review was made to determine if a breakdown had occurred in the quality assurance program of Transamerica Delaval. This review concluded that the program is adequate. A full-scope audit was conducted by Bechtel Procurement Supplier Quality Department on September 14-16, 1983.

Other cable reviewed was:

° the shielded cable from the terminal block to the Airpax tachometer relay in the engine control panel.

° The shielded cable on the engine from the Airpax magnetic pickups to the junction boxes.

These shielded cables provide the "ready to load" signal required to automatically close the diesel generator output circuit breaker. This signal also indicates that starting air should be turned off. As analyzed by Delaval, these cables are rated for 80°C (170°F). The expected operating temperature would exceed the manufacturer's rating when the ambient temperature is greater than 129°F. Since the design bases of the diesel generator building will keep the ambient temperature below 120°F, the cables would satisfactorily operate, except when a fire exists. (Note, these cables failed the IEEE-383 flame test.)

Corrective Action:

The above referenced cables will be replaced with 90°C (194°F) IEEE-qualified cable. These cables are fully qualified and the 90°C rating will ensure their temperature rating will not be exceeded. Georgia Power Company has ordered replacement cable. These cables will be replaced when the diesel generator is disassembled for the heat-treatment of the piston skirts. Mr. W. R. Evans of Georgia Power Company will be responsible for this corrective action.

Conclusion:

A significant deficiency exists since a failure of the diesels to perform their designed function is possible under certain conditions, which could occur during the lifetime of the units . This is representative of a deficiency in the final design as approved and released for construction such that the design does not conform to the bases stated in the VEGP FSAR.

Transamerica Delaval

Transamerica Delaval Inc.
Engine and Compressor Division
550 85th Avenue
P.O. Box 2161
Oakland, California 94621
(415) 577-7400

September 27, 1983

Director, Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir,

In accordance with the requirement of Title 10, Chapter 1, Code of Federal Regulations, Part 21, Transamerica Delaval, Inc., hereby notifies the Commission of a potential defect in a component of a DSRV or DSR Standby Diesel Generator. There exists a potential problem with Engine Mounted Electrical Cables which could result in engine performance deterioration.

Transamerica Delaval has supplied the DSR and DSRV Engines with the potential problem to the following sites:

Utility	Site	Serial No.	Model
Long Island Lighting	Shoreham	74010/12	DSR 48
Middle South Energy	Grandgulf	74033/36	DSRV 16
Duke Power Company	Catawba	75017/20	DSRV 16
Southern California Edison	San Onofre	75041/42	DSRV 20
TVA	Bellefonte	75080/83	DSRV 16
Texas Utilities Services	Comanche Peak	76001/04	DSRV 16
Georgia Power	Vogtle	76021/24	DSRV 16
Consumers Power	Midland	77001/04	DSRV 12
TVA	Hartsville/Phipps Bend	77024/35	DSRV 16

These Electrical Cables are installed by Transamerica Delaval.

The potential problem with the Electrical Cables is that the manufacturers temperature rating for the cable insulation may be exceeded during operation of the Diesel Generator.

There are two Cables in question. One is the Shielded Cables which run from the magnetic pickups to the engine junction boxes. These Cables are rated for 80°C. The maximum expected operation temperature depends on ambient temperature. Failure of these Cables could prevent the closing of the output breaker depending on how this signal is used in the plant.

Transamerica Delaval

The second cable is the Multi-conductor Cable which runs from the on engine terminal box to the Woodward Governor Actuator. This Cable is rated for 75°C. The operating temperature of this Cable also depends on ambient temperature. Failure of this Cable would result in the Governor operating as a Hydraulic Speed Sensing Governor. Operation in this manner would allow the Engine to run and carry load, but would provide a slightly slower response to load change or load pick-up.

In the case of the Governor Cable, if the ambient temperature exceeds 98.5°F., the rated insulation temperature would be exceeded. This would be the situation for all sites listed in paragraph two.

In the case of the Magnetic Pickup Cable, the expected operating temperature of the Cable would exceed the manufacturers rating when the ambient temperature is greater than 129°F. Long Island Lighting has an ambient temperature of 130°F. The other sites listed in paragraph 2 have an ambient temperature lower than 125°F.

Our review of this potential defect was concluded on September 26, 1983.

A copy of this letter will be sent to each of the cognizant parties as listed in paragraph two, no later than October 14, 1983. Transamerica Delaval will furnish parts and technical services as required, by request, and in accordance with each individual contract.

We recommend that these Cables be replaced with 90°C rated cable. We have available cable and instruction for this change out.

Since the correction of this potential defect depends on action by others, we cannot estimate when the action will be complete.

Very Truly Yours,

REBoyer

R. E. Boyer, Manager
Quality Assurance

REB:hw

9.4.7 DIESEL GENERATOR BUILDING VENTILATION SYSTEM

The diesel generator building heating, ventilation, and air-conditioning (HVAC) system functions to remove heat from the building during diesel generator operation and to supply sufficient heat, when diesels are not operating, to allow easy starting of the diesel generators and to allow personnel occupancy. The system is divided into two subsystems, one engineered safety features (ESF) and one non-ESF. A separate HVAC system is provided for each of the two diesel generator building trains.

9.4.7.1 Design Bases

9.4.7.1.1 Safety Design Bases

A. The system is designed on the basis of limiting the maximum temperature of the building to 120°F with the diesel generator operating. The building is ventilated with 100-percent outside air at summer design temperatures and employs recirculation and outside ventilation air as the outside air temperatures drop toward winter temperatures.

B. The safety functions of the diesel building ventilation system can be performed, assuming a single active component failure coincident with the loss of offsite power.

C. The diesel generator building ventilation system is connected to the 480-V ac Class 1E bus of the same train as the diesel generator set being ventilated.

D. The diesel generator building ventilation system is protected from the effects of natural phenomena, such as earthquakes, hurricanes, floods, and tornado missiles.

9.4.7.1.2 Power Generation Design Bases

A. The non-ESF heating system maintains the building temperature at 50°F minimum when the diesel generators are not running.

B. The non-ESF subsystem ventilates the building as required to allow for maintenance and personnel access.

9.4.7.1.3 Codes and Standards

The diesel generator building HVAC system is designed to conform to applicable codes and standards listed in table 3.2.2-1.

9.4.7.2 System Description

9.4.7.2.1 General Description

Figure 9.4.7-1 shows the diesel generator ventilation system. Component data are provided in table 9.4.7-1. Figure 9.4.7-2 is a flow diagram.

The non-ESF subsystem consists of 10 unit heaters, one non-ESF normal exhaust fan with a motor-operated backdraft damper to prevent backflow, and associated control for each diesel generator room. The ESF subsystem consists of two 50-percent-capacity supply fans. Each diesel generator room is provided with two air inlets/outlets located on the second floor and discharge/intake openings located in the penthouse. Automatic dampers are provided in each opening.

An exhaust fan is provided for continuously venting the fuel oil day tank room.

9.4.7.2.2 Component Description

A. Non-ESF Normal Exhaust Fans

The non-ESF exhaust fans are propeller type and are V-belt driven by electric motors. These fans are located on the floor of the diesel generator building penthouse and installed in series with a motor-operated damper.

B. Non-ESF Exhaust Fans

The non-ESF fuel oil day tank room exhaust fan is a centrifugal type, directly driven by electric motor.

C. Unit Heaters

Each unit heater consists of a resistance heater and a fan, both in one metal housing.

D. ESF Supply Fans

The two 50-percent-capacity ESF exhaust fans for each diesel generator room are heavy duty, vaneaxial type, directly driven by electric motors. These fans are located in the penthouse of the diesel generator building and are provided with backdraft dampers.

9.4.7.2.3 System Operation

During normal plant power operation, the diesel generator room is ventilated by the non-ESF fan exhausting to the atmosphere at el 266 ft 0 in. Supply air for the normal ventilation is drawn in through motor-operated intake dampers in the wall openings located on the first floor (el 224 ft 0 in.). The exhaust fan motors are started automatically by separate room thermostats whenever the temperature in the building exceeds thermostat settings of 90°F.

Each of the 10 unit heaters operates on a separate thermostatic control to maintain a 50°F minimum still-air temperature by heating and recirculating the air in the room. These heaters operate automatically and independently from the ventilation system. The fuel oil day tank room exhaust fan exhausts air to the atmosphere at el 241 ft 0 in.

Manual switchover to emergency operations is accomplished from the main control room. The ESF fans are manually activated after a high-temperature alarm is initiated in the control panel. During emergency operation, the airflows are reversed, and the air is drawn in through the openings at level 2 (el 266 ft 0 in.) and exhausted through the openings at el 224 ft 0 in.

9.4.7.3 <u>Safety Evaluation</u>

A. The ESF supply fans are sized to supply sufficient outside air to hold the maximum temperature in the building to the required limits.

B. Two 50-percent-capacity fans are provided to ensure that a single failure cannot cause complete loss of the safety function, as indicated in table 9.4.7-2.

C. The ESF supply fans are connected to the safety bus of the same train as the diesel generator in that room. Thus, a failure of one emergency power train cannot cause loss of function of the redundant generator and power train.

D. The safety-related portions of the diesel generator building ventilation system are located in the diesel building, which is designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, tornado missiles, and other appropriate natural phenomena. The ESF fans are designed and constructed as Seismic Category 1 to ensure that they will function during and after a safe shutdown earthquake (SSE). Sections 3.3, 3.4, 3.5, 3.7, and 3.8 provide the bases for the adequacy of the structural design of the diesel generator building.

9.4.7.4 Tests and Inspections

The system is designed to permit periodic inspection; it is tested for function and capability in the preoperational testing. Fans are tested in accordance with Air Moving and Conditioning Association (AMCA) Standard 210.[1]

9.4.7.5 Instrumentation Applications

Unit heaters are controlled by thermostats. Air flowing through the ESF supply fans is monitored by temperature and flow instrumentation. These fans are operable from both the control room and remote shutdown panel.

The following instrumentation for the diesel generator building ventilation system is provided in the control room:

- Alarm for high temperature in the diesel generator building.
- Alarm for low temperature in the diesel generator building.
- Alarm for low airflow.
- Position indication of intake and discharge louvers.
- Indication of the operational status of fans.

Attachment 15

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

FEB 22 1983

X7B610

Orig - RMS
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CWH
EST

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-424/83-04 AND 50-425/83-04

This refers to the routine safety inspection conducted by Mr. J. R. Harris of this office on January 11-14, 1983, of activities authorized by NRC Construction Permit Nos. CPPR-108 and CPPR-109 for the Vogtle facility. Our preliminary findings were discussed with Mr. H. H. Gregory, Construction Project Manager, at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to violate NRC requirements. This item and references to pertinent requirements are listed in the Notice of Violation enclosed herewith as Appendix A. Elements to be included in your response are delineated in Appendix A.

We have examined actions you have taken with regard to previously identified enforcement matters. These are discussed in the enclosed inspection report.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC's Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

The responses directed by this letter and the enclosures are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

FEB 22 1983

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

H. C. Dance, Chief
Project Branch 2
Division of Project and
Resident Programs

Enclosures:
1. Appendix A, Notice of Violation
2. Inspection Report Nos. 50-424/83-04
and 50-425/83-04

cc w/encls:
H. H. Gregory, III, Construction
Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Project General Manager
M. Manry, Plant Manager

APPENDIX A

NOTICE OF VIOLATION

Georgia Power Company
Vogtle

Docket Nos. 50-424
License Nos. CPPR-108

As a result of the inspection conducted on January 11-14, 1983, and in accordance with the NRC Enforcement Policy, 47 FR 9987 (March 9, 1982), the following violation was identified.

10 CFR 50, Appendix B, Criterion V, as implemented by Vogtle PSAR section 17.1.5 requires in part that activities affecting quality shall be prescribed by documented instructions and shall be accomplished in accordance with these instructions. Specification X2AP01 requires during cold weather (when ambient temperatures are generally below 40°F) that newly placed concrete shall be protected from freezing by covering and/or heating and that the concrete surface temperature shall be maintained 10°F above the temperature at which the concrete was placed.

Contrary to the above, on the day after placement and with an ambient temperature of 34°F, concrete from pour number A-093-007 was uncovered and the concrete surface temperature was 13 degrees below the placement temperature of 58°F.

This is a Severity Level V Violation (Supplement II).

Pursuant to the provisions of 10 CFR 2.201, you are hereby required to submit to this office within thirty days of the date of this Notice, a written statement or explanation in reply, including: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Date: FEB 22 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report Nos.: 50-424/83-04 and 50-425/83-04

Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Docket Nos.: 50-424 and 50-425

License Nos.: CPPR-108 and CPPR-109

Facility Name: Vogtle

Inspection at Vogtle site near Waynesboro, Georgia

Inspector: M D Hunt for 2/14/83
J. R. Harris Date Signed

Approved by: M D Hunt for 2/14/83
T. E. Conlon, Section Chief Date Signed
Engineering Programs Branch
Division of Engineering and Operational Programs

SUMMARY

Inspection on January 11-14, 1983

Areas Inspected

This routine, unannounced inspection involved twenty-six inspector-hours on site in the areas of previous enforcement matters and structural concrete.

Results

Of the two areas inspected, no violations or deviations were identified in one area; one violation was found in one area (Failure to provide proper cold weather protection to newly placed concrete, Paragraph 6).

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *H. H. Gregory, III, Project Manager, Construction
   *J. O. Dorough, Assistant Construction Project Manager
   *E. D. Groover, QA Site Manager
   *R. W. McManus, Manager Quality Control
   *N. Brooks, Assistant Project Section Supervisor Civil
   *N. L. Blocker, Civil QC Section Supervisor
   *D. Tamplin, Civil Area Engineer

   Other licensee employees contacted included three construction craftsmen, six technicians, two security force members and two office personnel.

   Other Organizations

   *W. G. Uhcuse, Resident Engineer, Bechtel
   *J. W. Duffy, Assistant Project Manager, Walsh Construction Company
   *G. Ryan, Quality Assurance - Quality Control Coordinator, Walsh Construction Company

   NRC Resident Inspector

   *W. F. Sanders

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on January 14, 1983, with those persons indicated in Paragraph 1 above. The licensee acknowledged the inspection findings. The following items were opened:

   a. Violation 424/83-04-01, Improper Concrete Cold Weather Protection
   b. Inspector Followup Item 424-425/83-04-02, Procedure and Specification Inconsistencies in Cold Weather Protection Requirements

3. Licensee Action on Previous Enforcement Matters

   (Closed) Deviation (424, 425/82-26-01), Failure to Perform Separate Test Cycles for Each Cadweld Splicing Crew. The inspector examined the licensee's response dated December 7, 1982, and implementation of the response. Examination of records and observation of ongoing Cadweld operations showed that separate test cycles are now being maintained for each splicing crew. This item is closed.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Independent Inspection (92706)

The inspector examined the following:

a. Soils and concrete laboratory and currentness of calibration of laboratory equipment.
b. Ongoing cadweld operations in the Unit 2 powerblock.
c. Completed concrete structures in Unit 1 and Unit 2 powerblock.

Within the areas examined, no violations or deviations were identified.

6. Containment, Structural Concrete II (47054) - Unit 1 and Unit 2

The inspector observed partial placement of pour numbers A-093-007, A-093-030, A-113-033A, A-113-020 and 1-44A-082-C80A in the Unit 1 and Unit 2 powerblock. Acceptance criteria examined by the inspector appear in the following documents:

a. Specification X2AP01, Forming, Placing, Finishing and Curing Concrete.
b. Procedure CD-T-02, Concrete Quality Control.
c. PSAR Sections 3 and 17.

Forms were tight and clean. Rebar was properly installed and clean. Placement activities pertaining to delivery time, free fall, flow distance, layer thickness, and consolidation conformed to specification requirements. Concrete placement activities were continuously monitored by construction and QC inspectors. Examination of batch tickets showed that the specified design mixes were being delivered. Samples of plastic concrete were obtained from designated sampling points and were tested in accordance with specification requirements. Test results showed that plastic concrete being placed met requirements for slump, air content, and temperature. Post placement inspection disclosed the following NRC inspector identified items.

Specification X2AP01, Revision 14, Forming, Placing, Finishing, and Curing Concrete. Section C3.2 Paragraphs J4 and J5, requires during cold weather (when daily temperature are generally below 40°F) that newly placed concrete be protected from freezing by covering or heating and that the concrete surface temperature be maintained 10°F above the temperature at which the concrete was placed. Observations of post placement controls on pour number A-093-007 on the first day after placement showed that the concrete was not covered and that the concrete surface was 13 degrees less than the placement temperature of 58°F. Failure to provide proper cold weather protection on pour number A-093-007 was identified to the licensee as violation 50-424/83-04-01, Improper Cold Weather Protection.

Review of procedure CD-T-02, Concrete Quality Control and Specification X2AP01, Forming, Placing, Finishing, and Curing Concrete indicated that the cold weather requirements specified in specification X2AP01 are not clearly defined in procedure CD-T-02. Apparent inconsistencies between specification X2AP01 and procedure CD-T-02 was identified to the licensee as Inspector Followup Item number 50-424/83-04-02 and 50-425/83-04-02, Procedure/Specification Inconsistencies in Cold Weather Protection Requirements.

mF

Georgia Power Company
Post Office Box 4545
Atlanta, Georgia 30302
Telephone 404 526-6526

Vogtle Project

March 22, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG10
Log: GN-222

Reference: 50-424/83-04
50-425/83-04

Attention: R. C. Lewis

Gentlemen:

The Georgia Power Company wishes to submit the following information concerning the violation discussed in your inspection report 50-424/83-04 and 50-425/83-04.

We concur with the violation concerning improper concrete cold weather protection.

During the placement of A-093-007 adequate housing was provided for weather protection; however, during the hours following, winds damaged the weather protection and the remaining parts of the housing were removed. Eight to ten hours passed before the responsible Georgia Power Company personnel were informed that the protection had been removed. Immediate action was then taken to protect the concrete. As a result of an inspection conducted January 18 through January 21, 1983 the failure to comply with specification X2AP01 was reported on a deviation report and evaluated by engineering.

To prevent reoccurrence, specification X2AP01 was revised to clarify cold weather requirements and procedure CD-T-02 was revised to incorporate more detailed requirements for cold weather concreting.

Full compliance was achieved February 21, 1983.

U. S. Nuclear Regulatory Commission
Atlanta, Georgia 30303
March 22, 1983
Page Two

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

E. J. Turner
for D. O. Foster

CWH/DOF/dab

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D. C. 20555

R. J. Kelly
R. E. Conway
G. F. Head
J. T. Beckham, Jr.
J. H. Boykin
D. E. Dutton
R. H. Pinson
D. L. McCrary
R. A. Thomas
J. A. Bailey
O. Batum
H. H. Gregory, III
E. D. Groover
L. T. Gucwa
M. Malcom
C. R. Miles, Jr.
M. Manry
R. W. Staffa
J. L. Vota
W. F. Sanders

Attachment 16

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

SEP 28 1983

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Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: SUMMARY OF MEETING - DOCKET NOS. 50-424 AND 50-425, VOGTLE 1 AND 2

This refers to the management meeting conducted, at your request, at the Georgia Power Headquarters Office on August 22, 1983, among members of your staff, myself and others of my staff. This meeting was held to discuss your findings concerning the investigation of subcontractor quality assurance performance at Plant Vogtle. A summary of this meeting and list of attendees are enclosed.

It is our opinion that this meeting was beneficial and has provided a better understanding of the investigation, evaluation and corrective actions being implemented by Georgia Power Company with regard to quality workmanship at the Vogtle site.

In accordance with section 2.790 of NRC's "Rules of Practice", Title 10, Code of Federal Regulations, a copy of this letter will be placed in the NRC's Public Document Room.

Should you have any questions concerning this matter, we will be pleased to discuss them.

Sincerely,

Robert D. Martin for
James P. O'Reilly
Regional Administrator

Enclosures:
1. Summary of Meeting
2. Attendance List

cc w/encls:
H. H. Gregory, III, Construction Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Vice President and General Manager
G. Bockhold, Jr., Plant Manager

ENCLOSURE 1

MEETING SUMMARY

Licensee: Georgia Power Company

Facility: Plant Vogtle Units 1 and 2

Docket Nos: 50-424 and 50-425

SUBJECT: SUBCONTRACTOR QUALITY ASSURANCE PERFORMANCE

The meeting conducted on August 22, 1983, covered the following three subjects with regard to subcontractor quality assurance performance:

1. The measures taken by Georgia Power Company to investigate allegations made by Pullman Power Products quality control personnel were discussed. A special task force was formed by Georgia Power Company to investigate the allegations. This investigation was conducted over a two week period and consisted of personnel interviews, observations of work progress and a review of relevant documentation. Specific technical areas covered were pipe support installation and piping installation.

   Since there was an expressed concern that salary administration practices and personnel policies in the Pullman site organization could have a detrimental effect on the attitude of quality control personnel with regard to the quality of workmanship, the investigation focused on three specific areas:

   - The rationale used in establishing salary increases in early 1983
   - Certain disciplinary actions
   - Transfer or job rotation practices among QC inspectors

   The Georgia Power Company investigation found that, although there are improvements which are still needed, the overall quality control program for the installation of the piping systems at Plant Vogtle, with the changes which have been implemented to date, is effective. The investigation found that the QC inspectors are well qualified and knowledgeable, and found no evidence that Pullman Power Products management had used the salary administration program to intimidate Quality Control Personnel. There was no indication that "short cuts" were being taken nor that the inspectors were being called on by their superiors to overlook problems. The investigation found that the craftsmen performing the work do not have adequate detail knowledge of the specifications and procedures and that considerable confusion existed concerning allowable dimensional tolerances.

The task force recommended that craft training be enhanced, that dimensional tolerances be clarified, that Pullman Power Products consider more formal salary administration and other personnel programs and that other improvements be made to the quality control program.

2. Status of investigation dealing with allegations made by a Walsh Company boilermaker that improper welding and work practices had occurred at the Vogtle site was presented. Specifically, there were 23 concerns expressed by the Walsh employee which dealt with 12 separate items. The investigation began on August 8 and includes interview as well as confirmatory nondestructive examinations. To date, of the 23 items of concern, 15 items have been completed. Georgia Power Company expects to complete the investigation by August 31, 1983.

3. A summary of defects found during the reinspection of Pullman Power Products manufactured piping spool pieces was presented. Georgia Power Company stated that there are an estimated 15,611 spools on site. Over 35,000 welds in these spools have been reinspected of which 966 welds required the resolution of a Non-conformance Report (NCR). Defects requiring an NCR include:

Defect	Percent of Total NCRS
Linear Indication	7
Cracks	1
Undercut	11
Invalid PT (PT over slag)	20
Incomplete Penetration	10
Porosite	3
Incomplete Weld	1
Lack of Fusion	2
Slag	16
Arc Strikes	13
Miscellaneous (Wrong ID rust, melt thru, physical damage)	16

Georgia Power Company has evaluated each NCR and has concluded that none could affect the safety of operation.

ENCLOSURE 2

ATTENDANCE LIST

Georgia Power Company Representatives

R. E. Conway, Senior Vice President, Engineering Construction and Project Management
D. O. Foster, Vice President and General Manager, Vogtle Project
E. E. Dutton, Vice President, Generating Plant Projects
W. E. Ehrensperger, Consultant
P. D. Rice, General Manager, Quality Assurance and Radiological Health and Safety
H. H. Gregory, III, Construction Project Manager, Vogtle
W. T. Nickerson, Manager, Nuclear Generating Plant Construction
C. W. Hayes, Vogtle Quality Assurance Manager
K. D. Handy, Assistant to Senior Vice President
W. D. Drinkard, Section Supervising Engineer

U. S. Nuclear Regulatory Commission

J. P. O'Reilly, Regional Administrator
R. C. Lewis, Director, Division of Project and Resident Programs (DPRP), RII
H. C. Dance, Chief, Project Branch 2, DPRP, RII
A. R. Herdt, Chief, Engineering Program Branch, Division of Engineering and Operational Programs, RII
V. W. Panciera, Chief, Project Section 2B, DPRP, RII
W. F. Sanders, Senior Resident Inspector, DPRP, RII

Attachment 17
SSINS No.: 6820
OMB No.: 3150-0011
IEB 83-06

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 22, 1983

IE BULLETIN NO. 83-06: NONCONFORMING MATERIALS SUPPLIED BY TUBE-LINE CORPORATION FACILITIES AT LONG ISLAND CITY, NEW YORK; HOUSTON, TEXAS; AND CAROL STREAM, ILLINOIS

Addressees:

For action:

All nuclear power reactor facilities or fuel facilities holding an operating license (OL) or construction permit (CP).

Purpose:

Power reactor and nuclear fuel facilities received initial notification of nonconformities with materials supplied by Tube-Line Corporation (T-L) in Information Notice No. 83-07 dated March 7, 1983. This bulletin is being issued as a result of the findings from Region IV Vendor Program Branch inspections at the T-L facilities at Long Island City, New York; Houston, Texas; and Carol Stream, Illinois. It was concluded from these inspections that there are potential generic safety implications at plants which either have received direct shipment materials furnished by T-L (i.e., pipe fittings and flanges) or receive piping subassemblies and other components from holders of ASME Certificates of Authorization which incorporate these materials. Therefore, we ask all recipients of this bulletin to review the information herein for applicability to their facilities and: (1) to take appropriate actions to confirm the adequacy of affected components for intended service; or (2) submit reports stating that T-L materials received from the referenced manufacturing locations will not be used in safety-related systems at their facilities.

Description of Circumstances:

1. On December 6, 1982, Capitol Pipe and Steel Products Company (CPSP Co.) sent a letter to the NRC which identified nonconformities with certain materials that they had obtained from T-L and furnished to their customers. In this letter, CPSP Co. identified the customers who had received the T-L materials and stated that the customers had been notified in regard to the material nonconformities. The products identified by CPSP Co. were carbon steel pipe fittings (caps, tees, and elbows) and flanges. The nonconformities identified were shipment from an unapproved nuclear source, failure to

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perform required heat treatment, and failure to perform nondestructive examinations in accordance with the applicable provisions of the ASME Code.

2. A 10 CFR Part 21 report dated January 10, 1983, was received from the Babcock & Wilcox Company (B&W) which additionally addressed T-L carbon steel materials that had been furnished by CPSP Co. This report stated that certain materials furnished to B&W had been supplied by the Houston, Texas, facility of T-L which was not an approved nuclear supplier per NCA-3800 requirements. It was additionally stated that B&W had performed mechanical testing of four heats of received fittings and flanges and had found that the strength properties in two heats of flange materials were lower than shown on the material certifications; one heat was below the minimum requirement in the procurement specification. Subsequent testing initiated by Carolina Power & Light Company has further identified strength properties in fittings (caps) that were below material specification requirements and differences in chemical analysis from that reported on the T-L Certified Material Test Report.

3. Direct inspections by the Region IV Vendor Program Branch have identified discrepant conditions at both the T-L carbon steel facilities in Long Island City, New York; and Houston, Texas; and the stainless steel facility in Carol Stream, Illinois. Known and intended recipients of these materials are listed in Table 1 - Carbon Steel (see Attachment 2) and Table 2 - Stainless Steel (see Attachment 3).

4. As a result of the inspections performed by the Region IV Vendor Program Branch, the following additional pertinent information has been obtained in regard to materials furnished by T-L:

   a. Certain carbon steel materials were procured from unsurveyed sources and improperly certified to CPSP Co. and Dravo Corporation, Marietta, Ohio, as having been manufactured in accordance with the quality program which CPSP Co. and Dravo Corporation had audited and approved as meeting the requirements of NCA-3800 in Section III of the ASME Code. Similarly, certain stainless steel materials were procured from unsurveyed sources and improperly certified to customers as having been manufactured in accordance with the requirements of NCA-3800 and referencing the T-L (Carol Stream) ASME Certificate of Authorization No. QSC-435. It was additionally established at T-L (Carol Stream) that the basis for approval of some vendors of materials and services (i.e., heat treatment, NDE, mechanical testing, and chemical analysis) was a self-evaluation form filled out by the vendor.

   b. Numerous instances were observed at T-L (Long Island City) in which commercial carbon steel materials (i.e., materials procured solely in accordance with a material specification) had been procured and manufacturing had been completed before the establishment of the T-L quality program. These materials were then furnished to CPSP Co. and Dravo Corporation and improperly certified as being both in accordance with specified Section III of the ASME Code requirements and as having been manufactured in accordance with the quality program

which these companies had audited and approved as meeting the requirements of Section III of the ASME Code. Similar procurement and certification of commercial stainless steel materials was identified by direct inspection at T-L (Carol Stream).

c. Several 3", 600 lb. ASTM A105 carbon steel weld neck flange forgings (e.g., 362 from T-L Heat Code EUUA and 1480 from T-L Heat Code EKP) which have not been appropriately heat treated have been shipped to various customers (both nuclear and non-nuclear). Similarly, the NRC has been notified that documentation was not available to demonstrate that approximately 530 SA-182 stainless steel flanges received the required solution annealing heat treatment. These flanges have been shipped to various customers.

d. Direct inspection has resulted in the current identification that T-L (Carol Stream) furnished 521 stainless steel fittings welded with filler metal to various customers and that vendor documentation for these fittings does not indicate compliance with Section III of the ASME Code requirements in regard to:

(1) Manufacture by a holder of a ASME Certificate of Authorization for an NPT symbol using the ASME accepted QA program.

(2) Use of welders and welding procedure specifications which have been qualified in accordance with the requirements of Section IX of the ASME Code.

(3) Inspection during manufacture by an Authorized Nuclear Inspector and certification by issue and signing of a Partial Data Report Form NM-1.

e. Foreign material manufacturers were surveyed and approved by T-L (Long Island City) representatives as having documented quality assurance programs which were in compliance with the requirements of NCA-3800 in Section III of the ASME Code. Interviews and examinations of available survey/audit records established, however, the following:

(1) Certain manufacturers were approved although they did not, in fact, have documented quality assurance programs.

(2) Other manufacturers were approved although their documented quality assurance programs were not available in the English language and the auditors were unable to read the applicable quality assurance manuals.

(3) Certain manufacturers were maintained on the Qualified Supplier List after identification that previous surveys/audits had improperly stated that the documented quality assurance programs were in full compliance with the requirements of NCA-3800 in Section III of the ASME Code.

f. Materials were procured for T-L (Long Island City) without specification of the applicable nondestructive examination (NDE) requirements contained in Section III of the ASME Code. Subsequent purchase orders to NDE subcontractors referenced either that only commercial NDE was required or specified that a paragraph in Section III of the ASME Code was applicable without reference to or assuring use of an approved NDE procedure. Failure to assure use of approved subcontractor NDE procedures was also identified at T-L (Carol Stream). It was further established that T-L (Carol Stream): (1) failed to have records demonstrating that UT and PT was performed; (2) had no records of eye examinations being given to their level II examiner; (3) failed to have qualification records of their NDE subcontractor's personnel; and (4) did not address acceptance criteria in their PT procedure.

5. Tube-Line Corporation representatives, in a letter to Region IV dated April 8, 1983, stated they would send letters to all identified end-users of their carbon steel products in nuclear facilities. The letters would request meetings with the affected end-users to determine whether the products met the necessary requirements, or to determine whether upgrading or replacements are necessary. Additionally T-L, in letters to Region IV (dated April 25, April 27, May 3, May 6, and May 13, 1983), stated they would send letters to the identified end-users of their stainless steel products of concern.

6. The information available to date indicates that T-L started supplying ASME Code components to the nuclear industry in 1981.

Actions To Be Taken by Holders of Operating Licenses or Construction Permits:

1. Review the lists of purchasing and receiving companies given in Attachments 2 and 3 and determine if any T-L supplied ASME Code materials have been furnished to your facility. The lists of purchasing and receiving companies given in Attachments 2 and 3 have been developed based on correspondence from T-L and inspections at T-L; however, NRC has not verified the completeness of these lists.

2. For ASME Code materials furnished by T-L which are either not yet installed in safety-related* systems at your facility or are installed in safety-related systems of plants under construction, the following actions are requested: (perform action a and either action b or c)

   a. Provide a list of T-L supplied materials and identify the systems in which these materials are/will be installed.

*For the purpose of the applicable actions of this bulletin "safety-related" constitutes those systems covered by the definition given in 10 CFR Part 100, Appendix A Sections III.(c)(1), III.(c)(2), and III.(c)(3). In assessing the impact of T-L supplied materials in other systems at their facilities, licensees should consider the provisions of GDC 1 to 10 CFR Part 50 Appendix A.

b. Implement a program which provides assurance that received materials comply with ASME Code Section III and applicable procurement specification requirements, or which demonstrates that such materials are suitable for intended service. This program should include specific verification that received austenitic stainless steels are in a nonsensitized condition.

c. Replace fittings and flanges with materials which have been manufactured in full compliance with ASME Code Section III and the applicable procurement specification requirements.

3. For ASME Code materials furnished by T-L which are installed in safety-related* systems in operating plants, the following actions are requested:

   a. Provide a list of the T-L supplied materials and identify the systems in which the materials are installed.

   b. Implement a program as discussed in 2b or 2c above.

   c. Provide a basis for continued plant operation if the program requested by Item 3b has not been completed by the time of the bulletin response.

4. Provide a written report within 120 days of the date of this bulletin that either:

   a. States that no T-L supplied materials have been furnished for your facility for use in safety-related systems.

   b. Provides the results of those actions taken in response to Items 2a, 2b, and 2c above, as they apply to materials not yet installed and to materials installed in plants under construction. The report should include your plan and schedule for completing actions 2b and/or 2c.

   c. Provides the results of those actions taken in response to Items 3a, 3b, and 3c above, as they apply to materials installed in operating plants. The report should include your plan and schedule for completing item 3b and your basis for continued operation if item 3b has not been completed.

Although the specific details involving the nonconforming materials supplied by T-L may not directly apply for your facility, you are requested to review the general concerns expressed in the bulletin for applicability at your facility. Your response should describe the results of the review, and if the general concerns apply, you should describe the short-term and long-term corrective actions to be taken and the schedules thereof.

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*For the purpose of the applicable actions of this bulletin "safety-related" constitutes those systems covered by the definition given in 10 CFR Part 100, Appendix A Sections III.(c)(1), III.(c)(2), and III.(c)(3). In assessing the impact of T-L supplied materials in other systems at their facilities, licensees should consider the provisions of GDC 1 to 10 CFR Part 50 Appendix A.

The written reports required by Items 4a, 4b and 4c above shall be submitted to the appropriate Regional Administrator under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954 as amended. In addition, 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 clearance number 3150-0011.

Although no specific request or requirement is intended, the following information would help the NRC evaluate the cost of implementing this bulletin:

1. Utility staff time to perform requested inspections and evaluations.

2. Radiation exposure attributed to requested inspections.

3. Utility staff time spent to prepare written responses.

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

R C DeYoung

Richard C. DeYoung, Director
Office of Inspection and Enforcement

Technical Contact: J. R. Fair, IE
(301) 492-4509

Attachments:
1. List of Recently Issued IE Bulletins
2. Table 1 - Listing of known and intended recipients of Tube-Line Corporation furnished carbon steel materials
3. Table 2 - Listing of known and intended recipients of Tube-Line Corporation furnished stainless steel materials

LIST OF RECENTLY ISSUED IE BULLETINS

Bulletin No.	Subject	Date of Issue	Issued to
Beb 83-05	ASME Nuclear Code Pumps and Spare Parts Manufactured by The Hayward Tyler Pump Company	05/13/83	Utilities with power reactor facilities holding an OL or CP use or plan to use ASME Nuclear Code Pumps Mfg by Hayward Tyler Pump Co.
83-04	Failure of the Undervoltage Trip Function of Reactor Trip Breakers	03/11/83	All PWR facilities holding an OL except W DB type breakers for action and other nuclear reactor facilities for information
83-03	Check Valve Failures in Raw Water Cooling Systems of Diesel Generators	03/11/83	All power reactor facilities holding an OL or CP
83-02	Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants	03/04/83	Table 1 BWRs for action and all other licensees and holders of a CP
83-01	Failure of Reactor Trip Breakers (Westinghouse DB-50) to Open on Automatic Trip Signal	02/25/83	All PWR facilities holding an OL and other power reactor facilities for information
82-04	Deficiencies in Primary Containment Electrical Penetration Assemblies	12/03/82	All power reactor facilities holding an OL or CP
82-03 Rev. 1	Stress Corrosion Cracking in Thick-Wall Large-Diameter Stainless Steel, Recirculation System Piping at BWR Plants	10/28/82	Operating BWRs in Table 1 for action and other OLs and CPs for information

OL = Operating License
CP = Construction Permit

TABLE 1 - CARBON STEEL

PURCHASER	RECEIVING COMPANY	NUCLEAR PLANT[1] (If Known)
Dravo Corporation	Dravo Corp.	Seabrook, Units 1 & 2 Wolf Creek
Capitol Pipe & Steel Products Company	Tennessee Valley Authority (TVA)	Bellefonte
	TVA	Watts Bar
	TVA	Sequoyah
	Pipe Lining & Coating Co., College Point, NY	--
	Metal Bellows Co., Chatsworth, CA	--
	Peter Kiewit	WPPSS
	Pullman Power Products, Paramount, CA	--
	Carolina Power & Light Co.	Shearon Harris
	Duke Power Co.	Catawba
	RECO Industries, Richmond, VA	Shearon Harris Catawba
	Florida Power & Light Co.	St. Lucie
	Babcock & Wilcox Co., Barberton, OH	--

(1) The NRC has not verified that the list of Nuclear Plants receiving Tube-Line materials through any of the listed purchasing and receiving companies is complete.

TABLE 1 - CARBON STEEL

PURCHASER	RECEIVING COMPANY	NUCLEAR PLANT[1] (If Known)
Capitol Pipe & Steel Products Company	Georgia Power Co.	E. I. Hatch
	Baltimore Gas & Electric Co.	Calvert Cliffs
	Stone & Webster Engineering Corp.	Nine Mile Point, Unit 2
	Pittsburgh Des Moines	Beaver Valley, Unit 2
	Duke Power Co.	Oconee
	Koch Process Systems, Westborough, MA	--
	Chicago Bridge & Iron, Birmingham, AL	--
	Public Service Electric & Gas Co.	Salem
	Baldwin Associates	Clinton
	Magnetrol, Downers Grove, IL	--
	Florida Power & Light Co.	Turkey Point
	Stone & Webster Engineering Corp.	Millstone, Unit 3
	Westinghouse	--

(1) The NRC has not verified that the list of Nuclear Plants receiving Tube-Line materials through any of the listed purchasing and receiving companies is complete.

TABLE 1 - CARBON STEEL

PURCHASER	RECEIVING COMPANY	NUCLEAR PLANT[1] (If Known)
Capitol Pipe & Steel Products Company	Crane Chempump, Warrington, PA	--
	ITT Grinnell, Milwaukee, WI	--
	Ionics, Inc., Bridgeville, PA	--
	Portland Eng., Co., Portland, ME	--
	Hap Dong Express, Inc., Brooklyn, NY	--
	Toledo Edison Co.	Davis-Besse
	United McGill Corp., Columbus, OH	--
	Woolley Mfg., Canton, OH	--
	Cherne Contracting	Monticello
	B. F. Shaw, Wilmington, DL	TVA
	Prefex McQuary, New Berlin, WI	--
	Northeast Utilities, Waterford, CT	--
	GPU Nuclear, Salem, NJ	--

(1) The NRC has not verified that the list of Nuclear Plants receiving Tube-Line materials through any of the listed purchasing and receiving companies is complete.

TABLE 2 - STAINLESS STEEL

PURCHASERS	RECEIVING COMPANY	DISCREPANCY(1)	NUCLEAR PLANT(2) (If Known)
Capitol Pipe & Steel Products Company	Bechtel Power Corp.	B	Palisades
	Tennessee Valley Authority	B	Watts Bar
	Baldwin Associates	A	Clinton
	Metal Bellows, Chattsworth, CA	-	--
	UE&C	-	Davis-Besse
	SMUD	-	--
Chicago Tube & Iron Co.	Commonwealth Edison Co.	A & B	Byron Station
Guyon Alloys	Reco Industries, Richmond, VA	A & B	--
	Northeast Utilities	A	Millstone Units 1 & 2
	Pittsburgh Des Moines Steel Corp., Provo, UT	B	San Onofre
	Bechtel Power Corp.	A	Hope Creek
	Bechtel Power Corp.	A & B	Limerick
	Baldwin Associates	B	Clinton
	D. G. O'Brien	A	Niagara Mohawk
	Bell Schneider	A	Ginna
	Duke Power	-	Catawba
	Bechtel	-	Palisades Midland
	Portland Engineering, Portland, ME	-	--
Tubeco	Tubeco., Inc., Brooklyn, NY	B	--
Joliet Valves, Inc.	Commonwealth Edison Co.	B	Quad Cities
	Liquid Carbonics, c/o R. Alger Co., Kenner, LA	B	--

(1) Discrepancy: A. Solution Annealing Heat Treatment Not Performed
B. Fittings Welded with Filler Metal

(2) The NRC has not verified the list of Nuclear Plants receiving Tube-Line materials through any of the listed purchasing and receiving companies is complete.

TABLE 2 - STAINLESS STEEL

PURCHASERS	RECEIVING COMPANY	DISCREPANCY(1)	NUCLEAR PLANT(2) (If Known)
	Liquid Carbonics, Buenos Aires, Argentina	A	--
	Joliet Valves, Inc., Minooka, IL	A	Braidwood
Barr Saunders	Commonwealth Edison Co.	A	Zion
Gulf Alloy	Gulf Alloy, Houston, TX	A	--
	Carolina Power & Light	-	H. B Robinson
Hub, Inc.	Hub, Inc., Tucker, GA	A	--
	Niagra Mohawk	-	Nine Mile Pt.
	Georgia Power	-	Hatch
	Florida Power & Light	-	Turkey Pt.
Atlas Alloys	Atlas Alloys	-	--
Louis P. Canuso Inc.	Bechtel	-	Susquehanna
TEK	TEK	-	--
Gray Pipe	Gray Pipe	-	--
Liberty Equipment	Bingham-Willamette, Portland, OR	-	--
	Bechtel	-	Midland
Consolidated Pipe & Supply Co.	Georgia Power	-	Hatch Vogtle

(1) Discrepancy: A. Solution Annealing Heat Treatment Not Performed
B. Fittings Welded with Filler Metal

(2) The NRC has not verified the list of Nuclear Plants receiving Tube-Line materials through any of the listed purchasing and receiving companies is complete.

Georgia Power

D. O. Foster

November 23, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

Reference:
RII: JPO:
50-424
50-425

Attention: Mr. James P. O'Reilly

File: X7BC24
Log: GN-287

Gentlemen:

The following is submitted in response to I&E Bulletin 83-06, "Non-conforming Materials Supplied by Tube-Line Corporation Facilities at Long Island City, New York; Houston, Texas; and Carol Stream, Illinois:"

The attached list contains all ASME III Code materials supplied by Tube-Line Corporation to the Vogtle Electric Generating Plant (VEGP) through primary vendors as of 11/18/83. Sub-vendors have been contacted and requested to provide information regarding any ASME code materials supplied by Tube-Line Corporation used in VEGP components. This information will be provided to the NRC when it becomes available.

Georgia Power Company has submitted one fitting from each heat of material supplied by Tube-Line Corporation to the Vogtle Project Architectural - Engineer and "N" Certificate Holder, Bechtel Power Corporation, for chemical and physical property analysis. Copies of the Certified Material Test Reports (CMTR's) were sent with each fitting. Bechtel will conduct a Supplier Quality Audit of Tube-Line Corporation and their sub-suppliers to verify by objective evidence that a program for material traceability was established during the time of manufacture and that no welding was performed on the fittings. If the audit produces satisfactory results, the sample of fittings from the jobsite will be tested for chemical and physical properties.

After satisfactory completion of the Tube-Line audit and material property tests, Bechtel will recommend that the Vogtle Project invoke Code Case N-242-1 in order to make the remainder of the fittings (other than the sample tested) supplied by Tube-Line acceptable for ASME Section III applications. Code Case N-242-1 has been accepted by the USNRC as discussed in Regulatory Guide 1.85.

This reply contains no proprietary information and may be placed in the NRC's Public Document Room.

Very truly yours,

D. O. Foster

D. O. Foster

REF/DOF/cc

D. O. Foster states that he is the Vice President and General Manager of Vogtle Project and is authorized to execute this oath on behalf of Georgia Power Company and that to the best of his knowledge and belief the facts set forth in this letter are true.

GPC: D. O. Foster
D. O. Foster

Sworn to and subscribed before me this 23rd day of November, 1983.

attachment

xc: U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

O. Batum
J. A. Bailey
E. D. Groover
L. T. Gucwa
M. Malcom
G. Bockhold, Jr.
P. D. Rice

The following ASME III materials, supplied by Tube-Line Corporation, have been received at the Vogtle Electric Generating Plant as of 11/18/83. System designations are not made until the materials are released from storage and installed in the plant.

Description	Quantity
½" Elbow, BW, seamless, 90° LR, S/160, SA-403, WP304L, Class 1	150
3/4" "	150
1" "	127
1½" "	126
2" "	169
½" Elbow, BW, seamless, 90° LR, S/40S, SA-403, WP304L, Class 2	200
3/4" "	200
1" "	200
1½" "	200
2" "	200
½" Elbow, BW, seamless, 45° LR, S/160, SA-403, WP304L, Class 1	80
3/4" "	81
1" "	80
1½" "	80
2" "	78
½" Elbow, BW, seamless, 45° LR, S/40S, SA-403, WP304L, Class 2	100
3/4" "	100
1" "	100
1½" "	98
2" "	100
½" Tee, BW, seamless, S/160, SA-403, WP304L, Class 1	50
3/4" "	100
1" "	50
1½" "	50
2" "	50
½" Tee, BW, seamless, S/40S, SA-403, WP304L, Class 2	100
3/4" "	100
1" "	100
1½" "	100
2" "	100
½" Cap, BW, seamless, S/40S, SA-403, WP304L. Class 2	50
3/4" "	50
1" "	50
1½" "	50
2" "	50
3/4"x½" Reducer, concentric, BW, seamless, S/40S, SA-403, WP304L, Class 2	40
1"x3/4" "	50
1"x½" "	40
1½"x1" "	50
1½"x3/4" "	40
1½"x½" "	20
2"x1½" "	100
2"x1" "	50
2"x3/4" "	30
2"x½" "	20
3/4"x½" Reducer, eccentric, BW, seamless, S/40S, SA-403, WP304L, Class 2	5
1"x3/4" "	5
1"x½" "	5

1½"x1" Reducer, eccentric, BW, seamless, S/40S, SA-403, WP304L, Class 2	5
1½"x3/4" "	5
1½"x½" "	5
2"x1½" "	5
2"x1" "	5
2"x3/4" "	5
2"x½" "	5
½" Flange, SW, RF, 2500#, S/160 Bore, SA-182, F316, Class 1	10
3/4" "	25
1" "	25
1½" "	25
2" "	25
½" Flange, WN, RF, 2500#, S/160, SA-182, F316, Class 1	10
3/4" "	20
1" "	20
1½" "	20
2" "	20
½" Flange, SW, RF, 1500#, S/160 Bore, SA-182, F316, Class 2	80
3/4" "	100
1" "	100
1½" "	100
2" "	102
½" Flange, SW, RF, 1500#, S/80S Bore, SA-182, F316, Class 2	10
3/4" "	20
1" "	20
1½" "	20
2" "	20
½" Flange, SW, RF, 600#, S/40S Bore, SA-182, F316, Class 2	10
3/4" "	20
1" "	20
1½" "	20
2" "	20
½" Flange, SW, RF, 300#, S/40S Bore, SA-182, F316, Class 2	150
3/4" "	205
1" "	200
1½" "	200
2" "	200
½" Flange, SW, RF, 150#, S/40S Bore, SA-182, F316, Class 2	150
3/4" "	200
1" "	200
1½" "	200
2" "	200
½" Flange, WN, RF, 1500#, S/160, SA-182, F316, Class 2	50
3/4" "	75
1" "	75
1½" "	75
2" "	75
½" Flange, WN, RF, 1500#, S/40S, SA-182, F316, Class 2	50
3/4" "	75
1" "	75
1½" "	75
2" "	75

½" Flange, blind, RF, 2500#, SA-182, F316, Class 1	30
3/4" "	40
1" "	40
1½" "	40
2" "	40
½" Flange, blind, RF, 1500#, SA-182, F316, Class 2	40
3/4" "	50
1" "	50
1½" "	50
2" "	50
½" Flange, blind, RF, 600#, SA-182, F316, Class 2	10
3/4" "	20
1" "	20
1½" "	20
2" "	20
½" Flange, blind, RF, 300#, SA-182, F316, Class 2	50
3/4" "	68
1" "	75
1½" "	75
2" "	75
½" Flange, blind, RF, 150#, SA-182, F316, Class 2	75
3/4" "	100
1" "	100
1½" "	100
2" "	100
2" Flange, orifice, WN, RF, 2500#, S/160, SA-182, F316, Class 1	5 pr.
1" "	5 pr.
2" Flange, orifice, WN, RF, 1500#, S/160, SA-182, F316, Class 2	10 pr.
1½" "	19 pr.
1" "	10 pr.
3/4" "	10 pr.
2" Flange, orifice, WN, RF, 1500# S/80S, SA-182, F316, Class 2	4 pr.
1½" "	8 pr.
1" "	8 pr.
3/4" "	4 pr.
2" Flange, orifice, WN, RF, 600#, S/40S, SA-182, F316, Class 2	4 pr
1½" "	4 pr.
1" "	4 pr.
3/4" "	4 pr.
2" Flange, orifice, WN, RF, 300#, S/40S, SA-182, F316, Class 2	20 pr.
1½" "	16 pr.
1" "	20 pr.
3/4" "	15 pr.
2" Flange, SW, FF, 300#, S/40S Bore, SA-182, F316, Class 2	6
1½" "	6
1" "	6
3/4" "	6
½" "	6
2" Flange, SW, FF, 150#, S/40S, SA-182, F316, Class 2	6
1½" "	6
1" "	6
3/4" "	6
½" "	6
4" Slip-on, RF, 150#, A-105, Grade 79	8

Some of the materials listed above have been installed in the following safety-related systems:

Nuclear Service Cooling Water System - Units 1 and 2
Safety Injection System - Units 1 and 2
Residual Heat Removal System - Unit 1
Containment Spray System - Units 1 and 2
Chemical Volume and Control System - Units 1 and 2
Containment and Auxiliary Building Drain System - Radioactive - Unit 1

[illegible]
[illegible]
Waynesboro Georgia [illegible]
Telephone 404 [illegible]
[illegible]

Georgia Power

D. O. Foster
Vice President and Project
General Manager
Vogtle Project

March 22, 1984

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

Reference:
RII: JPO:
50-424
50-425

Attention: Mr. James P. O'Reilly

File: X7BC24
Log: GN-331

Gentlemen:

The following is submitted further to our letter of November 23, 1983 (File: X7BC24, Log: GN-287), in response to I & E Bulletin 83-06, "Nonconforming Materials Supplied by Tube-Line Corporation Facilities at Long Island City, New York; Houston, Texas; and Carol Stream, Illinois:"

The Vogtle Project Architect/Engineer and "N" Certificate Holder, Bechtel Power Corporation, has completed its Supplier Quality Audit of Tube-Line Corporation's Carol Stream Facility. As a result of the audit, Bechtel's recommendations regarding the actions required to resolve I & E Bulletin 83-06 have changed significantly from those related in our earlier response. The audit verified that although Tube-Line sub-vendors' quality assurance programs did not meet all the requirements of Subsection NA-3700/NCA-3800 of the ASME Boiler and Pressure Vessel Code (hereafter referred to as the "Code"), Tube-Line audit checklists indicated that in all cases suppliers' material identification and control programs were satisfactory.

Based on the results of the audit, Bechtel now recommends that GPC invoke subarticle NX-2600 from the 1977 Edition of Section III of the Code in lieu of invoking Code Case N-242-1. All fittings supplied to the Vogtle Project directly from Tube-Line are two inches and smaller. Subarticle NX-2600 exempts two inches and smaller flanges and fittings from all NA 3700/NCA 3800 quality program requirements except for the requirements of NA 3767.4. Actions are being taken to satisfy the requirements of NA 3767.4 for this material.

To date, ninety-three (93) separate heats of materials two inches and smaller have been identified which were supplied to the Vogtle Project by Tube-Line. In lieu of performing chemical and physical property analyses on each of the ninety-three heats, Bechtel now recommends that only a representative sample of fourteen (14) heats (fifteen percent of the identified heats) be tested. The adoption of NX-2600 allows random sampling in lieu of testing each heat since testing is now for verification of original test results rather than for the fulfillment of code requirements. The attached list provides the heat identifications made to date as well as the subvendor supplying the material to Tube-Line.

In addition to material two inches and smaller, eight 4" raised-face, 150#, ASTM A-105, Grade 79, slip-on flanges have been identified as being supplied by Tube-Line through Johnston Pump Company of Glendora, California. These flanges were used in the fabrication of two Unit 2 Diesel Oil Storage Tank Pumps (Plant Equipment Nos. 2-2403-P4-001 and 2-2403-P4-002), which are Seismic Category 1, safety-related, Code Section III, Class 3 components. These fittings will require a complete evaluation relative to Code requirements.

Georgia Power Company is continuing to search all incoming material documentation packages for material supplied by Tube-Line Corporation. We will continue to keep the USNRC informed of the progress of our evaluation of this matter.

This report contains no proprietary information and may be placed in the NRC's Public Document Room.

Very truly yours,

D. O. Foster

D. O. Foster

REF/DOF/tdm

Attachment

D. O. Foster states that he is the Vice President and Project General Manager of Vogtle Project and is authorized to execute this oath on behalf of Georgia Power Company and that to the best of his knowledge and belief the facts set forth in this letter are true.

GPC: D. O. Foster
D. O. Foster

Sworn to and subscribed before me this 26th day of March, 1984.

Jean J. Bishop

Notary Public, Georgia, State at Large
Jan. 7, 1987

xc: U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

O. Batum
J. A. Bailey
E. D. Groover
L. T. Gucwa
M. Malcom
G. Bockhold, Jr.
P. D. Rice

SUMMARY OF HEAT NUMBERS AND PURCHASE ORDER ITEM NUMBERS BY VENDOR

VENDOR	HEAT #	P.O. ITEM #'s
1) NBK (Tokyo)	AABK	131
Nippon Benkan Kogyo	AABL	132
T/L P.O. Req'ts:	AABS	554, 555
1) ASTM Mat'l	AABT	558
2) Heat Treated (HT)	AACG	99
	AACF	062
1900°F Water Quench	AACS	133
	AADA	161
	AADAB	552, 572, 573
	AADAC	571, 575, 551
	AADC	097, 098
	AAEB	063
	AAEBT	536
	AAEBU	537
	AAEBUA	538
	AAECD	539
	AAECE	540
	AAEP	553
	AAES	559
	AAET	578
	AAYW	065
	ABYA	064
	ABYB	574
	ABYC	577
	ACHA	065, 557
2) KUZE (Tokyo)	AAWW	100
Taikyo Sangyo Co. Ltd	AAWU	99
T/L P.O. Req'ts:	AAWV	96
1) ASTM Mat'l	ABTA	556
2) Heat Treated	ABTB	560
	ABZN	555
1900° for 1 hr/inch	ABZO	559
Water Quenched	ABZY	554
	ACKQ	576
	ACKR	579

VENDOR	HEAT #	P.O. ITEM #'s
3) METALFAR (Como, Italy) T/L P.O. Req'ts: 1) ASTM Mat'l 2) No Heat Treatment Tubeline Subcontracted HT, Chemical & Physical Tests	ABHA	670
	ABJB	669, 727, 614, 611, 666, 730
	ABNE	667, 729, 612
	ABOC	651, 641, 596, 636, 637, 591, 642, 592, 609, 722, 597, 638, 652, 653, 598, 643, 593
	ABOD	668, 728, 613
	ABPD	610
	ABRD	585, 590, 650, 691, 584, 694, 595
	ABRE	647, 594, 599, 639, 644, 653, 654, 695, 696, 699, 700, 703, 704, 707, 708, 581, 587, 582, 646, 586, 643
	ABUD	605, 659, 660, 604, 692, 588, 583, 648
	ABUE	615, 726, 693, 694, 697, 701, 702, 705, 706, 601, 602, 603, 654, 656, 657, 658, 600, 640, 655, 695, 645
	ABUF	698, 694, 598, 584, 649
	ABWA	608, 663, 723, 661, 725, 606
	ABWB	609, 664
	ACAB	607, 665, 724, 610, 662, 721, 607
4) HACKNEY (Dallas, Texas) T/L P.O. Req'ts: 1) ASTM Mat'l 2) Heat Treated	ABVAG	552
	$ABVA^{(U)}_{(V)}$	134
5) MACLINE (Montreal, Canada) T/L P.O. Req'ts: 1) ASTM Mat'l 2) Heat Treated HT was not performed by Macline. Sub- contracted by T/L	ACBB	132
	ACBC	134
	ACBD	135
	ACNA	131

VENDOR	HEAT #	P.O. ITEM #	P.O. NUMBER	P.O. DATE
6) CUSTOM ALLOY	PAUA	LATER	# 8172	6/30/81
(Califon, N.J.)	PAUAB	LATER	# 8172	6/30/81
T/L P.O. Req'ts:	PAUAC	LATER	# 8172	6/30/81
1) ASME III Class 3	PAUAD	LATER	# 8172	6/30/81
to NCA-3800	PAUAF	LATER	# 8172	6/30/81
2) Heat Treated	PAUAG	LATER	# 8172	6/30/81
	PAUAH	LATER	# 8172	6/30/81
T/L relieved in house	PAUAI	LATER	# 8172	6/30/81
then subcontracted	PAUAJ	LATER	# 8172	6/30/81
UT/PT - T/L did some	PAUAK	LATER	# 8172	6/30/81
PT in house to a	PAUAL	LATER	# 8172	6/30/81
procedure without	PAUAM	LATER	# 8172	6/30/81
acceptance criteria	PAUAN	LATER	# 8172	6/30/81
meeting NB requirements.	PAUAO	LATER	# 8172	6/30/81
Mat'l furnished as	PAUAP	LATER	# 8172	6/30/81
Class 1 to Consolidated	PAUAQ	LATER	# 8172	6/30/81
	PAUAR	LATER	# 8172	6/30/81
	PAUAS	LATER	# 8172	6/30/81
	PAUAT	LATER	# 8172	6/30/81
	PAUAV	LATER	# 8172	6/30/81
	PAUAW	LATER	# 8172	6/30/81
	PAUB	LATER	# 8172	6/30/81
	PAUE	LATER	# 8172	6/30/81
	PAUF	LATER	# 8172	6/30/81
	PAUG	LATER	# 8172	6/30/81
	PAUH	LATER	# 8172	6/30/81
	PAUI	LATER	# 8172	6/30/81
	PAUJ	LATER	# 8172	6/30/81
	PAUK	LATER	# 8172	6/30/81
	PAUL	LATER	# 8172	6/30/81
	PAUM	LATER	# 8172	6/30/81
	PAUO	LATER	# 8172	6/30/81
	PAUQ	LATER	# 8172	6/30/81
	PAUT	LATER	# 8172	6/30/81
	PAUW	LATER	# 8172	6/30/81
	PAUX	LATER	# 8172	6/30/81
	PAUY	LATER	# 8172	6/30/81
7) CAPITOL	HEAT #'s	698		
	LATER	699		
		705		
		707		

Attachment 18

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Georgia Power
Southern Company Services

D. O. Foster
Vice President and General Manager
Vogtle Project

July 20, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II-Suite 3100
101 Marietta Street
Atlanta, Georgia

File: X7B603-M45
Log: GN-244

Reference: Vogtle Electric Generating Plant - Units 1 and 2
40-424, 40-425, NSSS Protection System Relay Cards

Attention: Mr. James P. O'Reilly

Gentlemen:

On June 16, 1983, Georgia Power Company reported a potential deficiency to the NRC concerning Westinghouse NSSS Protection System Relay Cards. Westinghouse reported this problem and a solution to the NRC in Westinghouse letter NS-EPR-2774, dated June 1, 1983. At this time, Georgia Power Company is reviewing the design at Plant Vogtle to determine if any modifications are needed. Georgia Power Company will submit a final report on this issue to the NRC by August 30, 1983.

Yours truly,

D. O. Foster

D. O. Foster

DOF/REF/cc

xc: U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

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Georgia Power

the southern electric system

D. O. Foster
Vice President and General Manager
Vogtle Project

September 6, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
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Atlanta, Georgia 30303

File: X7BG03 - M45
Log: GN-253

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424, 50-425; Westinghouse NSSS Protection System
Relay Cards, GPC Letter GN-244, dated July 20, 1983

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has completed its evaluation and determined that the above referenced concern represents a significant deficiency for the Vogtle Electric Generating Plant-Units 1 and 2. Westinghouse had previously informed the NRC of the existence of this problem in their letter NS-EPR-2774, dated June 1, 1983. Since Vogtle was referenced in this letter, Georgia Power Company considers this Westinghouse letter to constitute notification of a substantial safety hazard as required in Part 10 CFR 21. Enclosed is a copy of our evaluation.

I acknowledge that our evaluation report is six days late of our committed response date of August 30, 1983. An extension was verbally granted by Mr. J. Roggi of your office.

This response does not contain any proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/REF/cc

enclosure

xc: U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

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D. E. Dutton	H. H. Gregory III	J. L. Vota
W. F. Sanders	R. E. Folker	

EVALUATION FOR A SUBSTANTIAL SAFETY HAZARD
EVALUATION FOR A SIGNIFICANT DEFICIENCY

Westinghouse NSSS Protection System Relay Cards

Initial Report:

The failure of NTC card relays in the process protection system during seismic testing was initially reported by Westinghouse to the NRC in their letter NS-EPR-2774, dated June 1, 1983. Georgia Power Company reported this concern to the NRC as a potential deficiency on June 16, 1983. In an interim report to the NRC, Georgia Power Company indicated that the NRC could expect a final report on August 30, 1983.

Background Information:

The 7300 series temperature channel test (NTC) card provides a means of conveniently testing the temperature measurement channels while the system is online. This is accomplished by switching the resistance temperature detector (RTD) bulb outputs to monitoring test points.

During seismic testing of the temperature channel test (NTC) card, contact bounce was experienced in the mercury relay utilized on this card. The intermittent contact bounce will result in signal saturation of the downstream RTD amplifier (NRA) card in the T_{HOT} and T_{COLD} circuits of the Westinghouse 7300 process protection system. The Vogtle Electric Generating Plant uses unfiltered signals. Saturation of the downstream RTD amplifier (NRA) card could delay initiation of the overtemperature Delta T and overpressure Delta T trips.

The overtemperature Delta T trip is designed to protect against a departure from nucleate boiling (DNB) which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant, resulting in high fuel clad temperatures.

The overpower Delta T trip is designed to protect against a high fuel power density (excessive kw/ft) and subsequent fuel rod cladding failure and fuel meltdown.

Engineering Evaluation:

The intermittent contact bounce of the 7300 temperature channel test (NTC) card will result in signal saturation of the downstream RTD amplifier (NRA) card in the T_{HOT} and T_{COLD} circuits. Saturation of the NRA card on the unfiltered signal could delay initiation of the overtemperature Delta T and overpower Delta T trips. Failure to obtain these trips could result in the departure from nucleate boiling (DNB) ratio being reduced below 1.30. This is a violation of a safety limit.

Conclusion:

The failure mode identified by Westinghouse in conducting seismic testing of the Temperature Channel Test (NTC) card represents a significant deficiency found in design

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Georgia Power
the southern electric system

D. O. Foster
Vice President and General Manager
Vogtle Project

November 3, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II-Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M48
Log: GN-275

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424, 50-425; Westinghouse-NLP Printed Circuit Cards

Attention: Mr. James P. O'Reilly

Gentlemen:

On October 4, 1983, Mr. C. W. Hayes identified a potential deficiency to Mr. J. Rogge concerning the possibility of heat sinks becoming detached from the back of Westinghouse NLP printed circuit cards. Since these sinks are metallic, the possibility exists that some circuits could be shorted out. At this time, Georgia Power Company is conducting a review to determine which, if any, cabinets contain these printed circuit cards. Georgia Power Company also intends to conduct an engineering review to ensure possible system interaction effects are also reviewed.

Based upon the completion schedule proposed by the architect/engineer, Georgia Power Company expects to submit a final report to the NRC on this concern by December 21, 1983.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/REF/cc

xc: U. S. Regulatory Commission
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Washington, D. C. 20555

xc:

R. J. Kelly	R. H. Pinson	E. D. Groover
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W. F. Sanders	C. W. Hayes	

Georgia Power

Foster

December 27, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M48
Log: GN- 298

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424, 50-425; Westinghouse NLP Printed Circuit Cards;
Letter GN-275 dated 11/3/83

Attention: Mr. James P. O'Reilly

Gentlemen:

In our previous letter on this subject, Georgia Power Company indicated that the NRC could expect to receive notification from Georgia Power Company concerning the results of our evaluation by December 21, 1983. Some delays have been experienced on completing the engineering evaluation as planned. The evaluation is now scheduled for completion during the week of January 8, 1984. Georgia Power Company expects to inform the NRC of the results of this evaluation on January 13, 1984.

This response does not contain any proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

R. E. Conway for

D. O. Foster

DOF/REF/sb

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Washington, D.C. 20555

R. J. Kelly	D. E. Dutton	J. A. Bailey	L. T. Gucwa
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Power

January 11, 1984

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II-Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M48
Log: GN-303

Reference: Vogtle Electric Generating Plant - Unit 1,
50-424; Westinghouse NLP Printed Circuit Cards
GN-275, dated 11/3/83, GN-298, dated 12/27/83

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has completed its investigation into the above referenced concern and has concluded that a reportable event could exist. Based upon NRC guidance in NUREG-0302, Revision 1 and other NRC correspondence, Georgia Power Company is reporting this concern under 10 CFR 50.55(e), since duplicate reporting of a 10 CFR Part 21 and 10 CFR 50.55(e) event is not required. Enclosed is a copy of the evaluation performed for this concern.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/REF/cc

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EVALUATION FOR A SUBSTANTIAL SAFETY HAZARD
EVALUATION FOR A SIGNIFICANT DEFICIENCY

Westinghouse NLP Printed Circuit Cards

Initial Report:

On June 1, 1983, Westinghouse (E. P. Rahe) informed the NRC (R. C. DeYoung) of two potential problems within the Westinghouse 7300 Process Protection System. One of these problems involved heat sink adhesive failures of printed circuit cards. (The other concern involved the failure of NTC relay cards and was reported by Georgia Power Company to the NRC on September 6, 1983.)

Westinghouse did not list the Vogtle Electric Generating Plant in their list of plants having potentially defective NLP cards. Subsequent investigations identified that three defective NLP cards were located in the balance of plant (BOP) process cabinet. Georgia Power Company notified the NRC of this potential concern on October 4, 1983.

Background Information:

Heat sink adhesive failures were identifed at several utilities and were reported to Westinghouse as field deficiencies All reported field failures have occurred only on the loop power supply (NLP) cards. Westinghouse has determined that NLP printed circuit cards shipped from Westinghouse Industry Electronic Division (WIED) between August 1, 1980, and September 1, 1982, were equipped with a thermal heat sink assembly on the inverter transistors that is subject to potential failure in the adhesive bond in the thermal link assembly. The adhesive bond is between an insulating washer and the thermal link. Failure of the bond can cause the heat sink plate to separate from the thermal links and fall off of the printed circuit board. The plate is made of conductive metal and under certain circumstances could cause shorting of the low level signals if it became wedged between cards in the card frame. Westinghouse has not defined the adhesive failure mechanism or the expected number of hours of system operation before failure. For Plant Vogtle, the potentially defective heat sink assembly can be recognized by the hex nuts visible on the top side of the assembly. The engineering review identified the following defective cards in the balance of plant process cabinet 1604QSPP1 (C-1) for Unit 1 only.

1. PQY-3000 -- Provides power supply to instrumentation loop P-3000 which controls the atmospheric power operated relief valve PV-3000.

2. QY-2791B -- Provides an isolated signal for the Train A diesel generator power output indication and computer input.

3. FQY-77744 -- Provides power supply to instrumentation loop F-7774 which provides the automatic start signal to a reactor makeup pump.

An analysis of potential interactions was conducted to determine if there was a potential impact on plant safety. The analysis assumed the preexistence of a short circuit resulting from potential physical interactions with the NLP card conductive plate concurrent with the most limiting single active failure following the onset of an event (transient or accident condition) requiring a response from the impacted card.

Engineering Evaluation:

A total of 12 potential interactions were identified and analyzed for impact on plant safety. Of these 12 interactions, 2 resulted in unacceptable consequences.

In one case the potential interactions of an NLP card conductive plate could render the train A penetration area ESF filtration unit inoperable. Failure of the train A unit concurrent with a single failure of the train B unit following a LOCA results in the inability to maintain a negative pressure in the penetration area. This could result in the unfiltered release of radioactivity to the environment in excess of analyzed releases. Although a qualitative assessment of this scenario indicates that the total offsite doses, including the unfiltered release, are within the 10 CFR 100 guidelines, the scenario represents an unanalyzed event which is not considered in the present safety analysis.

The other case involved the potential interaction of an NLP card conductive plate which renders the isolation valve on the auxiliary component cooling water (ACCW) return header from the reactor coolant pump (RCP) #2 thermal barrier inoperable. This valve functions to automatically isolate the ACCW return header in the event of a thermal barrier failure. A backup isolation valve is provided for redundancy. Failure of one valve due to the instrument loop interaction with the NLP card conductive plate concurrent with a single failure in the backup valve results in the inability to mitigate the consequences of a thermal barrier failure and the transport of reactor coolant to the auxiliary building environment through the ACCW surge tank overflow. Although a qualitative assessment of this scenario indicates that the total offsite dose is a small fraction ofthe 10 CFR 100 guidelines, the scenario represents an unanalyzed event which is not considered in the present safety analysis.

It should be pointed out that the potential interaction of these heat sinks with other systems is a low probability event. The system interaction study was conducted to identify possible failure modes.

Review for Reportability

NUREG-0302 Revision 1, "Remarks Presented at Public Regional Meetings to Discuss Regulations for Reporting of Defects and Noncompliance" was reviewed for guidance relating to what the NRC may consider to be a major reduction in the degree of protection provided to public health and safety. The following guidance was obtained from page 21.3(k)-1 concerning the definition of substantial safety hazard:

Question 1: Please elaborate on the definition of "Substantial Safety Hazard" as used in Part 21. For instance, give examples of what the NRC would consider to be a "major reduction in the degree of protection provided to public health and safety."

Response:

Appendix A to NUREG-0090-7, Report to Congress on Abnormal Occurrences, June 1977 lists a number of events that may help to illustrate the NRC concept of "Substantial Safety Hazard." Specific illustrations of what we would consider to be "major reduction in the degree of protection provided to public health and safety" include:

- o Exposure in excess of 25 rems, whole body (10 CFR 20.403(a)(1))
- o Exposure of an individual in an unrestricted area to more than 0.5 rem in one calendar year (10 CFR 20.105(a))
- o Release of radioactive material to an unrestricted area in excess of 500 times the limit of Appendix B, Table II, 10 CFR Part 20 (10 CFR 20.403(b))
- o Exceeding a safety limit as defined in the facility technical specifications
- o A deficiency which seriously compromised the ability of a confinement system to perform its designated function

Also, in the section concerning "Remarks by The Office of Standards Development to Public Regional Meetings on 10 CFR Part 21 by W. E. Campbell, Jr." page 6 states:

"Basic component" as applied to nuclear power reactors means a plant structure, system, component or part thereof necessary to assure (1) integrity of the reactor coolant pressure boundary or (2) the capability to shut down the reactor and maintain it in a safe shut down condition or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in Section 100.11 of Title 10. (This definition also appears in Part 10 CFR 21 as definition of a basic component).

NOTE: Attachment A is a restatement of Sec. 100.11.
Attachment B is a restatement of Sec. 20.403
Attachment C is a restatement of Sec. 20.105

Based upon NRC guidance in the above referenced reports and attachments, Georgia Power Company realizes that these events may not exceed Part 10 CFR 100 limits and may therefore not be considered reportable under Part 10 CFR 21 or 10 CFR 50.55(e). However, the assumed failure of one train of the ESF B train penetration area filtration unit and resulting failure of the A train from the failure of the NLP card does represent a potential system interaction that results in the inability of a confinement system to perform its designated function. Thus this event can constitute a substantial safety hazard due to the potential to have unfiltered plant releases to the atmosphere. Since the magnitude of the releases have not been absolutely quantified, Georgia Power Company considers this concern reportable since the health and safety of the public may be affected.

Also, guidance in 10 CFR 50.55(e) requires the reporting of any event that could adversely affect the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant. The potential to have plant releases in excess of those analyzed in the FSAR does constitute an effect on the safety of operations since health and safety of the public may be affected.

ATTACHMENT A

Restatement of Sec. 100.11

Sec. 100.11. Determination of exclusion area, low population zone, and population center distance - (a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultanous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

ATTACHMENT B

RESTATEMENT OF 10 CFR 20 SEC. 20.403

Sec. 20.403. Notifications of incidents - (a) Immediate notification. Each licensee shall immediately report any events involving byproduct, source or special nuclear material possessed by the licensee that may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiaton; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $200,000.

(b) Twenty-four hour notification. Each licensee shall within 24 hours of discovery of the event, report any event involving licensed material possessed by the licensee that may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $2000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

(d) Reports made by licensees in response to the requirements of this section must be made as follows:

(1) Licensees that have an installed Emergency Notification System shall make the reports required by paragraphs (a) and (b) of this section to the NRC Operations Center in accordance with § 50.72 of this chapter.

(2) All other licensees shall make the reports required by paragraphs (a) and (b) of this section by telephone and by telegram, mailgram, or facsimile to the Administrator of the appropriate NRC Regional Office listed in Appendix D of this part.

Note: Item (b)2 of Sec. 20.403 refers to Appendix B; Table II refers to concentrations in air and water above natural backgrounds. This table is not reproduced due to its length.

ATTACHMENT C
RESTATEMENT OF SECTION 20.105

Sec. 20.105. Permissible levels of radiation in unrestricted areas. (a) There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive material and other sources of radiation. Such applications should include information as to anticipated average radiation levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem.

(b) Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possession:

(1) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose of excess of two millirems in any one hour. or

(2) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of 100 millirems in any seven consecutive days.

(c) In addition to other requirements of this part, licensees engaged in uranium fuel cycle operations subject tothe provisions of 40 C. F. R. Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," shall comply with that part.

Attachment 19

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Georgia Power
Southern Company Services

D. O. Foster
Vice President and General Manager
Vogtle Project

June 13, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M40
Log: GN-235

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424, 50-425; Westinghouse DS-416 Reactor Trip Breakers

Attention: Mr. James P. O'Reilly

Gentlemen:

On May 5, 1983, Mr. C. W. Hayes, Quality Assurance Manager - Plant Vogtle, reported a potential deficiency to the NRC concerning the above referenced subject. At this time, Georgia Power Company is still reviewing this subject. Georgia Power Company expects to resolve this item and submit a final report if necessary, by September 9, 1983.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/CWH/skr

xc: U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

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Georgia Power
the southern electric system

D. O. Foster
Vice President and General Manager
Vogtle Project

Read ...

September 6, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II-Sutie 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M40
Log: GN-254

Reference: Vogtle Electric Generating Plant-Units 1 and 2
50-424, 50-425; Westinghouse DS-416 Reactor Trip Breakers,
GPC Letter GN-235, dated June 13, 1983

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has completed its evaluation and determined that the above referenced concern represents a significant deficiency for the Vogtle Electric Generating Plant-Units 1 and 2. Westinghouse had previously informed the NRC of the existence of this problem in their letter NS-EPR-2744, dated March 31, 1983. Since Vogtle was referenced in this letter, Georgia Power Company considers this Westinghouse letter to constitute notification of a substantial safety hazard as required in Part 10 CFR 21. Enclosed is a copy of our evaluation.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/REF/cc

enclosure

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W. F. Sanders	R. E. Folker	

EVALUATION FOR A SIGNIFICANT DEFICIENCY

Westinghouse DS-416 Reactor Trip Switchgear Breakers

Initial Report:

On May 5, 1983, Georgia Power Company notified the NRC of a potential deficiency concerning the reactor trip switchgear breakers. In an interim report dated 6/13/83, Georgia Power Company stated a final report would be submitted by September 9, 1983.

Background Information:

In their letter NS-EPR-2744 dated 3/31/83, Westinghouse informed the NRC that operating utility customers had been informed of the potential for intermittent malfunction of the reactor trip switchgear that employs the DS-416 undervoltage (UV) trip device. In this letter, Westinghouse stated that they were requesting additional information from the operating utilities to assist in their evaluation of this unreviewed safety question. Westinghouse also enclosed a table indicating all units under construction or currently operating that would employ the DS-416 breakers. Vogtle Electric Generating Station-Units 1 and 2 was included on this list.

On April 20, 1983, Westinghouse advised the NRC that the potential for misoperation of the DS-416 reactor trip switchgear undervoltage attachments was reportable under 10 CFR 21 and 10 CFR 50.55(e).

Additional information is also contained in I & E Information Notice 83-18, "Failures of the Undervoltage Trip Function of Reactor Trip System Breakers." I & E Bulletin 83-04 was directed to plants using RTS breakers with undervoltage trip devices other than those using Westinghouse DB-type breakers (see I & E Bulletin 83-01). Failures of Westinghouse type DS-416 breakers were experienced during the tests required by Bulletin 83-04. At McGuire, a Westinghouse DS-416 breaker intermittently failed to trip on successive tests.

Engineering Evaluation:

The design of the reactor trip switchgear for the Vogtle Electric Generating Station feeds the automatic protection signals to the undervoltage trip attachment of the reactor trip breakers. The manual signals are fed to the undervoltage trip and to a shunt trip coil of each breaker. The undervoltage trip attachment on the reactor trip switchgear allows for a safe shutdown of the reactor when power to the switchgear is below acceptable limits.

A potential for misoperation exists based on the reported malfunctions. The deviations from the recommended clearances could increase the potential for misoperation of the attachment, thereby creating a condition wherein the reactor trip switchgear might not open on automatic demand from the reactor protection system.

The Westinghouse evaluation also indicated a discrepancy in the design of the retaining ring. The groove in the shaft receiving the retaining ring was not increased in width to be consistent with an earlier (1972) retaining ring design change. The new retaining ring is wider than the original design and does not seat properly in the existing grooves. This discrepancy increases the potential for malfunction of the DS-416 undervoltage attachment thereby creating a condition wherein the reactor trip switchgear might not open on automatic demand from the reactor protection system.

Review for Reportability:

Part 10 CFR 50.55(e) requires the reporting of a deficiency found in design and construction which, were it to have remained uncorrected, could have adversely affected the safety of operation of the nuclear power plant at any time throughout the expected lifetime of the plant. The data in the bulletins and information notice indicated that a breaker problem does exist that could affect plant safety.

This event also represents a significant deficiency in final design as approved and released for construction such that the final design does not conform to the criterion and bases stated in the safety analysis report.

Conclusion:

This concern is reportable under 10 CFR 50.55(e). Westinghouse has already reported the concern to the NRC as a Part 10 CFR 21 through their referenced correspondence to the NRC.

Corrective Action:

(1) Westinghouse has issued a technical bulletin addressing testing procedures for undervoltage and shunt trip mechanisms for model DB and DS circuit breaker shunts and undervoltage coils. Georgia Power Company will incorporate the Westinghouse technical bulletin into plant maintenance procedures.

(2) Westinghouse has committed to replace the undervoltage trip attachments on DS-416 reactor trip switchgear.

(3) Unit 1 breakers have been repaired. Unit 2 breakers are being held at the jobsite pending Westinghouse instructions. Georgia Power Company expects to have the Unit 2 breakers repaired and returned to the jobsite by July 31, 1984

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Vogtle Project

December 12, 1980

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, GA 30302

Attention: Mr. James P. O'Reilly

File: X7BG03-M15
Log: GN-102

Reference: 59-424, 50-425, Westinghouse 3" Gate Valve Closure Problem

Gentlemen:

On November 13, 1980, Mr. Charles Hayes of Georgia Power Company reported to Mr. John Rausch of the NRC a potential deficiency concerning Westinghouse EMD Division manufactured three-inch gate valves Models 3GM88 and 3GM99. The problem associated with the valves was that full stroke was not accomplished during pre-operational testing at a domestic station.

Georgia Power Company is conducting engineering evaluations on the uses of the above type valves in piping systems at Plant Vogtle. These evaluations will determine whether or not a deficiency exists.

At this time, Georgia Power Company cannot commit to a firm date for the final report concerning this problem. We will submit an interim report to the Commission on or before March 3, 1981.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

Doug Dutton
Project General Manager

CWH:tp

xc: (See Page 2)

xc: U. S. Nuclear Regulatory Commission
Attn: Victor Stello, Jr., Director
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mf

W. E. Ehrensperger

March 5, 1981

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, GA 30302

File: X7BG03M15
Log: GN-117

Attention: Mr. James P. O'Reilly

Reference: 59-424; 59-425; Westinghouse 3" Gate Valve Closure Problem

Gentlemen:

On November 13, 1980, Georgia Power Company informed Mr. John Rausch of the NRC of a potential deficiency involving Westinghouse EMD Division 3" gate valves (Models 3GM88 and 3GM99). The valves did not fully close under preoperational test conditions.

A review of the piping systems at Plant Vogtle - Units 1 and 2 revealed the valves were used for:

(1)	PORV Block Valves	HV-8000A,B	RCS	3GM88
(2)	Normal Charging Isolation Valves	HV-8105	CVCS	3GM99
		HV-8106	CVCS	3GM99

Georgia Power Company has also contracted with the vendor for the PORV block valves to be upgraded. Since this proposal was presented to Georgia Power Company prior to having knowledge of this problem, the PORV block valves are being deleted from the review for a potential deficiency and substantial safety hazard.

Georgia Power Company is currently in the final review stage for determining whether or not this problem is reportable under 10CFR50.55(e) and 10CFR21. A final report should be submitted to the Commission on or before April 17, 1981.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon request.

Yours truly,

W. E. Ehrensperger

W. E. Ehrensperger

WEE:tp
xc: (See Page 2)

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Vogtle Project

March 18, 1981

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
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101 Marietta Street
Atlanta, GA 30303

File: X7BG03-M17
Log: GN-119

Attention: Mr. James P. O'Reilly

Reference: 59-424; 59-425 Westinghouse 4" Gate Valve Closure Problem

Gentlemen:

On February 12, 1981 Mr. John Rausch was notified of a potential significant deficiency concerning Westinghouse 4" gate valves, models 46M87 and 46M88. Westinghouse 4" gate valves (above models) are used for the following applications at Plant Vogtle:

(a)	Boron Injection Tank Isolation	HV 8801 A & B
(b)	SI Pump Hot Leg Recirculation Isolation	HV 8802 A & B
(c)	Boron Injection Tank Isolation	HV 8803 A & B
(d)	SI Pump Discharge	HV 8821 A & B
(e)	SI Pump Injection Isolation	HV 8835

Georgia Power Company is conducting an evaluation to determine if any additional 4" gate valves have been purchased for use in other safety systems at Plant Vogtle - Units 1 & 2. Evaluations are being conducted to review the significance of a failure of any of the above valves to close.

Georgia Power Company expects to file an interim report with the Commission on May 1, 1981, concerning the closure problem with the 4" valves.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

Doug Dutton

Doug Dutton
Project General Manager

DED:tp
xc: (See Page 2)

xc: U. S. Nuclear Regulatory Commission
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R. E. Conway
Vice President

Georgia Power
the southern electric system

April 21, 1981

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30302

Attention: Mr. James P. O'Reilly

File: X7BG03-M15
Log: GN-128

Reference: 50-424, 50-425 Westinghouse 3" Gate Valve Closure Problem
Reference Interim Report GN-117, dated 3/5/81

Gentlemen:

Georgia Power Company has reviewed the problem concerning the failure of Westinghouse 3" gate valves to fully close under certain conditions. Based on the possibility that the safety analysis of the plant could have been invalidated by the failure of the valves to close, Georgia Power Company has decided this represents a reportable deficiency and a reportable safety hazard. Enclosed is our final evaluation for this problem.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

R. E. Conway

R. E. Conway

CWH:sew

Enclosure

xc: (see page two)

Georgia Power

xc: U. S. Nuclear Regulatory Commission
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EVALUATION FOR A POTENTIAL DEFICIENCY
WESTINGHOUSE 3" GATE VALVE CLOSURE PROBLEM

Initial Report:

On November 13, 1980, Georgia Power Company informed Mr. John Rausch of the NRC of a potential deficiency involving Westinghouse EMD Division 3" gate valves (models 3GM78 and 3GM88). Some valves undergoing preoperational tests failed to fully close.

Background:

On October 29, 1980, Westinghouse verbally advised Georgia Power Company that during preoperational testing at a domestic station and at a foreign station, problems were encountered when testing the Westinghouse Electro-Mechanical Division manufactured 3" gate valves, Class 1 Model 3GM88, 1500 lb. Westinghouse furnishes this valve to its customers in Class 1, 2, and 3 applications. In the Class 2 and 3 applications the Identification Number only is changed to GM78 and GM58, respectively. The tested valves failed to completely close under preoperational test conditions (i.e., approximately 2700 psi as flow approaches zero) which is less severe than the equipment specification design conditions (i.e., 2750 psi as flow approaches zero). The valves stroked to significantly restrict flow (87%), but the full stroke was not accomplished to trip the "closed" position indication contacts in the motor operator or to seat the valve line within the valve body. Westinghouse also advised the later redesign version of 3GM88, that is 3GM99, may also be subject to the same problems.

Engineering Evaluation:

A detailed plant review for Plant Vogtle-Units 1 and 2 indicated that some of the above Model 3GM88 valves were to be used in piping systems. These systems and valves were:

Model	Tag No.	System	Function
3GM88	HV-8000A	RCS	PORV Block Valve
3GM88	HV-8000B	RCS	PORV Block Valve
3GM78	HV-8105	CVCS	Normal Charging Isolation Valve
3GM78	HV-8106	CVCS	Normal Charging Isolation Valve

(Note: 3GM78 indicates a 3GM88 valve used in a Class 2 application. Also, HV-8105 and HV-8106 have a functional requirement to close against 2750 psi.)

The consequences of incomplete closures of the block valves have been reviewed, and it has been determined that the failure of a PORV to close (and assuming no block valve) is an analyzed condition as detailed in WCAP-9600. Georgia Power Company has accepted the cold shutdown modification program. This program will assure that the upgraded replacement valves for the PORV block valves will be designed in accordance with the Westinghouse functional requirements.

The failure of valves HV-8105 and HV-8106 to close could invalidate the safety analysis of the plant since the analysis was made assuming these valves fully close.

Reportability:

Since the failure of the charging line isolation valves to fully close could invalidate the plant safety analysis, it can be concluded that the safety of operation of the plant could have been affected adversely at any time throughout the expected lifetime of the plant.

Additionally, the problem represents a significant deficiency from performance specifications which required extensive evaluation and redesign to establish the adequacy of the valves to perform their intended safety function.

It can be concluded that failure of the valves to fully close constitutes a reportable deficiency and a reportable safety hazard.

Corrective Action:

Modifications for the 3" valves have been qualified by testing by Westinghouse. These modifications are as follows:

For Model 3GM88 (note 3GM78 is Class 2 application of 3GM88valve)

(1) Change the operator gear ratio to insure 80% voltage closing capability.

(2) Implement limit closing control utilizing the capabilities of spring compensation on the Limitorque SB-00 operators.

The engineering valves test program was comprised of a series of flow tests performed in a hydraulic test laboratory using a centrifugal charging pump to provide flow and pressure. These tests consisted of 75 to 100 closing and opening cycles against flows and pressures as high as 600 gpm and 2600 psid, respectively. These cycles provided the expected valve seat to disc wear, which resulted in stabilized valve closing loads. Once the stabilized level was reached, closing tests were run at lower flows and differential pressures to determine the extent of the closure problem.

Georgia Power Company will correct the valves by following the above instructions, and the corrective action will be completed by December 31, 1981.

Summary:

Based on the possibility that the safety analysis of the plant could have been invalidated, the problem with Westinghouse 3" gate valves to fully close has been considered a reportable deficiency and a reportable safety hazard.

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Vogtle Project

May 1, 1981

United States Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

Attention: Mr. James P. O'Reilly

File: X7BG03 - M17
Log: GN-131

Reference: 50-424; 50-425 Westinghouse 4" Gate Valve Closure Problem
Reference GN-119, 3/18/81

Gentlemen:

In our March 19, 1981, letter concerning the 4" Gate Valve closure problem, Georgia Power Company stated an evaluation was being conducted to determine if any additional 4" Gate Valves had been purchased. Our list in the letter of March 18, 1981, is complete.

Georgia Power Company has been advised by Westinghouse that additional information concerning system reviews and corrective actions will be sent by generic letters in May to various utilities. Based on this information, Georgia Power Company expects to submit a final response by June 12, 1981.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

Doug Dutton
Project General Manager

CWH:skf

xc: (see page two)

xc: U.S.Nuclear Regulatory Commission
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Vogtle Project

July 2, 1981

United States Regulatory Commission
Office of Inspection and Enforcement
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Attention: Mr. James P. O'Reilly

File: X7BG03-M17
Log: GN-136

Reference: 50-424; 50-425 Westinghouse 4" Gate Valve Closure Problem Part 10CFR50.55(e) Evaluation and Response to I.E. Bulletin 81-02.

Gentlemen:

Further to our March 19, 1981 and May 1, 1981 letters, enclosed is Georgia Power Company's evaluation of the 4" gate valve closure problem. This report contains the information required by Parts 10CFR50.55(e) and 10CFR21 and I.E. Bulletin 81-02.

Westinghouse has stated the failure of the valve to close would invalidate the safety analysis, but has been unable to develop plant specific consequences as required by the bulletin.

Georgia Power Company considers these valves a reportable deficiency and a reportable safety hazard. Georgia Power Company submitted an evaluation for Westinghouse 3" valves on April 21, 1981.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

Doug Dutton
Project General Manager

CWH/skr

Attachments

xc: U.S. Nuclear Regulatory Commission
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EVALUATION FOR POTENTIAL DEFICIENCY

Initial Report:

On February 12, 1981, Mr. John Rausch, NRC Inspector, was notified of a potential deficiency concerning Westinghouse 4" gate valves.

Background:

The EMD manufactured gate valve closing problem first surfaced when several valves failed to fully close against high flow and high differential pressure conditions during plant start-up testing at the Almaraz Nuclear Station in Spain and during the EPRI PORV block valve tests at Duke Power's Marshall Station. Typically, these valves operated through 75 percent of their full disc travel leaving about 5 percent of the flow passage unsealed. Subsequent strain gage testing showed the stem thrust loads required to fully close the valve to be 50 percent larger than original design calculations predicted.

In these original design calculations, the stem thrust required to fully seat the wedge or disc against flow was calculated using what were considered valve industry standard equations which are part of the actuator manufacturer's sizing criteria. This calculated required stem thrust load was then used to size the Limitorque motor operator that powers the valve. Since the strain gage testing showed the actual required thrust load to be 50 percent higher than the calculated load which was used to size the operators, the operators do not have the capacity to fully close the valve particularly under their difficult 90 percent voltage requirement.

To determine the source of the higher than expected valve closing loads, a series of analyses and tests were initiated. The testing programs as listed on Attachment 1 include: full flow testing, mechanical fixture testing, seat friction factor testing, and motor operator testing. Kinematic, force, flow, deflection, and stress analyses have been performed. Although this work is not totally complete, it indicates that higher than expected seat friction forces are the source of the majority of the excess load. Backseat friction reactions and stem-disc connection reactions also contribute.

Design-wise little can be done to reduce the actual required loads so attention must focus on increasing the force capabilities of the valve actuators. Depending on the valve being considered, the force capability can be increased by some combination of gearing changes, motor size changes, and operator frame size changes. In some valve sizes these changes have potential for overstressing valve parts so other changes to the valve may also be required.

Engineering Evaluation:

The failure of the valves to close could invalidate the Westinghouse safety analysis. The valves affected are:

Valve Function	Valve Location Number	WEMD Model Reference
Boron Injection Tank Isolation	8801A 8801B	4GM88 4GM88
SI Pump Hot Leg Recirculation Isolation	8802A 8802B	4GM88 4GM88
Boron Injection Tank Isolation	8803A 8803B	4GM88 4GM88
SI Pump Discharge x O Isolation	8821A 8821B	4GM87 4GM87
SI Pump CC Injection Isolation	8835	4GM88

Reportability:

Since the failure of the valves to close could invalidate the Westinghouse safety analysis, these valves represent a deficiency found in the design and construction which, were it to have remained uncorrected, could have adversely affected the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant.

The 4" gate valve closure problem also represents a significant deviation from performance specifications which will require extensive evaluation and repair to establish the adequacy of these components to perform their intended safety function.

Corrective Action:

Westinghouse is currently conducting flow tests to determine the modifications necessary to close the valves. Note that in the background discussion, the requirement for closing against the original equipment specification is extremely conservative and only a small percentage of the valves will be made capable of closing against the original equipment specification. Westinghouse has furnished the following recommended corrective actions. Georgia Power Company will modify the valves to these requirements or to other requirements which are developed by Westinghouse through the 4" valve testing program. Modifications are expected to be completed by March 1982.

Model 4GM87 (Tentative):

1. Change operator gear ratio to guarantee adequate thrust capability at 80 percent voltage.

2. Rewire the operator for limit closing control.

Model 4GM88 (Tentative):

1. Change to Limitorque SB-0-25 motor operator.

2. Rewire the operator for limit closing control.

3. Change to new yoke that will accept an SB-0-25.

4. Change valve internals to accept higher operator stall loads.
   a. New stem (same size: 1.25" Ø)
   b. New stem disc connection
   c. Remachined valve disc to accept new stem-disc connection parts

"ACTIVE" VALVE APPLICATIONS

VALVE FUNCTION	VALVE LOCATION NUMBER	WEMD MODEL REFERENCE	MAXIMUM ΔP (PSID) AS FLOW APPROACHES ZERO		ΔP (PSID) BELOW WHICH VALVE WILL CLOSE (AS SHIPPED)	VALVE CAPABILITY SATISFIES FUNCTION REQUIREMENT (AS SHIPPED)
			EQUIPMENT SPECIFICATION	FUNCTIONAL REQUIREMENT (1)		
Boron Injection Tank Isolation	8801 A & B	4GM88	2750	1200	750	No
SI Pump Hot Leg Recirculation Isolation	8802 A & B	4GM88	2750	1200	750	No
Boron Injection Tank Isolation	8803 A & B	4GM88	2750	1200	750	No
SI Pump Discharge x O Isolation (cross over)	8821 A & B	4GM87	1500	1500	750	No
SI Pump CC Injection Isolation	8835	4GM88	2750	1200	750	No

NOTE:

(1) Functional requirements under final review.

Attachment 21

Georgia Power Company
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Telephone 404 526-6526

Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202
Telephone 205 870-6011

Georgia Power
Southern Company Services

D. O. Foster
Vice President and General Manager
Vogtle Project

January 24, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M37
Log: GN-211

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424; 50-425; Reliance Electric Company - Cable Termination

Attention: Mr. James P. O'Reilly

Gentlemen:

On December 23, 1982, Mr. E. D. Groover of Georgia Power Company reported a potential deficiency concerning the above referenced subject to Mr. V. L. Brownlee of the US NRC.

Georgia Power Company is obtaining the information necessary to complete our evaluation to determine the reportability of this item. Based upon our latest schedule, Georgia Power Company expects to notify the NRC of the results of this evaluation by May 6, 1983.

This response contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

CWH/DOF/tlp

xc: U. S. Nuclear Regulatory Commission
Attn: Victor J. Stello, Jr., Director
Office of Inspection and Enforcement
Washington, D.C. 20555

R. J. Kelly	D. L. McCrary	M. Malcom
R. E. Conway	R. A. Thomas	C. R. Miles, Jr.
G. F. Head	J. A. Bailey	M. Manry
J. T. Beckham, Jr.	O. Batum	P. D. Rice
J. H. Boykin	H. H. Gregory, III	R. W. Staffa
D. E. Dutton	E. D. Groover	J. L. Vota
R. H. Pinson	L. T. Gucwa	W. F. Sanders

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Southern Company Services

D. O. Foster
Vice President and General Manager
Vogtle Project

May 6, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M37
Log: GN-227

Attention: Mr. James P. O'Reilly

Reference: Vogtle Electric Generating Plant - Units 1 and 2
50-424; 50-425 Reliance Electric - Cable Terminations

Gentlemen:

Georgia Power Company has been informed by the architectural engineer that the engineering evaluation for the above referenced subject will not be completed until June 1983. Based on this information, Georgia Power Company now expects to submit a final report to the NRC on the above referenced subject by July 29, 1983.

This response contains no proprietary information and may be placed in the Public Document Room upon receipt.

Very truly yours,

D. O. Foster

D. O. Foster

DOF/CWH/jr

xc: U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

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Southern Company Services

D. O. Foster
Vice President and General Manager
Vogtle Project

July 28, 1983

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

File: X7BG03-M37
Log: GN-245

Reference: Vogtle Electric Generating Plant-Units 1 and 2, 50-424, 50-425; Reliance Electric-Cable Terminations

Attention: Mr. James P. O'Reilly

Gentlemen:

Georgia Power Company has concluded its evaluation to determine if a reportable occurrence does exist. The evaluation has concluded that the event is reportable under 10 CFR 50.55(e) and 10 CFR 21. According to NRC guidance on the subject of duplicate reporting, Georgia Power Company is reporting this event under 10 CFR 50.55(e). Enclosed is our evaluation for this event.

This report contains no proprietary information and may be placed in the NRC Public Document Room upon receipt.

Yours truly,

D. O. Foster

D. O. Foster

DOF/CWH/cc

enclosure

xc: U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

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E. D. Groover
L. T. Gucwa
M. Malcom
G. Bockhold
P. D. Rice
J. L. Vota

EVALUATION FOR A SUBSTANTIAL SAFETY HAZARD
EVALUATION FOR A SIGNIFICANT DEFICIENCY

Reliance Electric Cable Terminations

Initial Report:

On December 23, 1982, Mr. E. D. Groover of Georgia Power reported a potential deficiency to Mr. Virgil Brownlee of the USNRC. The deficiency concerned cable terminations in panels and cabinets received from Reliance Electric.

Background Information:

Georgia Power Company (GPC) identified the following potential problems concerning the panels delivered to GPC by Reliance Electric:

(1) Connectors crimped with tools other than manufacturer of connector.

(2) Manufacturer of connector may not certify terminations.

(3) Uninsulated versus insulated connections installed.

(4) Halogenated compounds used in conflict with the design manual.

(5) End of conductor not flush with or extends beyond connector barrel, as required by specification.

A meeting was held at the jobsite on January 12 and 13, 1983, between engineering and Reliance Electric to review the above referenced audit findings and to perform an inspection of the cited problems. Reliance Electric manufactured panels for the following specifications:

X3AE01 - Q	Electrical Auxiliary Boards
X3AE02 - NON Q	Protective Relay Panels
X3AE04 - NOM Q	Miscellaneous Control Panels
X3AE06 - Q	Isolation Device Panels
X3AE08 - Q	Auxiliary Relay Panels
X3AE09 - NON Q	Fault Recorder Panels
X3AE10 - NON Q	Underfrequency Panels
X3AH05 - Q	Junction Boxes
X3AB01 - Q and NON Q	Mixcellaneous Control Panels

Cable and wire terminations inside Class 1E control panels and cabinets are an integral part of the electrical circuits which have been designed to

perform safety control functions as well as to provide necessary indications of various safety systems.

Engineering Evaluation:

A. A review of the Audit Finding Report was performed by engineering to assess the potential deficiencies with respect to project criteria and applicable procurement specifications. The following conclusions were established:

a. The applicable procurement specifications provide general criteria and installation requirements for wiring inside panels and cabinets.

The specification requires that the vendor utilize the proper crimping tools for the application of wire connectors when terminating. It is the responsibility of the supplier to utilize the proper crimping tools and to maintain compliance to his in-house quality control procedures.

Reliance's in-house quality program includes procedures for cable terminations, inspection and tool calibration. Their tool calibration program references conformance to UL 486A "Wire Connectors and Soldering Lugs" as the basis for the application of the proper tool. Standard UL 486A delineates the acceptance criteria (by testing) for an acceptable connection of a cable conductor (wire) to a connector.

b. All terminations in Class 1E panels were made using insulated lug barrels.

c. The procurement specifications prohibit the use of polyvinylchloride (PVC) or other halogenated compounds in the insulation of cables or connectors. However, the project design criteria allow limited usage of such materials outside the reactor containment building on a case-by-case basis. Deviations from the specifications require the vendors to submit supplier Document Deviation Requests (SDDRs) quantifying the amount of PVC utilized. Engineering assessment and disposition of the SDDR is required prior to shipment. SDDRs for the cited panels were not submitted by the vendor prior to shipment.

B. Based on engineering's inspection of all Reliance Class-1E panel terminations, the following observations and conclusions were made:

a. The majority of terminations appear to have been applied correctly with good workmanship.

b. Several panels contained terminations where crimps on the lug (connector) barrel were misaligned along the axis causing:

- Partial crimping of he wires at the tongue end of the barrel.

- Compression of lugs ı the sides of the barrel rather than top or bottom.

- Distortion of lug insulation due to misaligned tool application.
- Wires extending too far past the tongue end of the barrel.
- Wire strands were cut as they entered the lug (minimal quantity noticed)

The above do not meet the specification requirements or Reliance's in-house procedures for cable terminations. They are considered to be due to poor workmanship and not a result of the type of tool used.

c. Minimal (less than four ounces per panel) use of PVC was observed in Class 1E panels. The application was found to exist only on the insulation of lug (connector) barrels.

d. Uninsulated lugs were found to be used on power supply circuits inside several non-Class 1E panels. This however is considered to be an acceptable application.

Review of Reportability Requirements:

Part 10 CFR 50.55(e) requires the holder of the Construction Permit to notify the Commission of each deficiency found in the design and construction which, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant. Also, the deficiency must represent either:

(1) a significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements, or

(2) a significant deficiency in the construction of a component which will require extensive repair to meet the criteria and bases stated in the Safety Analysis Report

Cable and wire terminations inside Class 1E control panels and cabinets are an integral part of electrical circuits which have been designed to perform safety control functions as well as provide necessary indication of various safety systems. Although no terminations were found to have loose connections, the potential for a break in electrical continuity at an improperly crimped connector does exist. Therefore, the safety of operations could have been adversely affected.

This event also represents a significant deficiency in the construction of safety-related components due to the large number of connections that were reworked.

Additionally, a review was made to determine if there was a significant breakdown in the quality assurance program of Reliance Electric. This review concluded that the concern with "the end of the conductor not flush with or extending beyond the connector board" was the result of poor workmanship and should have been detected by inspection personnel employed by Reliance Electric. This indicates a breakdown in the quality assurance program with regard to the following:

(1) Criteria II - Quality Assurance Program - Training of inspection personnel

(2) Criteria V - Instruction, Procedures and Drawings - Inadequate or non-existence of procedures which include appropriate acceptance criteria

(3) Criteria VI - Document Control - Distribution to and use of inspection procedures at locations where needed.

It should be noted that Bechtel's Procurement Supplier Quality Department conducted an audit of Reliance on September 29 through October 1, 1982. Although this audit was prior to the identification of these workmanship deficiencies, it was a verification of the quality program in effect at the time the deficiencies occurred. This audit, which noted several discrepancies with each of the above criteria, resulted in a stop work notice.

Major degradation is considered to be a loss of redundancy if in conjunction with a single-failure, a required safety function could not be performed. Due to the large number of questionable wiring connections, required safety function may not have been able to be performed. Thus, this also represe a substantial safety hazard.

Conclusion:

This event represents a significant deficiency and substantial safety hazard, and is reportable under Part 10 CFR 21 and Part 10 CFR 50.55(e).

Corrective Action:

The following corrective actions were taken to resolve the deficiencies and poor workmanship on terminations in all Reliance panels at the Vogtle jobsite.

1. Deviation reports were generated at the jobsite to disposition the minimal use of PVC as permitted by the project design criteria.

2. An acceptance criteria on terminations was developed by engineering and given to Reliance for ther use in reworking all terminations which exhibited conditions as described in 6.0 B.b above. Rework of all unacceptable terminations (per criteria) by Reliance was completed on April 21, 1983.

3. Reliance was requested to certify through sample testing (by an independent qualified facility) that crimps made using tools from manufacturers different from lug the manufacturers comply with UL 486A as referenced in their calibration procedures. Tests were performed by Underwriters Laboratories for Reliance and a report submitted to them on April 26, 1983. Engineering's evaluation of the report concluded that compliance to UL 486A was satisfied.

4. Reliance was directed by engineering that all future terminations made in their facilities shall be performed using calibrated tools manufactured by the same vendor as for the connectors. Crimps made with crimping tools other than those made with the connector manufacturers' recommended tools shall be certified for compliance to standard UL 486A (following acceptance by independent laboratory) testing of Reliance crimping methods. Reliance has subsequently decided to utilize only tools recommended by the connector manufacturer.

5. Subsequently, Reliance has extensively revised their QA Manual (Issue 4, revision 1) which Bechtel reviewed and approved on November 16, 1982. Bechtel's Procurement Supplier Quality Department audit of December 10, 1982, verified adequate implementation of the program and procedures with regard to each of the above criteria.

Attachment 22

BBC
BROWN BOVERI

Brown Boveri

Switchgear Products Division

BBC Brown Boveri, Inc.
Rts. 309 & Norristown Road
Spring House, PA 19477
Phone: (215) 628-7400

March 19, 1984

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: HK Control Device
(10 CFR Part 21 Report) -

Gentlemen:

On March 22, 1983 Brown Boveri issued a report to the NRC on a potentially reportable deficiency of a broken contact carrier in the control device of the 7.5 KV through 15 KV, 500 through 750 MVA; 5 KV 350-3000 and all HKV circuit breakers. This report was a follow-up to a 10CFR50.55(e) report filed by the W. H. Zimmer Power Station on February 23, 1983.

The report and its enclosed IB-8303 were specifically issued for circuit breakers manufactured between March 1974 and July 1978.

Recently a non-nuclear customer reported a similar condition of a limit switch breaking in a 15HK750 circuit breaker manufactured after July 1978. No reports of this condition have been received from any nuclear station.

On the circuit breakers manufactured between March 1974 and July 1978 the breaking of the contact carrier could be caused by the overtravel of the lever due to the location of the lever stop. Instructions for lever stop adjustments were issued in IB-8303.

An evaluation of this most recent condition, wherein the contact carrier was broken shows that the breakage was caused by the gaps between the lever and lever stop and the limit switch crank and lever adjusting screw not being adjusted properly in the factory. Proper adjustment of the gap setting corrected this condition.

Steps have been taken to assure that proper procedures are being followed. Instruction of assembly and inspection personnel has been conducted and an audit of these procedures is being accomplished periodically.

The gap setting between the lever stop and lever had always been considered to be a factory adjustment and was not included in the General Instruction Books IB 6.2.2.7-1 (8.2.7-4) and IB 6.2.1.7 (8.2.7-3).

BBC
BROWN BOVERI

Document Control Desk
March 19, 1984
Page 2

Brown Boveri is sending addendums to these instruction books to all Nuclear Stations where 7.5HK, 15HK and 5HK350-3000 circuit breakers have been delivered after July 1978. It is recommended that the settings for the control relay be checked and adjusted if required, during checkout or at the next maintenance period.

The instruction book addendums are included as enclosures to this letter. (Exhibits A & B). Also included is a list of Nuclear stations where the 7.5HK, 15HK and 5HK350-3000 circuit breakers have been delivered after July 1978. (Exhibit C). Not all of these circuit breakers are used in Class 1E applications. The HKV circuit breakers have not been utilized in Class 1E applications.

A copy of the Brown Boveri report to the NRC of March 22, 1983 (Exhibit D) is enclosed to facilitate your review of this matter and to communicate to users the difference in breakers delivered between March 1974 to July 1978 and subsequent years.

It should be noted that a broken limit switch carrier will probably cause the electrical close to be inoperative, but has no affect on the manual closing and the electrical or mechanical tripping of the circuit breaker.

D. D. DUVALL
Vice President

DDD/jm

Enclosures

cc: W. D. Donaldson
W. E. Laubach
E. W. Rhoads
L. H. Schmidt
J. R. Silverio
W. R. Wilhelm
W. Laudan, NRC

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EXHIBIT "A"

ADDENDUM TO BROWN BOVERI

INSTRUCTION BOOK 6.2.2.7-1 (8.2.7.4)

INSTALLATION AND MAINTENANCE INSTRUCTIONS

TYPE 7.5HK500, 15HK500 AND 15HK750

5.10 CONTROL RELAY ADJUSTMENT (See Figure 7)

The control relay does not normally require any adjustment in the field. However, if necessary, adjust the gap between the control device lever adjusting screw and the limit switch crank arm for a gap no less than 0.06" and no more than 0.09" with the closing springs charged. With the closing springs discharged, the gap between the lever stop and the lever to be 1/64" to 1/16".

NOTE: Figure 7 will have a note to add shims under the lever stop "as required" to adjust gap between lever and lever stop.

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EXHIBIT "B"

ADDENDUM TO BROWN BOVERI

INSTRUCTION BOOK IB 6.2.1-7 (6.2.7-2)

TYPE 5HK 350-3000 ONLY

5.10 CONTROL RELAY ADJUSTMENT
(See Figure 6)

The control relay does not normally require any adjustment in the field. The gap between the control device lever and the limit switch crank should be between 0.06" and 0.09" with the closing springs charged. If the gap measurement is incorrect adjust all 5HK circuit breakers except 5HK350, 3000 by shimming either at the lever bracket or the control device mounting screws as illustrated in Figure 6. Before changing shims at the two control device screws, loosen all three screws, replace shims and then tighten all three screws. The 5HK350, 3000 is adjusted by means of an adjusting head screw fastened to the lever. With the closing springs discharged, the gap between the lever stop and lever to be 1/64" to 1/16".

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EXHIBIT "C"

7.5 HK, 15 HK AND 5HK350-3000A

CIRCUIT BREAKERS

MANUFACTURED AFTER 1978

TAIWAN	KOESHING
DUSQUESNE	BEAVER VALLEY
GEORGIA POWER *	VOGTLE
PHILADELPHIA ELECTRIC	LIMERICK (NON 1E)
PSI	MARBLE HILL
PSNH	SEABROOK (NON 1E)
PSE&G	HOPE CREEK
SCE	SAN ONOFRE
HL&P	SOUTH TEXAS
TVA	SEQUOYAH
TVA	HARTSVILLE/PHIPPS BEND
TVA	YELLOW CREEK
WPPSS	UNITS 1 & 4

BBC BROWN BOVERI

Brown Boveri Electric, Inc.

Manufacturer of I-T-E Electrical Power Equipment

March 22, 1983

EXHIBIT "D"

Mr. Victor Stello, Jr., Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: W. H. Zimmer Nuclear Power Station
10CFR50.55(e) Report Dated February 23, 1983
Brown Boveri Electric Control Device for
Medium Voltage Circuit Breakers

Dear Mr. Stello:

On February 23, 1983, the W. H. Zimmer Nuclear Power Station filed an interim 10CFR50.55(e) Report with Region III of the U. S. Nuclear Regulatory identifying a potentially reportable deficiency of a broken limit switch carrier on the HK control device used on some Brown Boveri Electric HK circuit breakers. This particular control device is used on circuit breaker types 7.5KV through 15KV, 500 through 750MVA; 5HK 350, 3000; and all HKV circuit breakers.

The purpose of this report is to furnish the NRC with supplemental information to the report filed by C G & E and clarify the extent of this potential problem at other Nuclear Power plants.

Enclosed is a copy of the BBE Engineering Analysis of the broken limit switch carrier of this HK control device and a copy of the BBE IB-8303 which provides instructions for inspection and corrective action, if required, for the control device.

Factory Specification records indicate that circuit breakers manufactured between March, 1974 and July, 1978, inclusive, could be subject to this condition. The date of manufacture for the circuit breaker is stamped on the nameplate which is mounted in the lower front right hand corner.

A broken limit switch carrier will probably cause the electrical close to be inoperative, but has no effect on the manual closing and electrical or manual tripping of the circuit breaker.

Normally, the main protective function of the circuit breaker is to provide trip function under all conditions but there may be applications where it is important that the circuit breaker be able to close on demand.

BBC Brown Boveri Electric, Inc.

March 22, 1983
Page 2

Enclosed is a list of Nuclear Power Stations for which these circuit breakers were manufactured during the period of March, 1974 through July, 1978. Also, since a dimension "B" has been added to Figure 2 of the enclosed IB-8303, this information will be transmitted to all Nuclear Power Stations utilizing these circuit breakers. The dimension "B" was previously considered to be a factory check only, however this information will be added to the general Instruction Bulletin, IB 6.2.2.7-1 (8.2.7.4) for the HK circuit breakers.

D. D. DUVALL
Vice President

EWR/jm
Enclosure

cc: W. P. Cooper - CG&E
W. E. Laubach
W. J. Leister
E. W. Rhoads
L. E. Schmidt
J. F. Silverio
C. J. Village

Brown Boveri Electric, Inc.

ENCLOSURE TO 3/22/83 LETTER TO
VICTOR STELLO, JR. - NRC

DETROIT EDISON COMPANY	FERMI
COMMONWEALTH EDISON	LASALLE COUNTY
CONSUMERS POWER	MIDLAND
SOUTHERN CALIFORNIA EDISON	SAN ONOFRE 2 & 3
CINCINNATI GAS & ELECTRIC	ZIMMER
DUKE POWER COMPANY	McGUIRE
TVA	BELLEFONTE

Brown Boveri Electric, Inc.

Manufacturer of I-T-E Electrical Power Equipment

ANALYSIS OF HK CONTROL DEVICE (LIMIT SWITCH) FAILURE
(HK BREAKER TYPES - 7.5KV THRU 15KV, 500 THRU 750 MVA; 5HK350, 3000)
ALL HKV BREAKERS

BACKGROUND

One defective HK control device, BBC part number 191921-T6, was returned from the Cincinnati Gas and Electric Co. to the BBC production plant in Columbia, South Carolina for evaluation. The device was returned on BBC RGA B10112, Material Non-conformance Report 101150 dated 2-10-83.

NATURE OF FAILURE

When the failed device was examined at Columbia, the plastic contact carrier, containing the three limit switch contacts (2-LSb contacts, 1-LSa contact), was found broken. Re: typical HK wiring diagram 196270 attached. The most common reason for breaking the carrier is due to excessive overtravel of the lever. See IB-8303 attached, Fig. 1. This overtravel overdrives the limit switch crank which causes breakage of the carrier. The overtravel of the lever is limited by the lever stop and as long as the dim "B", Fig. 2, is not greater than 1/16 inch, breakage should not occur. Tests conducted in July of 1978 have substantiated this.

AFFECT OF FAILURE ON BREAKER OPERATION

Failure of these contacts can affect the electrical closing function of the breaker, i.e., the closing springs may not charge electrically and if the springs are charged electrically or mechanically, the breaker may not be able to close electrically. The failure of the limit switch contacts does not affect the manual closing of the breaker and does not affect electrical or manual tripping of the breaker.

CORRECTIVE ACTION

Instruction Bulletin - The dim. "B", IB-8303 Fig. 2, has always been considered to be a factory check and therefore the Instruction Bulletin has not addressed dim. "B". However, due to its importance, dim. "B" will be added to the general Instruction Bulletin, IB 6.2.2.7-1 (8.2.7.4).

Breakers in the field - IB-8303 has been issued for the inspection and/or correction of breaker in the field for obtaining dim. "B". This bulletin also checks out and corrects for dim. "A", however dim. "A" has no affect on the overtravel of the contact carrier.

Brown Boveri Electric, Inc.

ANALYSIS OF HK CONTROL DEVICE (LIMIT SWITCH) FAILURE
(HK BREAKER TYPES - 7.5KV THRU 15KV, 500 THRU 750 MVA; 5HK350, 3000)
ALL HKV BREAKERS

Factory specification records indicate that breakers manufactured between March 19
and July 1978, inclusive, are subject to exceeding dim."B" limits and therefore s
be checked per IB-8303.

CONCLUSION

A broken limit switch carrier will probably cause the electrical close to be inoperative but it has no affect on manual closing or electrical or manual tripping. Corrective action should be taken as outlined above.

Prepared by: D. H. Lewis
D. H. Lewis

Date: 2-22-83

Approved by: L. H. Schmidt
L. H. Schmidt

Date: 2-22-83

Attached
IB-8303
196270

HK. WIRING DIAGRAM
TEST CIRCUIT BREAKER

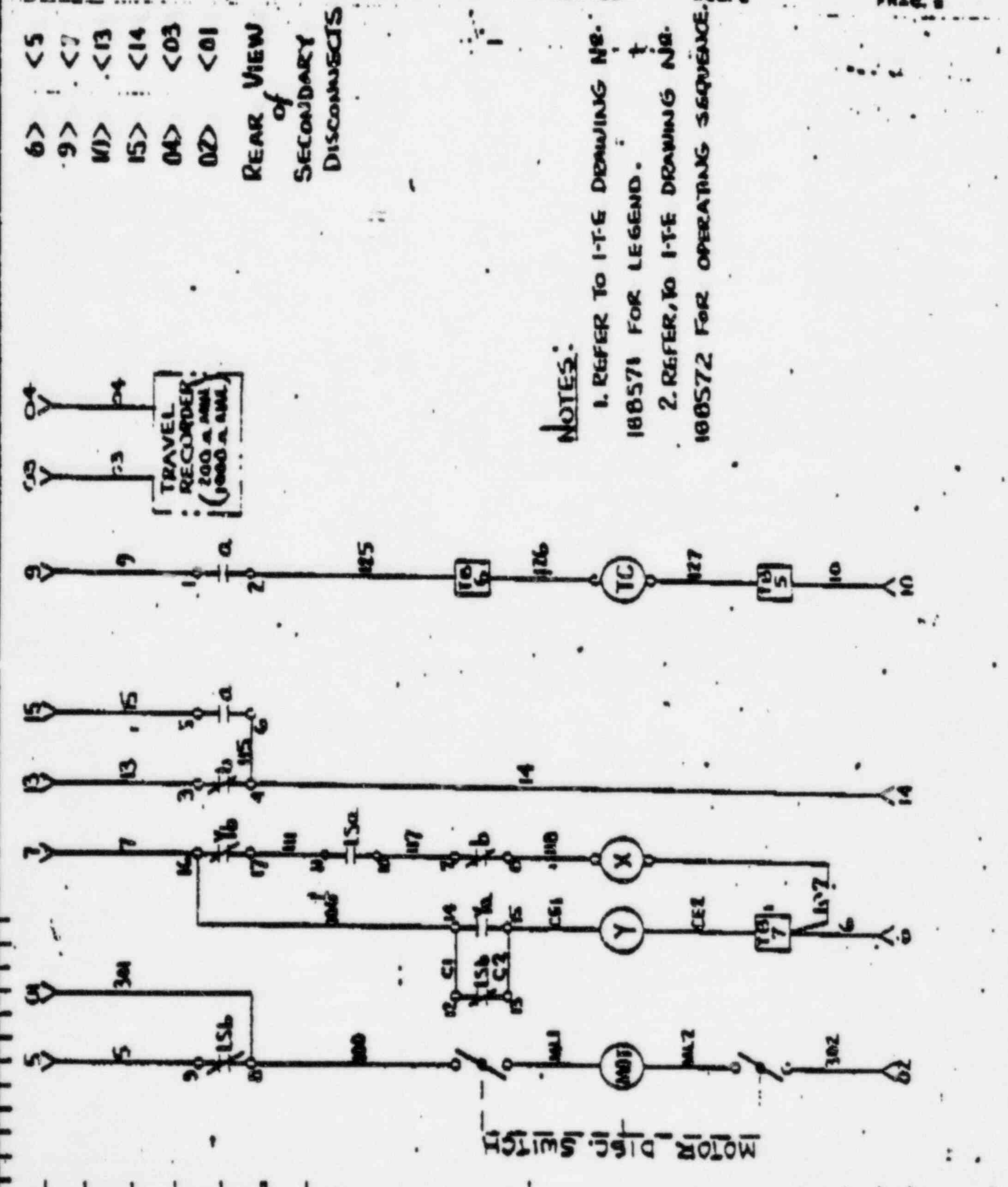
I-T-E CIRCUIT BREAKER COMPANY

196270

SHEET NO. 1 CONT. ON — REV.

NEXT ASSY.	BY	CHD.	APP.	SCALE
S.O. 951-9105	DATE 9-4-64	DATE	DATE	

STANDARD TOLERANCE INFO. ON DR. 32016
TOLERANCES — UNLESS OTHERWISE SPECIFIED — DEC. ± FRAC. ±



BBC
BROWN BOVERI

INSTRUCTIONS FOR INSPECTING, ADJUSTING AND/OR REPLACEMENT OF PARTS FOR OBTAINING LIMIT SWITCH ADJUSTMENTS ON BROWN BOVERI 7.5KV THRU 15KV, 500 THRU 750 MVA; 5HK350, 3000; ALL HKV BREAKERS

IB-8303

BBC
BROWN BOVERI

INSTRUCTIONS FOR INSPECTING, ADJUSTING AND/OR REPLACEMENT OF PARTS FOR OBTAINING LIMIT SWITCH ADJUSTMENTS ON BROWN BOVERI 7.5KV THRU 15KV, 500 THRU 750 MVA; 5HK350, 3000; ALL HKV BREAKERS

(MANUFACTURED BETWEEN MARCH 1974 AND JULY 1978 INCLUSIVE)

1. GENERAL

1.1 For general breaker instruction, refer to the appropriate Brown Boveri Instruction Bulletin.

1.2 The breaker must be removed from the switchboard.

1.3 Unless otherwise stated, the breaker is to remain open with closing springs discharged.

1.4 The reason for a gap at Dim. "A" is to insure the "LSa" contact is made. The gap at Dim. "B" is to insure that the "LSb" contacts are made and to limit the overtravel of the lever so that the limit switch plastic carrier inside the control relay is not broken.

1.5 This Instruction Bulletin supersedes IB-7803.

2. INSPECTION AND ADJUSTMENT FOR DIM. "A" (See Fig. 1)

2.1 Charge the closing springs electrically or manually.

2.2 Check Dim. "A" with feeler or pin gages.

2.3 If required, readjust the screw (4) to Dim. "A" after first loosening the jam nut (5) and then retightening the jam nut after the adjustment.

2.4 Close and open the breaker to discharge the closing springs.

3. INSPECTION AND ADJUSTMENT FOR DIM. "B" (See Fig.'s 1 and 2)

3.1 For this procedure, the breaker is to be open and the closing springs discharged.

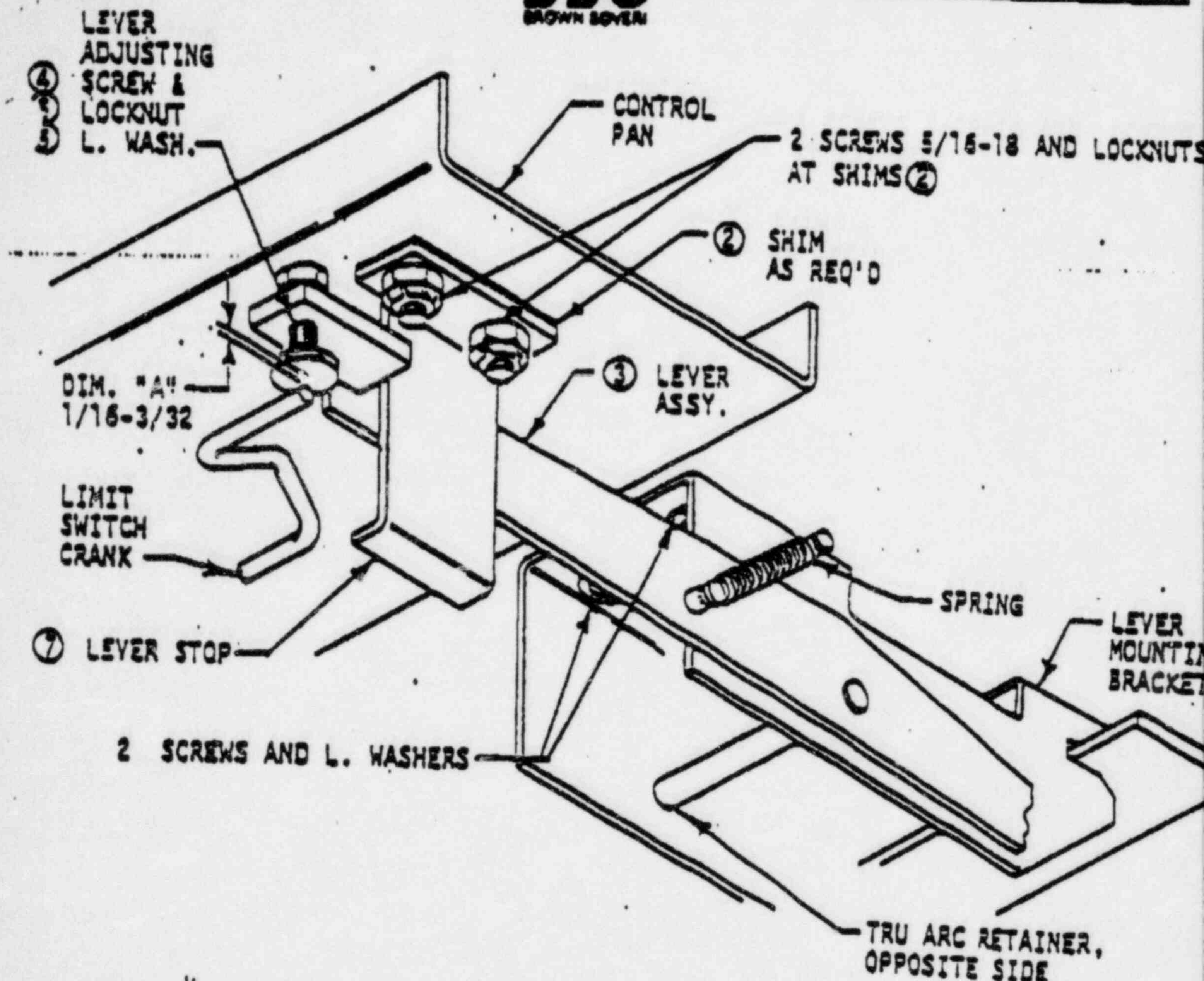
3.2 Check Dim. "B" with feeler or pin gages. (See Fig. 2)

3.3 If Dim. "B" is not within dimensions, it is necessary to shim the lever stop(7) up or down as required. This is done by adding or removing shims(2) at the lever stop. (See Fig. 1) Note that if all shims are removed and the Dim."B" still exceeds the max. allowed, the length of the lever stop should be checked. The lever stops made after July 1978 are 2 17/32 long. Previous to this, the length was 2 25/32. If required, install the shorter lever stop (7).

BBC
BROWN BOVERI

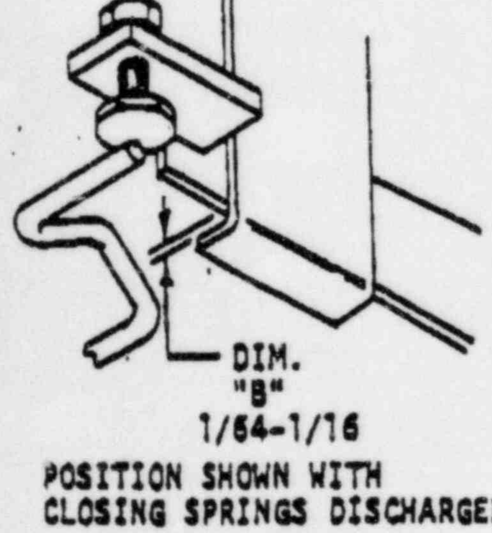
4. NEW PART NUMBERS

	Qty. Per Breaker	Notes
2. Shim 650451-A27	As req'd (3 est.)	
7. Lever stop 193392-A	1	2 17/32 overall leng Used in breakers bui after July 1978.



POSITION SHOWN WITH
CLOSING SPRINGS CHARGED.

FIG. 1



POSITION SHOWN WITH
CLOSING SPRINGS DISCHARGED.

FIG. 2

Attachment 23

V.A.13.C5 83 16

SSINS No.: 6835
In 83-76

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

November 2, 1983

IE INFORMATION NOTICE NO. 83-76: REACTOR TRIP BREAKER MALFUNCTIONS (UNDERVOLTAGE TRIP DEVICES ON GE TYPE AK-2-25 BREAKERS)

Addressees:

All nuclear power reactor facilities holding an operatng license (OL) or construction permit (CP).

Purpose:

To advise licensees of malfunctions of the undervoltage trip attachments in General Electric Type AK-2-25 breakers which are used in safety-related systems. Although no specific action is being required by the NRC at this time, it is expected that licensees will review the general problem described for applicability to their facilities and take appropriate action.

Description of Circumstances:

Five malfunctions of the undervoltage (UV) trip device used on General Electric Type AK-2-25 reactor trip breakers (RTB) occurred at the San Onofre Nuclear Generating Station, Units 2 and 3. On October 28, 1983, with Unit 3 in Mode 3 (Hot Standby), while routine monthly surveillance was being performed on the reactor protection system, the undervoltage armatures for RTBs #5 and #8 were observed to be in a midposition rather than fully down and in contact with the air gap adjusting screw (See Figure 1). The RTBs were reset, observed to operate properly, and the undervoltage armatures were observed to be properly positioned at that time.

On October 31, 1983, with both Units 2 and 3 in Mode 1 (Power Operation), the positions of all the undervoltage armatures of the RTBs were visually inspected. This inspection revealed that undervoltage armatures of RTB #4 on Unit 2 and RTBs #5 and #8 on Unit 3 were not properly positioned. It has been established that the Unit 3 breakers had been tripped and reset subsequent to the October 28, 1983 event without further UV armature position verification. These RTBs were reset, observed to operate properly and the undervoltage armatures remained properly positioned.

Southern California Edison Company, the licensee, has indicated that these malfunctions (i.e., the undervoltage armature in midposition rather than in contact with the air gap adjusting screw) could result in the RTB failing to trip within the specified criteria of the undervoltage trip device. Southern California Edison Company's preliminary conclusion is that the undervoltage armature can remain in a midposition as a result of interference (vertical)

between the undervoltage armature and the copper shading ring around the core of the coil (See Figure 1).

No action or written response to this notice is required; however, licensees using AK-2 type breakers with undervoltage trip devices may find it prudent to visually inspect each undervoltage armature to assure it is in its proper position after each operation. If the UV armature were to be found in an improper position, the NRC would consider the RTB to be inoperable. If you have any questions regarding this notice, please contact the Regional Administrator of the appropriate NRC Regional Office, or a Technical Contact listed below.

E. L. Jordan

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contacts: I. Villalva, IE
(301) 492-9635

J. T. Beard, NRR
(301) 492-7465

Attachment:
1. Figure 1
2. List of Recently Issued IE Information Notices

Attachment 1
IN 83-76
November 2, 1983

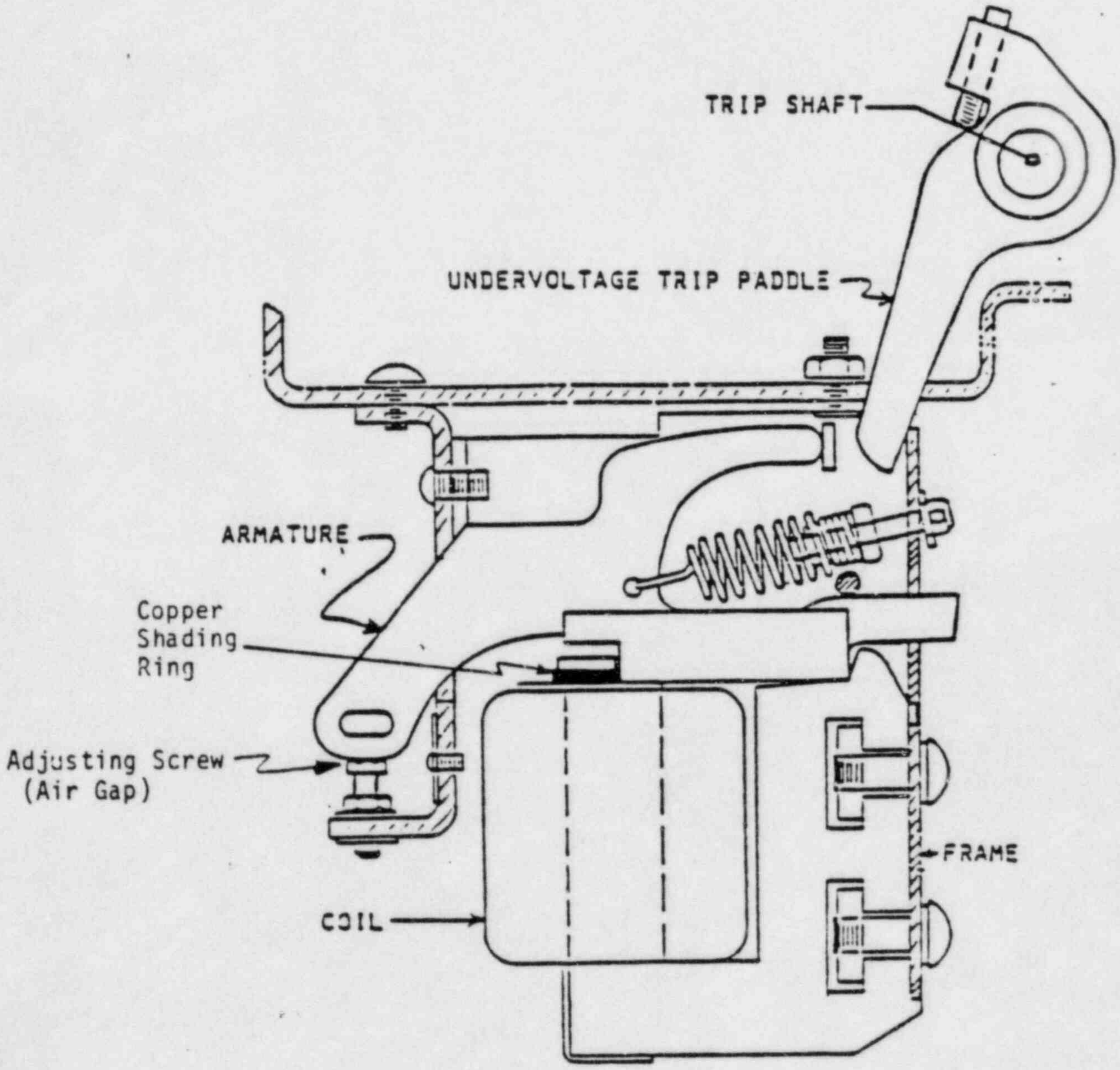


Figure 1. Undervoltage Trip Device, Coil Energized Position

Attachment 24

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

X76610
Orig: RMS
Copy: m/E, ~~DEF, ...~~

SEP 2 0 1982

Docket Nos.: 50-424
and 50-425

PROJECT MANAGEMENT

Mr. Doug E. Dutton
Vice President - Generating Plant Projects
Georgia Power Company
Post Office Box 4545
Atlanta, Georgia 30302

Dear Mr. Dutton:

Subject: Controls for Field Change Notices - Request for Additional Information
Alvin W. Vogtle Nuclear Plant, Units 1 and 2

On October 15, 1981, after conference calls with NRC Headquarters and Region II and IV and at the request of the NRC, Georgia Power Company (GPC) submitted a description of the Field Change Notice (FCN) modification of the Vogtle PSAR and the related controls. The normal method of handling field changes by the Field Change Request (FCR) system and the FCN system was described as follows in the October 15, 1981 GPC submittal:

"Georgia Power Company's (GPC) normal method for site originated changes to approved design documents is the Field Change Request (FCR). The FCR is transmitted to Engineering and no work relating to the change is authorized until the FCR is approved. The change requested by an FCR, if approved, is incorporated into the design by a design change notice (DCN) which is forwarded to GPC for its use. The provisions of the DCN are than routinely incorporated into other approved design documents as they are revised. This method will continue to be the preferred method for implementing field originated changes."

"The Field Change Notice (FCN) will be used only when GPC must begin implementing a change prior to final Engineering approval. The construction site field procedures establish the required approval for FCN's. Only individuals (Assistant Project Section Supervisor level or above) authorized in writing by the Manager of Field Operations may approve FCN's. Authorized personnel may only approve FCN's within their respective discipline. These authorized individuals are experienced in nuclear power plant construction engineering. They are knowledgable with and have access to the Engineering criteria and procedures.

"Once the FCN is approved by GPC it is telecopied to the Engineering home office for review and approval. FCN's are controlled and tracked by use of logs at both the site and the Engineering office. Engineering is to notify GPC within five (working) days after issue if their review indicated the FCN is not acceptable. In such cases GPC will place the affected work on hold as required by our nonconformance report (NCR) procedure. Acceptable FCN's are incorporated into the design by a design change notice (DCN) within five (working) days after receipt."

Checks and audits of the system, which has been in use for approximately one year, are made by GPC's QA/QC organization.

Concern has been expressed within NRC regarding GPC's FCN system. One concern is that because the revised system permits construction to proceed prior to final engineering approval, it can lead to implementation of marginally acceptable design changes and general abuse of the process. A second concern is that the FCN system may be contrary to the following requirements of 10 CFR 50 Appendix B:

a) Criterion VI. Document control measures "shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization."

b) Criterion V. "Activities affecting quality shall be prescribed by document instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings."

c) Criterion III. "Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization."

Because of the differences in viewpoint regarding the acceptability of GPC's FCN system, a meeting was held in Bethesda on August 3, 1982 to discuss these concerns. The meeting was attended by representatives of NRR (LB#1 and QAB), IE (QAB), OELD, and NRC Regions II and IV.

As a result of the August 3rd meeting, we find that additional information is required to permit the staff to further assess the acceptability of GPC's FCN system. Therefore, we request your response to the enclosed questions by October 20, 1982.

Please contact J. Grant, Project Manager, at 301-492-7793 if you have any questions concerning the enclosed request.

for L L Kintner
B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure:
Request for Additional Information

cc: See next page

Mr. Doug Dutton
Vice President - Project
Management
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

cc: Mr. L. T. Gucwa
Chief Nuclear Engineer
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Mr. Ruble A. Thomas
Vice President
Southern Services, Inc.
P. O. Box 2625
Birmingham, Alabama 35202

Mr. J. A. Bailey
Project Licensing Manager
Southern Company Services, Inc.
P. O. Box 2625
Birmingham, Alabama 35202

George F. Trowbridge, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. D. O. Foster
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Mr. James P. O'Reilly
Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. William S. Sanders
Resident Inspector/Nuclear Regulatory Commission
Post Office Box 572
Waynesboro, Georgia 30830

Request for Additional Information
Georgia Power Company - Vogtle
Field Change Notices

260 Quality Assurance Branch

260.1 Discuss how GPC's FCN system satisfies the following requirements of Appendix B to 10 CFR Part 50:

a) Criterion VI. Document control measures "shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization."

b) Criterion V. "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings."

c) Criterion III. "Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization."

260.2 Describe the criteria established by GPC for restricting the types of design changes that can be approved under the field change notice (FCN) system.

260.3 Describe in more detail the criteria for Bechtel notifying GPC of acceptable FCNs and unacceptable FCNs.

260.4 Provide a detailed summary of the pertinent experience under the field change request (FCR) method and the FCN method of handling field changes since initiating the FCN method.

260.5 Describe the GPC organizational arrangement at the site including the position, responsibilities, and authority of the individuals authorized to approve FCNs, i.e., Assistant Project Section Supervisor level or above. Describe the extent that these individuals are isolated from the pressures of construction schedules and cost.

260.6 Describe in greater detail the qualifications of the individuals at the Assistant Project Section Supervisor level or above who are authorized to approve FCNs.

260.7 Discuss the advisability of GPC individuals authorized to approve FCNs contacting Bechtel Project Engineering in Los Angeles by telephone/telefax to obtain preliminary approval before implementing FCNs, of using Bechtel on-site engineering personnel to assist or perform the FCN activities currently performed by GPC, or of Bechtel Project Engineering being represented on-site to the extent that they can accomodate field changes in an expeditious manner.

Attachment 25

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 29 1983

Docket No. 50-424/425

MEMORANDUM FOR: Assigned Reviewers for Vogtle Electric Generating Plant

FROM: James G. Partlow, Acting Director
Division of Quality Assurance,
Safeguards, and Inspection Programs, IE

M. A. Miller, Project Manager
Licensing Branch No. 4
Division of Licensing, NRR

SUBJECT: DETERMINATION OF ACCEPTABILITY OF LIST OF VOGTLE ELECTRIC GENERATING PLANT ITEMS UNDER QA PROGRAM

This memorandum is written to request that all reviewers assigned to the Vogtle Electric Generating Plant FSAR application review the information given in the FSAR Table 3.2.2-1 to determine if it is an adequate list of items that should be controlled under the Appendix B quality assurance (QA) program.

The criterion to be used in determining whether items should be controlled by requirements of the Appendix B QA programs is as follows:

> Structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Additional guidance in this regard is provided in the regulatory position of Regulatory Guide 1.29, "Seismic Design Classification."

It is our intent to arrive at a safety-related list for the Vogtle Electric Generating Station application that is generally consistent, both in scope and level of detail, to safety-related lists shown in the past OL applications. Therefore, it is expected that the list will be directed primarily to the system level of detail (i.e., there is no need to identify and list every safety-related component or structure within a system). However, if a component consists of a combination of safety-related and nonsafety-related functions (e.g., reactor coolant pump), you should ascertain that the safety-related portions of that component are included in the list. It is requested that adequate justification be provided for substantive additions, deletions, or expansion in level of detail should such situations arise.

In conducting your review, you should use the applicant's FSAR Table 3.2.2-1. You are requested to:

1. Provide a listing of those items that fall within your areas of review responsibility.

2. Identify those safety-related items that should be added to the FSAR and provide the justification for these additions based on the above criterion. Included in your considerations should be the need to expand the level of detail of specific items for clarity.

3. Confer with your counterparts in other branches, as necessary, to assure that those items whose responsibility for review you may question are indeed addressed by at least one reviewer.

Your response should be transmitted by memorandum to the QA Branch by no later than December 9, 1983. If you have any questions, contact J. Spraul on x24530.

JGPartlow

James G. Partlow, Acting Director
Division of Quality Assurance,
Safeguards, and Inspection Programs
Office of Inspection and Enforcement

Melanie A. Miller

M. A. Miller, Project Manager
Licensing Branch No. 4
Division of Licensing
Office of Nuclear Reactor Regulation

cc: E. Adensam, NRR

Attachment 26

October 22, 1982

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-II-82-115

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region II staff on this date.

FACILITY: Georgia Power Company
Vogtle Units 1 and 2
Docket Nos. 50-424 and 50-425
Waynesboro, Georgia

Licensee Emergency Classification:
____ Notification of Unusual Event
____ Alert
____ Site Area Emergency
____ General Emergency
__X__ Not Applicable

SUBJECT: ARRESTS OF CONTRACTOR EMPLOYEES ON DRUG CHARGES

[illegible]month investigation into drug sales and use, initiated by the licensee, has resulted in the arrest of 12 persons employed by two contractors at the Vogtle construction site. No licensee employees are involved. Further arrests are expected.

Arrests were made on October 21 and 22 by the sheriff of Burke County, Georgia, on a variety of drug-related charges. Substances involved include marijuana, cocaine and valium.

Vogtle Unit 1 is 29 percent complete, and Unit 2 is 10 percent complete. No fuel is on site.

Media interest has occurred. The licensee notified the media in the plant area of the arrests and referred them to the sheriff for more details. The NRC does not plan a news release.

Region II was advised of the investigation at its outset and has been kept informed periodically during its course.

The Information Assessment Team was notified.

The State of Georgia has been informed.

This information is current as of 9 a.m. today.

Contact: D. R. McGuire, RII 242-5545 R. C. Lewis, RII 242-5680

DISTRIBUTION:

H. Street 10:54	MNBB 3:00	Phillips	E/W	Willste	MAIL:
Chairman Palladino	EDO	NRR 10:51	IE 10:56	NMSS 11:00	ADM:DMB
Comm. Gilinsky	PA		OIA	RES	DOT: Trans Only
Comm. Ahearne	MPA		AEOD		Applicable State
Comm. Roberts	ELD				
Comm. Asselstine		Air Rights	INPO		
SECY		SP 11:03	NSAC		
ACRS					

8211040000 821022
PDR I&E
PNO-II-82-115 PDR

Attachment 27

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

MAR 9 1983

X7BG10
Orig - RMS
Copy - M1
EST
CWH
XP - 433

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-321/83-06, 50-366/83-06, 50-424/83-06, AND 50-425/83-06

The NRC Systematic Assessment of Licensee Performance (SALP) Board has completed its periodic evaluation of the performance of the subject facilities. The Hatch facility, Units 1 and 2, and the Vogtle facility, Units 1 and 2, were evaluated for the period of July 1, 1981 through October 31, 1982. The results of the evaluation are documented in the enclosed SALP Board Assessment. This evaluation will be discussed with you at your offices in Atlanta, Georgia on March 17, 1983.

The performance of your Hatch facility, Units 1 and 2, was evaluated in the functional areas of plant operations, radiological controls, maintenance, surveillance, fire protection, emergency preparedness, security and safeguards, refueling, licensing activities, and quality assurance.

Construction performance at the Vogtle facility, Units 1 and 2, was evaluated in the functional areas of soils and foundation, containment and other safety related structures, piping systems and supports, safety related components, licensing activities and, construction quality assurance program. The SALP Board's evaluation of your performance in these functional areas is contained in the SALP Board Assessment which is enclosed with this letter.

The SALP Board evaluation process consists of categorizing performance in each of the functional areas. The categories which we have used to evaluate the performance of your facilities are defined in section II of the enclosed SALP Board Assessment. Any comments which you have concerning our evaluation of the performance of your facility should be submitted to this office within twenty days following the date of our meeting in Atlanta, Georgia.

Your comments, if any, and the SALP Board Assessment, will both appear as enclosures to the Region II Administrator's letter which issues the SALP Board Assessment as an NRC Report. In addition to the issuance of the assessment, this letter will, if appropriate, state the NRC position on matters relating to the status of your safety programs.

In accordance with 10 CFR 2.790(a), a copy of this letter, the enclosure, and your response, if any, will be placed in the NRC's Public Document Room unless you notify this office, by telephone, within ten days following the date of our meeting in Atlanta, Georgia, and submit written application to withhold informa-

tion contained therein within twenty days following the date of our meeting. Such application must be consistent with the requirements of 10 CFR 2.790(b)(1).

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

H C Dance /for

R. C. Lewis, Director
Division of Project and
Resident Programs
Region II SALP Board Chairman

Enclosure:
SALP Board Assessment for
Georgia Power Company

cc w/encl:
J. T. Beckham, Vice President and General Manager, Nuclear Generation
H. C. Nix, Plant Manager
C. E. Belflower, Site QA Supervisor
H. H. Gregory, III, Construction Project Manager
E. D. Groover, QA Site Supervisor
D. O. Foster, Project General Manager
M. Manry, Plant Manager

U. S. NUCLEAR REGULATORY COMMISSION
REGION II

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE BOARD ASSESSMENT

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NUMBERS 50-321 AND 50-366

ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2

DOCKET NUMBERS 50-424 AND 50-425

JULY 1, 1981 THROUGH OCTOBER 31, 1982

INSPECTION
REPORT NUMBERS

50-321/83-06; 50-366/83-06
50-424/83-06; 50-425/83-06

I. INTRODUCTION

A formal licensee performance assessment program has been implemented in accordance with the procedures discussed in the Federal Register Notice of March 22, 1982. This program, the Systematic Assessment of Licensee Performance (SALP), is applicable to each operator of a power reactor or holder of a construction permit (hereinafter referred to as the licensee). The SALP program is an integrated NRC staff effort to collect available observations of licensee performance on a periodic basis and evaluate performance based on these observations. Positive and negative attributes of licensee performance are considered with emphasis placed on understanding the reasons for a licensee's performance in important functional areas, and sharing this understanding with the licensee. The SALP process is oriented toward furthering NRC's understanding of the manner in which: (1) the licensee directs, guides, and provides resources for assuring plant safety; and (2) such resources are used and applied. The integrated SALP assessment is intended to be sufficiently diagnostic to provide meaningful guidance to the licensee. The SALP program supplements the normal regulatory processes used to ensure compliance with NRC rules and regulations.

II. CRITERIA

Licensee performance is assessed in certain functional areas depending on whether the facility has been in the construction, preoperational, or operating phase during the SALP review period. These functional areas encompass a wide spectrum of regulatory programs and represent significant nuclear safety and environmental activities. A functional area may not be assessed because of little or no licensee activities in this area, or lack of meaningful NRC observations.

One or more of the following evaluation criteria were used to assess each functional area:

- Management involvement in assuring quality
- Approach to the resolution of technical issues from a safety standpoint
- Responsiveness to NRC initiatives
- Enforcement history
- Reporting and analysis of reportable events
- Staffing (including management)
- Training effectiveness and qualification

The SALP Board has categorized functional area performance at one of three performance levels. These levels are defined as follows:

<u>Category 1</u>: Reduced NRC attention may be appropriate. Licensee management attention and involvment are aggressive and oriented toward nuclear safety; licensee resources are ample and effectively used such that a high level of performance with respect to operational safety or construction is being achieved.

Category 2: NRC attention should be maintained at normal levels. Licensee management attention and involvement are evident and are concerned with nuclear safety; licensee resources are adequate and are reasonably effective such that satisfactory performance with respect to operational safety or construction is being achieved.

Category 3: Both NRC and licensee attention should be increased. Licensee management attention or involvement is acceptable and considers nuclear safety, but weaknesses are evident; licensee resources appear to be strained or not effectively used such that minimally satisfactory performance with respect to operational safety or construction is being achieved.

III. SUMMARY OF RESULTS

A. Overall Utility Evaluation

The Georgia Power Company (GPC) corporate organization has developed into a strong and well managed group which exhibits excellent control over the nuclear projects. They are responsive to NRC concerns and initiatives with increased commitment to improving performance throughout the nuclear projects. The corporate staff is highly energetic in resolving technical issues and in presenting a complete assessment of safety significances. Examples of resolved technical issues for each facility are, for Hatch, the chemical intrusion event and for Vogtle, the Millet Fault Study. A particular strongpoint for Vogtle is the concept of a Project Management Board whereby GPC corporate staff, Westinghouse, Bechtel, and Southern Company Services, Inc. can direct and resolve issues. No significant utility weaknesses have been identified.

B. Overall Facility Evaluation - Hatch 1 and 2

The Hatch facility appears to be well managed. The licensee has demonstrated strong corporate management support for, and commitment to, the improvement of overall plant performance. Increased corporate management involvement at the site, the addition of sufficient resources to resolve technical issues, and several site management changes have enhanced overall plant management effectiveness and plant performance. Improvements have been made in the area of radiation protection where the previous SALP report had identified a weakness in the licensee's contamination control program. A major strong area was noted in the area of licensing activities while no significant weak areas were identified. Although there have been two civil penalties assessed during the period, improvements in the areas of security and operations have been made.

C. Facility Performance - Hatch 1 and 2

Tabulation of ratings for each functional area:

Operations (Units 1 and 2)

1. Plant Operations - Category 2
2. Radiological Controls - Category 2
3. Maintenance - Category 2
4. Surveillance - Category 2
5. Fire Protection - Category 2
6. Emergency Preparedness - Category 2
7. Security and Safeguards - Category 2
8. Refueling - Not Rated
9. Licensing Activities - Category 1
10. Quality Assurance Program - Category 2

D. Overall Facility Evaluation - Vogtle 1 and 2

GPC has developed, and is implementing, a vigorous construction project management effort with well qualified and experienced personnel. Major strengths were noted in the areas of containment and other safety related structures, licensing activities, and quality assurance program.

Although the number and severity levels of the violations in the areas inspected were significant, they do not indicate a programmatic breakdown; they appear to be a result of a failure to prepare adequate procedures to implement NRC requirements and licensee commitments. There were no significant weaknesses identified.

E. Facility Performance - Vogtle 1 and 2

Tabulation of ratings for each functional area:

Construction (Units 1 and 2)

1. Soils and Foundation - Category 2
2. Containment and Other Safety Related Structures - Category 1
3. Piping Systems and Supports - Category 2
4. Safety Related Components - Category 2
5. Support Systems - Not rated
6. Electrical Power Supply and Distribution - Not rated
7. Instrumentation and Control Systems - Not rated
8. Licensing Activities - Category 1
9. Quality Assurance Program - Category 2

F. SALP Board Members:

R. C. Lewis, Director, Division of Project and Resident Programs (DPRP) (Chairman), Region II (RII)
J. A. Olshinski, Director, Division of Engineering and Operational Programs, RII
J. P. Stohr, Director, Division of Emergency Preparedness and Materials Safety Programs, RII
H. C. Dance, Chief, Project Branch 2, DPRP, RII

G. SALP Board Attendees:

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M. V. Sinkule, Chief, Operational Support Section, Program Support Staff, RII
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IV. PERFORMANCE ANALYSIS FOR HATCH UNITS 1 AND 2

A. Functional Area Evaluations

Licensee Activities

Major Unit 1 outages occurred during the appraisal period as follows: October 9 through November 26, 1981 to repair leaking fuel; December 25, 1981 through February 8, 1982 to investigate and repair damage to the main turbine; April 24 through June 12, 1982 for a chemical intrusion outage; July 3 through July 11, 1982 for investigation of main safety relief valve problems; and from October 9, 1982 through the end of the review period for performing torus modifications and refueling.

Major Unit 2 outages occurred from February 19 through May 25, 1982 for performance of torus modifications and refueling. The torus modification program was completed for Unit 2.

Inspection Activities

Routine NRC inspections were performed during this evaluation period. An emergency preparedness appraisal was conducted in December 1981, and a performance appraisal team inspection was conducted in April and May of 1982. Additionally, special inspections were performed involving steam relief valve problems and successive ruptures of valve diaphrams in the high pressure coolant injection (HPCI) system.

1. Plant Operations

   a. Analysis

      During the evaluation period, two inspections were performed by regional based inspectors in the area of plant operations. These complimented the routine inspections performed by the resident inspection staff. In the area of plant operations, 16 violations were identified as follows:

      (1) Severity Level III violation (6 examples) regarding the HPCI valve diaphram problem. A $50,000 civil penalty was assessed for facility changes which were made without prior NRC approval or without the licensee having conducted an evaluation as required.

      (2) Severity Level IV violation for failure to maintain two independent plant service water (PSW) loops operable.

      (3) Severity Level IV violation for failure to verify PSW system valve position.

      (4) Severity Level IV violation for a PSW valve not being locked open as required by procedures.

5) Severity Level IV violation for failure to declare Unit 2 "A" loop RHR system inoperable when a portion of a control panel was deenergized.

(6) Severity Level IV violation for failure to follow procedures requiring opening of safety relief valves to prevent plant overpressurization after a scram.

(7) Severity Level IV violation for loading fuel with a control rod withdrawn.

(8) Severity Level IV violation for failure to declare the reactor core isolation cooling (RCIC) system inoperable.

(9) Severity Level IV violation for failure to report nonconservative settings for components of the RCIC system.

(10) Severity Level V violation for failure to follow procedures (3 examples).

(11) Severity Level V violation for failure to maintain isolation valves to low vacuum switches locked opened.

(12) Severity Level V violation for failure to operate the HPCI system in accordance with plant procedures.

(13) Severity Level V violation for failure to make appropriate notifications to the NRC. (2 examples).

(14) Severity Level V violation for failure to maintain the motor cooling water inlet valve for "A" residual heat removal (RHR) service water pump locked open.

(15) Severity Level V violation for valves in the RHR service water system not being locked in the appropriate positions.

(16) Severity Level V violation for failure to make a one hour notification to NRC of a reactor scram.

Management involvement and control to assure that quality of operation and product are accomplished, has improved during this assessment period. Corporate management was involved in site activities on a more frequent basis. Decision making was usually at a level which ensured adequate management review. Resolution of technical issues was generally performed in a timely manner.

The increased management involvement resulted in improved resolution of technical issues. Safety considerations were

consistantly given priority over economic considerations. This was evident when management freely extended planned shutdowns or kept the plant shut down during unplanned events such as the chemical intrusion and the safety relief valve problems until the safety consequences were understood or the needed changes were made.

Records were generally well maintained and available. Corrective action systems not only address significant problems, but generally addressed nonreportable concerns.

Procedures and policies were occasionally violated as shown by several violations listed above.

Responsiveness to NRC initiatives was generally sound and thorough. Generally, acceptable resolutions were proposed. The licensee has had no problems meeting deadlines in this area. Key staff positions have been identified and authorities and responsibilities for these positions have been defined. The training and qualification programs contributed to an adequate understanding of areas of responsibility.

Licensee event report submittals were acceptable with respect to information required by the LER submittal form. The information, to the extent it was provided, appeared to be technically accurate and understandable. There appeared to be a general tendency, though, to provide no additional descriptive information than the available space on the form allowed. Screening of the LERs for trend and patterns provided cases which appeared to be indicative of management weaknesses in the areas of: late or incorrectly performed surveillance tests, incorrect operating personnel actions or activities; and deficient technician workmenship; and inadequate procedures.

There was one Severity Level III violation identified during this period. This violation was not indicative of a programmatic breakdown. However, there were indications of a minor programmatic breakdown with regard to timely reporting of events and prevention of recurrence of violations. Although events are normally reported in a timely manner, there were four violations for instances in which time requirements were not met. With regard to prevention of recurrence of violations, the corrective actions were timely in most cases, but apparently ineffective in some instances. There were several repeat violations for improper positioning and locking of valves and for failure to follow procedures. The root cause for the 16 operations area violations seemed to be lack of attention to detail on the part of the plant staff. Some improvement was noted in the closing months of the SALP

period; however, continued effort for licensee improvement in this area is needed.

Areas of improvement over the previous SALP period were:

- A program to reduce the number of alarmed annunciators was undertaken and has proven to be effective.
- The noise level in the control room has been reduced by improvements to the ventilation system; and
- An onsite simulator was placed in operation. This provides a means for improving plant-specific training. The simulator was used effectively during the 1982 emergency preparedness drill.

b. Conclusion

Category 2

c. Board Comments

The board is aware of a number of improvements that have occurred during the period. However, licensee management must continue to be involved to ensure additional improvements are implemented.

2. Radiological Controls

a. Analysis

Six routine inspections were performed during the evaluation period by regional based inspectors. This included a confirmatory measurements inspection using the Region II mobile laboratory and an environmental protection inspection. Additionally, routine inspections were performed in this area by the resident inspectors. Two special reactive inspections were performed by regional based inspectors in this area during the review period. They covered overflow of the radwaste management system and subsequent flooding of the condensate bay and the radwaste building, and high conductivity in the reactor coolant system due to solvent intrusion. Four violations were identified as follows:

(1) Severity Level V violation concerning first line supervisors with less than the minimum experience required by the technical specifications for their assigned positions.

(2) Severity Level V violation for failure of management personnel to conduct timely reviews of environmental data sheets.

(3) Severity Level V violation for failure to correct misuse of personnel dosimetry.

(4) Severity Level V violation concerning incomplete licensee audits to determine compliance with environmental technical specification requirements.

The above violations are not indicative of significant programmatic deficiencies. Licensee management is adequately involved in radiological controls and is generally responsive to NRC concerns. Changes in senior site management since the last review period have resulted in increased management awareness of radiological concerns and have helped to expedite necessary improvements. Increased corporate involvement and cooperation has produced improvements in emergency planning, staffing and organization, ALARA, training, personnel contamination control, and low level radwaste management.

Licensee audits in this area were constructive and resulted in improvements in the licensee's program. Radiological problems are properly identified and reported.

Radioactive waste handling was satisfactory due to a constant effort in radwaste management. A review of records of the transport of radioactive material revealed no violations with NRC or Department of Transportation regulations. This is an improvement since the last review period. Inspections of liquid and gaseous radioactive effluent releases revealed compliance with applicable regulations, technical specifications, and licensee procedural requirements. By comparison to other facilities of similar size, effluent releases are considered normal.

The licensee's permanent health physics staff, although smaller in size than generally found at similar size facilities, has improved since the last review period and is adequate for routine plant operations. However, the staff may be placed under considerable stress during a nonroutine event. Further improvements in health physics, the staffing level and personnel qualifications are still needed.

Personnel radiation exposure totals for this review period show a significant increase compared to previous periods. This increase is attributable to extensive plant modifications required by NRC and the nuclear steam supply system vendor. In general, the management controls exercised by the

licensee during maintenance outages have been adequate and have resulted in exposure totals less than projected. Improvements have been seen in the licensee's dosimetry program due to constant attention and thorough work effort in this area.

The quality control and confirmatory measurements inspection performed during the review period, using the Region II mobile laboratory, identified no violations. The results of liquid and gaseous effluent samples analyzed during the inspection by both the licensee and the NRC showed agreement. The need was identified for upgrading quality control activities associated with sampling and analysis of reactor coolant and feedwater samples. All other aspects of the laboratory program were found to be satisfactory.

The environmental protection inspection performed during the review period identified violation (4) above regarding failure to conduct annual quality assurance audits of the radiochemical analytical program. This violation is not indicative of programmatic weaknesses. In general, all remaining requirements of the environmental program were well managed and were oriented toward continued maintenance of effective environmental protection. No additional problem areas were disclosed during the appraisal period.

b. Conclusion

Category 2

c. Board Comments

None

3. Maintenance

a. Analysis

During the evaluation period, five inspections were performed by regional based inspectors in the area of plant maintenance. These complimented the routine inspections performed by the resident inspection staff. In this area, one violation was identified as follows:

> Severity Level V violation for failure to follow procedures during welding operations.

Corrective action was timely and effective for the above violation.

Management involvement and control in assuring quality has been satisfactory. There is evidence of prior planning and assignment of priorities. Activities are controlled by defined procedures. Corporate management is frequently involved in site activities; decision making is usually at a level that ensures adequate management review. Reviews are generally timely, thorough, and technically sound. Records are generally complete, well maintained, and available. Procurement is generally well controlled and documented.

In resolving technical issues, the licensee's understanding of the issues is usually apparent; conservatism is generally exhibited.

b. Conclusion

Category 2

c. Board Comments

The board believes that the high component failure rate as identified in paragraph IV.B.1.a. of this report, is indicative of inadequate management attention in the preventive maintenance area and improvements are needed.

4. Surveillance

a. Analysis

During the evaluation period, one inspection was performed by regional based inspectors in the area of inservice inspection. This complimented the routine inspections performed by the resident inspection staff. In the area of routine plant surveillance, two violations were identified:

(1) Severity Level IV violation for an inadequate surveillance testing of the standby gas treatment system.

(2) Severity Level V violation for an inadequate surveillance procedure for testing a portion of the RHR system.

Management involvement and control in assuring quality has been satisfactory. There is evidence of prior planning and assignment of priorities. Policies have been well stated, disseminated, and understood. Decision making is usually at a level that ensures adequate management review. Audits are generally complete and thorough; reviews generally timely, thorough, and technically sound. Records are usually complete, well maintained, and available. Procedures and policies are rarely violated.

Resolution of technical issues by the licensee is usually performed in a sound, thorough, conservative, and timely manner.

Responsiveness to NRC initiatives has been above average in the surveillance area. Acceptable resolutions have been proposed initially in most cases. These resolutions have been technically sound and have been arrived at in a timely manner. The licensee's responsiveness to resolving problems associated with the emergency core cooling system (ECCS) functional testing program is one example of their fine performance.

One discrepancy identified during the reporting period found that the ECCS functional tests did not fully test the complete path from the sensor to the activated device, including some trips and alarms. Once informed of the problem, the licensee responsed immediately by initiating a program complete with special procedures to ensure operability of all Unit 2 ECCS systems. Unit 2 was operating with Unit 1 shut down for commencement of a four month outage at the time of discovery. The licensee committed to a complete review of all surveillances which test operability of the ECCS systems. Subsequent to the reporting period, the licensee has completed the review of all these surveillances.

b. Conclusion

Category 2

c. Board Comments

None

5. Fire Protection

a. Analysis

During this assessment period, one inspection was conducted by regional personnel. Additional inspections were performed in this area by the resident inspectors. The following violations were identified:

(1) Severity Level IV violation for failure to post fire watches for nonfunctional fire barriers.

(2) Severity Level V violation for failure to report a fire main rupture to NRC within 24 hours.

(3) Severity Level V violation for failure to implement the fire protection program, for improper issuance of fire prevention welding permits and for unauthorized storage of combustible materials within a safety related heating, ventilation, and air conditioning (HVAC) equipment room.

The licensee's fire protection and prevention administrative procedures were reviewed and found to comply with NRC guidelines. Adherence to these procedures was, as a whole, satisfactory, with the exception of the above identified violations. The plant fire brigade was found to be adequately organized, well trained, and provided with sufficient fire fighting equipment. The fixed plant fire protection equipment and systems were found to be adequate with the exception of a number of deficient fire doors and shutoff fire protection water spray systems supplying the charcoal filter units for several HVAC systems. The deficient fire doors were included in the above nonfunctional fire barrier violation and have been repaired and restored to service. The filter fire protection systems were shut off due to leaking deluge valves apparently caused by the system's being supplied from water systems having too low a head pressure. However, these deluge valves were capable of being manually operated. Subsequently, the valves have been replaced and/or a new higher pressure water supply has been provided to each system. The systems were reported restored to automatic operation in June and July of 1982. An NRC review of the maintenance and tests of the fire protection systems was not made during this assessment period, but was reviewed during the previous assessment period and found to be satisfactory, except for the above identifed shutoff fire protection systems.

Overall, management involvement and control of the major fire protection elements is adequate. The responsiveness to NRC initiatives is generally timely, with major violations rare. Fire protection related events have been properly reported, except for item (2) above. Staffing and training of the fire protection program is adequate.

b. Conclusion

Category 2

c. Board Comments

None

5. Emergency Preparedness

a. Analysis

During the evaluation period, three inspections were performed in the area of emergency preparedness. One inspection involved the observation of a full scale radiological emergency exercise, one was an overall appraisal of the emergency preparedness program for the Hatch facility, and the third was a routine inspection.

During the emergency exercise of October 21-22, 1981, five deficiencies were identified; these resulted in a Confirmation of Action letter to the licensee, dated November 3, 1981.

These deficiencies involved:

- Plant manager's role with respect to command and control functional responsibilities;
- Assignment of responsibility and training associated with emergency notifications;
- Function and staffing of the technical support center;
- Control, staffing, and equipment for the operational support center; and
- Differences between the utility and the State regarding offsite dose calculations and protective action decision making.

The emergency preparedness appraisal was conducted November 30 - December 11, 1981. Five programmatic deficiencies were identified and documented in a Confirmation of Action letter to the licensee dated December 17, 1981. These deficiencies involved:

- Lack of clarification and definition of the emergency organization;
- Failure to establish or implement a comprehensive, coordinated program for training emergency response personnel;
- Not having equipment for obtaining and handling a post-accident sample of the containment atmosphere;
- Inadequate procedures for offsite radiological surveying and containment atmosphere post accident sampling; and

- Inadequate initial dose assessment procedure and capability.

Region II concerns with the overall state of emergency preparedness at plant Hatch were discussed with Georgia Power Company management at a meeting in the Region II office on January 7, 1982.

Georgia Power Company has been responsive to the expressed concerns and has taken prompt action to correct the identified deficiencies and, in general, to upgrade the emergency preparedness program in accordance with current criteria and regulations. A routine safety inspection in August 1982 verified that all deficiencies had been corrected and that the licensee had devoted considerable resources and effort to upgrading the emergency preparedness program for the Hatch facility. Additional inspection effort subsequent to this assessment period indicated that the licensee's effort had upgraded the program substantially.

b. Conclusion

Category 2

c. Board Comments

During the initial portion of the reporting period, licensee performance was considered to be category 3. However licensee management has been responsive to identified deficiencies, and the level of performance improved to category 2 at the end of the reporting period.

7. Security and Safeguards

a. Analysis

Five routine unannounced inspections and one special announced inspection were performed by regional based inspectors. Additional routine inspections by the resident inspectors were performed throughout the evaluation period. Three violations were identified as follows:

(1) Severity Level III violation for failure to conduct a thorough search of a vehicle prior to its entry into the protected area and the failure to provide compensatory protection at the vehicle gate when the intrusion alarm system was placed in the ACCESS mode. A $20,000 Civil Penalty was assessed for failure to conduct a thorough search of a vehicle prior to entering a protected area.

(2) Severity Level V violation for failure to search for explosives with explosive detection equipment or conduct a physical hands-on search of personnel entering the protected area.

(3) Severity Level V violation concerning the failure to provide sufficient authorized escorts and to maintain continuous surveillance of escort-required visitors within the protected area.

Violations (1) and (2) above were due to failure to control access to the protected area. While Violation (1) was the subject of civil penalty, the following mitigating circumstances applied: the driver was authorized unescorted access to the protected area; driver and vehicle were escorted in and out of the protected area; the steering wheel and doors to the cab were locked while in the protected area, and the vehicle escort guard had the keys during this time. The cause was inadequate procedures for control of vehicles entering the protected area. A new Security Local Order was prepared; security personnel were retrained; and, a permanent manned post was established at the vehicle gate. This appears to be adequate to prevent further reccurence.

Violation (2) involved noncompliance with the physical security plan and was attributed to personnel error. Security procedures were changed to increase the amount of required supervision to ensure compliance.

These violations indicated a programmatic and systemmatic deficiency in access controls of the protected area.

The licensee provided prompt and thorough corrective actions to the above violations and all identified technical issues raised during security inspections. These violations were not indicative of the total effectiveness and proficiency of the security program at the Hatch plant.

Corporate and site managements' support and security awareness is positive as indicated by their professional approach to provide a safe and secure environment onsite; their responsiveness to all NRC comments and discussions; and the non-adversary relation with onsite personnel. The proprietary security guard force is adequately staffed to meet all commitments of the security plan and of the contingency plan. Review of the training and qualification plan, observations of on-the-job training and structured training classes, and interviews with security force personnel indicated that the security training, as programmed, was being efficiently and effectively implemented. This was also demonstrated by the positive morale of the security force.

b. Conclusion

Category 2

c. Board Comments

None

8. Refueling

a. Analysis

Although observed by the resident inspection staff, both the refueling of Unit 2 and the Unit 1 fuel movements during the chemical intrusion event, an in-depth review of the refueling program was not performed. One violation, however, was identified on Unit 1 as follows:

Severity Level IV violation for loading fuel into the reactor with a control rod withdrawn.

This violation was due to improper training and an inadequate procedure. Immediate and appropriate corrective actions were initiated and completed by the licensee. The example is not an indication of a programmatic breakdown.

b. Conclusion

Not Rated

c. Board Comments

There was insufficient inspection activity in this area to determine a rating.

9. Licensing Activities

a. Analysis

The assessment was based on an evaluation of the following significant licensing actions:

-Responses to NUREG-0737 items
-Appendix R activities
-Torus modifications
-Implementation of equipment qualification program
-Reload reviews
-Operator licensing
-Scram discharge volume system modifications

For the individual licensing actions in the above list, management involvement and control was excellent. The technical competence demonstrated by the licensee and his consultant. Bechtel, during the Appendix R alternate safe shutdown review, was excellent. In addition, the licensee's responsiveness to NRC questions on fire protection was excellent. Management promptly became involved in issues, kept NRC informed, and provided NRC with completion schedules.

The licensee has demonstrated an understanding of the safety consequences of technical issues and has provided acceptable resolutions to problems. Resolution is sometimes hampered by the need to factor in the inputs of Southern Company Services, Inc. and Bechtel, whose offices are located away from the licensee's corporate headquarters.

Regarding licensee responsiveness, a certain number of submittals were late; but the licensee always requested an extension of time and had adequate reasons for the delay. In many instances the delay was due to a large build-up of NRC requirements at the time.

As noted in the previous SALP assessment, the licensee's corporate staff is limited in size but appears to be composed of highly qualified individuals. Additional technical competence is provided by the efficient use of consultants such as General Electric, Southern Company Services Inc., and Bechtel.

During the current SALP interval, five individuals out of thirteen failed the senior reactor operator examination, and five out of eighteen failed the reactor operator examination. This amounted to ten failures out of a total of 31 examinations.

b. Conclusion

Category 1

c. Board Comments

A high level of performance with respect to licensing activities has been acheived.

10. Quality Assurance Program

a. Analysis

Five inspections were performed in this area during the review period. Two of these inspections were conducted to

review licensee actions taken as a result of previous inspections. Two of the remaining inspections involved the areas of training and requalification training of personnel. During the inspections of the training areas, three violations were identified and are discussed in the operations area. There were three violations identified during this period specific to quality assurance programs:

(1) Severity Level IV violation for failure to provide quality assurance (QA) indoctrination to all non-QA personnel.

(2) Severity Level V violation for failure to adequately describe the process to be used to review procedures to assure that quality requirements were incorporated.

(3) Severity Level V violation for failure to apply quality control (QC) hold-tags to equipment that had not been inspected.

Audits performed by management and QA personnel were not always timely, as evidenced by failure to submit an audit schedule by four of the seven members of the Safety Review Board. This was identified as an inspector follow-up item.

Committees were properly staffed, and functioned where required. Reviews were generally timely, thorough, and technically sound. Violation (2), above, involves failure to provide adequate control for maintenance and testing equipment found outside their calibration requirements.

A corrective action system existed to identify and address nonreportable concerns; however, several items disclosed by the system were not given supervised management attention to correct the particular concerns. In addition, several inspector follow-up items from previous inspection reports (81-17 and 81-27) indicated that adequate attention was not provided to correct the items of concern in a timely manner.

Procurement activities were generally well controlled. Violation (3) was associated with this area. The licensee's responsiveness to NRC enforcement actions was poor, in that time extensions and repeated submittals were required to obtain acceptable responses. This was apparently due to a lack of thoroughness or depth of review in the initial submittals.

Key positions were identified and authorities and responsibilities were defined. Staffing appeared to be adequate as discussed in inspection reports 81-20 and 82-24.

The training and qualification program was not effectively applied for a significant segment of the staff, as discussed in inspection reports 81-20 and 82-24. One violation of this area is identified in item (1) above.

Although significant improvement was achieved in early 1981, as stated in the previous SALP assessment, several areas continue to indicate the need for increased management attention to QA requirements. Near the end of the appraisal period, the licensee made organizational changes intended to create a stronger corporate QA staff. A new corporate QA manager was recently hired to provide more effective management involvement and control in assuring quality at the Hatch plant. The licensee appears to be providing more meaningful support and attention to the operational QA program.

b. Conclusion

Category 2

c. Board Comments

Although licensee performance during the initial portion of the period was marginal, licensee management appears to be making the changes necessary to improve performance in the area. It is recommended that increased licensee management attention be continued in this area.

B. Supporting Data

1. Reports Data

a. Licensee Event Reports (LERs)

Unit 1:

Of the more than 190 LERs submitted almost 50% were identified by the licensee as having been caused by component failures. These failures were evenly divided between electrical malfunctions and mechanical failures. The electrical malfunctions were in turn evenly divided between circuit failures and amplifier or set-point drift requiring recalibration or adjustment. Roughly 10% of the LERs were tied to various types of personnel error; approximately 7% were procedural in nature, and the rest were assigned to design/fabrication or some other non-repetitive cause.

A detailed breakdown of causes as submitted by the licensee is as follows:

Cause	Number of LERs
Component Failure	96
Other	40
Personnel Error	38
Design, Manufacturing Construction/Installation	11
Defective Procedures	5
External Cause	3

Unit 2:

Of the more than 185 LERs reviewed, the distribution with respect to cause is remarkably similar to that shown for Unit 1. Once again about 50% were due to component failure; an even split between mechanical and electrical; the latter evenly divided between circuit failure and drift. Personnel errors contributed to 11% of the reports, and procedure problems were cited for a little over 7% of the reports. A detailed breakdown as submitted by the licensee is as follows:

Cause	Numbers of LERs
Component Failure	104
Other	33
Personnel Error	26
Defective Procedures	10
Design, Manufacturing Construction/Installation	9
External Cause	1

b. Part 21 Reports

Unit 1: 1

Unit 2: 0

2. Investigation and Allegation Review

No major investigations were performed during the reporting period.

3. Enforcement Actions

a. Violations

Severity Level I, II - No violations
Severity Level III - 2 violations
Severity Level IV - 11 violations
Severity Level V - 19 violations
Severity Level VI - no violations

b. Civil Penalties

March 29, 1982 - concerning facility changes which were made without prior NRC approval or without having conducted an evaluation as required.

July 13, 1982 - concerning an inadequate search of a vehicle and the lack of a posted guard at a gate while the alarm zone was in ACCESS.

c. Orders

No orders were issued.

d. Administrative Actions

(1) Confirmation of Action Letters

November 3, 1981 - concerning resolution of emergency preparedness program deficiencies.

December 17, 1981 - concerning emergency preparedness program deficiencies.

(2) Management Conferences

May 17, 1982 - enforcement conference involving an unauthorized handgun in the Protected Area.

July 26, 1982 - enforcement conference concerning the loading of fuel with control rods withdrawn.

4. Reactor Trips

During the evaluation period 12 trips were experience on Unit 1 and 13 were experienced on Unit 2. These unplanned trips are delineated below.

a. Unit 1

7/6/81	Condenser vacuum switches valved out
8/10/81	Auto scram on RPS
8/11/81	IRM spike
8/14/82	Turbine stop valve fast closure
10/1/81	APRM functional test error

11/27/81	MSIV closure and group 1 isolation on A and B channels - spurious signal.
2/12/82	Low reactor water level
*4/24/82	High conductivity due to chemical intrusion
*6/1/82	High conductivity due to chemical intrusion
*6/12/82	Blown gaskets on "A" water box
7/3/82	False high water pressure indications
8/13/82	High level in A and B moisture separators

b. Unit 2

9/2/81	Turbine control valve fast closure
9/9/81	MSIV closure - group 1 isolation unknown cause
10/1/81	Turbine control valve and MSIV fast closure
10/5/81	Moisture Separator high level trip
11/4/81	MSIV closure due to loss of service air
11/27/81	Unidentified cause
1/23/82	Low reactor water level
2/17/82	Low reactor water level
6/5/82	Fast Closure - controller failed upscale
6/18/82	Low reactor water level
7/18/82	Low suction on condensate booster pump
*7/30/82	Diesel generator problems
8/25/82	MSIV closure above to high steam flow

*Forced to manually scram to stay within LCO.

V. PERFORMANCE ANALYSIS FOR VOGTLE UNITS 1 AND 2

A. Functional Area Evaluations

Licensee Activities

Between July 1, 1981 to October 31, 1982, the construction has progressed from 22.4% to 37.1% completion. Unit 1 progressed from 27% to 43% over the same period. Staffing for the project has gradually increased to the present level of manpower.

Construction		7626
Engineering		797
Power Generation		141
Others		32
	Total	8596

The manpower was distributed over the following shifts:

A - Day shift, ten hours for four days (Monday - Thursday)
B - Night shift, ten hours for four days (Monday - Thursday)
C - Day shift, twelve hours for three days (Friday - Sunday)
D - Night shift, twelve hours for three days (Friday - Sunday)

The Unit 1 primary containment building has progressed to the spring line on the outside with all of the concrete in place inside to and including the refueling floor at elevation 220. The Unit 1 containment dome has been fabricated, with the containment spray piping being installed prior to final placement.

During this period, the Millet fault was investigated and evaluated. The report has been completed and presently is under review by NRC. The investigation concluded that the postulated fault is incapable and has no impact on the design of the Vogtle project.

During this period, GPC management volunteered the Vogtle project for a pilot construction QA program developed by the Institute for Nuclear Power Operations (INPO) and also volunteered for an INPO designed self-evaluation program.

Inspection Activities

Routine inspection programs were performed during this evaluation period. In November 1981, an NRC Resident Inspector (Construction) was permanently assigned to the facility.

1. Soils and Foundation

   a. Analysis

      Portions of six onsite inspections, performed by regional inspectors during the evaluation period, were made in this

area. The resident inspector also conducted periodic inspections. The inspectors reviewed quality assurance (QA) implementing procedures, observed work activities including site preparation and foundations, compaction of Category 1 backfill, installation of caisons for the radioactive waste building foundation, and review of quality records. No violations or deviations were identified.

The quality assurance and quality control (QA/QC) procedures and controls met NRC requirements. Work activities were found to have been performed in accordance with QA/QC procedure requirements. The QA records were generally complete, well maintained, and retrievable. No construction deficiency reports (CDRs) were submitted by the licensee in this area.

Management involvement, resolution of technical issues, and staffing and training were adequate for the level of activity involved.

In addition to the inspections, an investigation was performed by a regional inspector and an investigator, of three allegations made by a former employee pertaining to improper QA testing of backfill materials and falsifications of backfill QC test results. Two of the allegations were not substantiated. The one allegation which was substantiated had been identified by the licensee's QA program and corrected prior to the investigation. No violations or deviations were identified in this area during the investigation. The licensee was very cooperative with the NRC investigators.

b. Conclusion

Category 2

c. Board Comments

None

2. Containment and Other Safety Related Structures

a. Analysis

Portions of several regional office inspections were performed in the area of structural concrete construction activities during the evaluation period. The resident inspector also conducted periodic inspections. The inspections involved review of QA implementing procedures; observations of work activities including containment structure; containment concrete; weld procedures; weld material embeds; grounding cable; rebar installations; layout of walls and

lines; plate welds; and review of quality records. Five violations and two deviations were identified. One of the violations was identified during the investigation discussed below. The violations and deviations were as follows:

(1) Severity Level V violation for failure to have documented instructions or procedures to control survey work activities relating to concete embed locations and equipment installation.

(2) Severity Level V violation for insufficient weld metal on structural steel weldments.

(3) Severity Level V violation for a weld procedure not meeting code requirements.

(4) Severity Level VI violation for failure to control welding expendables such as red marker and masking tape.

(5) Severity Level VI violation concerning failure to identify on the revelant QA records, the data recorder or inspector who performed the testing related to quality control of concrete.

(6) Deviation for failure to perform separate tensile test cycles for each splice crew as required by Regulatory Guide 1.10.

(7) Deviation from Preliminary Safety Analysis Report commitments regarding failure to sample and test cadwelds in accordance with the recommendations of Regulatory Guide 1.10.

The violations and the deviations are not indicative of a programmatic breakdown, but are a result of a failure to prepare adequate procedures to implement NRC requirements and licensee commitments. The violations and deviations are considered minor. With the exception of the identified violations and deviations, QA/QC procedures and controls were found to meet NRC requirements. Work activities was found to have been performed in accordance with QA/QC procedure requirements. Except for the area noted in the violations which were the result of an inadequate procedure, QA records are generally complete, well maintained, and retrievable. No CDRs were identified by the licensee in this area.

Management involvement, resolution of technical issues, staffing, and training were definitely above average for the level of activity involved. The licensee was responsive in correcting the violations and deviations. Training, instruction, experience, and procedure revision resulted in improved performance in this area.

In addition to the inspections, an investigation was performed, by a regional inspector and an investigator of four allegations made by a former employee pertaining to inadequate concrete QC testing and falsification of concrete QC test records. Two allegations were not substantiated. The remaining two allegations were partially substantiated. However, the licensee's QA program had detected and corrected the problems prior to the investigation. During the investigation, one violation was identified, (5) above. This violation was not associated with any of the allegations but was identified during review of concrete records. The licensee was cooperative with NRC investigators.

b. Conclusion

Category 1

c. Board Comments

A high level of performance with respect to containment and other safety related structures has been achieved. No decrease in licensee or NRC attention in this area is recommended.

3. Piping Systems and Supports

a. Analysis

During this evaluaton period, portions of seven inspections performed by regional based inspectors addressed this area. Additionally, routine inspections of this area were performed by the resident inspector. Inspections included observations of piping prior to concrete placement, hydrostatic testing of imbedded piping, welder qualification, storage of piping, inspection of pipe supports, welding, nondestructive examinations, and handling. Four violations were identified:

(1) Severity Level IV violation for failure to store pipe off-ground.

(2) Severity Level V violation for failure to provide adequate procedural requirements for visual inspection of weld transitions at joints of piping with differing wall thicknesses.

(3) Severity Level V violation for failure to follow procedural requirements for removal of an arc strike on piping.

(4) Severity Level VI violation concerning pre-weld cleanliness and control of welding materials.

A review of the violations does not indicate a breakdown of the program; the violations are considered minor in nature. No repetitive violations were identified, and corrective actions appeared to be prompt and effective. Improvements have been made in this area and further improvements are expected with the use of automatic pipe welding equipment (Diametric) on the large bore pipe. A training and qualification program is underway to utilize eight pieces of this equipment. This equipment, with training and experience, should result in a reduction of weld defects and improved performance in this area.

Within this report period, certain weld deficiencies were discovered on fabricated pipe spool pieces which were in controlled storage at the site. This prompted an extensive program to reinspect all of the spool pieces on site (approximately 15,611 pieces) and institute an inspection plan for future shipments.

The organization in this functional area appears to be adequately staffed with trained and qualified personnel. Procedural requirements implemented in this area appear generally satisfactory, but not outstanding.

b. Conclusion

Category 2

c. Board Comments

None

4. Safety Related Components

a. Analysis

During this evaluation period, portions of three inspections performed by regional based inspectors addressed this area. Additionally, the resident inspector performed routine inspections in this area. The inspections involved observations of the reactor pressure vessels, steam generator modifications, diesel generators, and storage of safety-related equipment. One violation was identified:

Severity Level VI violation for failure to load test a transporter of steam generators.

A related violation concerning the storage of a charging pump was identified and is addressed in the quality assurance program section of this report.

The violation noted above is not considered indicative of any programmatic breakdown. The procedural requirements implemented in this area appear satisfactory but not outstanding.

b. Conclusion

Category 2

c. Board Comments

None

5. Support Systems

a. Analysis

No inspections were performed in this area due to the early stages of construction activity.

b. Conclusion

Not rated

c. Board Comments

There was not sufficient licensee and NRC inspection activities in this area to justify a rating.

6. Electrical Power Supply and Distribution

a. Analysis

During the evaluation period, four inspections we. conducted by regional based inspectors. Additionally, inspections and routine surveillance were performed by the resident inspector on review of drawings, procedures, records, storage of equipment, maintenance, and installations. One violation was identified:

> Severity Level V violation for failure to perform periodic maintenance.

This violation resulted from personnel failure rather than a system breakdown; it was corrected by reinstructing the personnel involved. During this period, two concerns were identified. One related to electrical control panels. A reinspection was made of the panels received as a follow-up

to IE Information Notice 82-34, Rev. 1. This identified weld deficiencies in the panel received from Reliance Electric Company. This prompted an inspection of the supplier's facility by the licensee. The results of this inspection confirmed problems which warranted licensee issue of a temporary stop work order. Because of these events, a decision was made to perform an inspection of the electrical components in the control panels on site. The inspection identified deficiencies in the certifications and integrity of electrical connectors, identification of components, use of halogenated materials, and questionable wiring. This item was designated as a CDR.

A second concern involved cable tray bolts. During this period, a problem was identified relative to defective cable tray bolts which fracture at the bolt heads. Two types of splice bolts were furnished in bulk. One type had a silver-colored zinc finish made of C-1022 material and the other type has a gold-colored iridite finish made of C-1018 material. After the evaluation was completed by the licen[illegible] program to change all of the installed bolts to the go[illegible]-colored C-1018 material was instituted as corrective action. This was determined to be a CDR.

The program for the installation of cable and electrical equipment is in a very early stage with the total overall construction progress of Unit 1 is estimated to be 41%. These described problems, identified in the early stages have shown the licensee to be very responsive to problems and to have the resources to develop corrective actions with technical competence.

b. Conclusion

Not rated

c. Board Comments

It was recognized by the Board that licensee performance in this area was very good. However, it was felt that due to the early stages of construction, sufficient observations of the licensee's activities had not been made to warrant a rating.

7. Instrumentation and Control Systems

a. Analysis

No inspections were performed in this area due to the early stages of construction activity.

b. Conclusion

Not rated

c. Board Comments

There was not sufficient licensee or NRC inspection activity in this area to determine a rating.

8. Licensing Activities

a. Analysis

This assessment was based on an evaluation of the following licensing activities:

- Millett Fault Study
- QA Program Changes
- High Critical Damping Values for Cable Tray Supports
- Content of Environmental Report

Because substantially more staff time was involved in the Millett Fault Study, this activity has been weighted more heavily in the evaluation.

Management is stongly involved in assuring quality and is highly aware of licensing issues which ensure adequate management review. Management involvement was especially evident in the Millett Fault Study.

The approach to resolution of technical issues from a safety standpoint was an area of strong performance. Resolution in almost all cases was timely and offered a sound and thorough interpretation of the issues.

The licensee was very responsive and prompt to NRC initiatives and suggestions, offering cooperation and information. There appeared to be adequate staffing in those licensing activities that were evaluated.

The evaluation is limited due to the post construction permit licensing stage of Vogtle. Other than the Millett Fault Study, the licensing activities for Vogtle have provided limited staff/licensee interaction. Therefore, the conclusions are largely, but not entirely, based on the Millett Fault Study. Specifically, management attention was more than adequate with management actively involved with licensing issues. Technical issues were resolved in a timely

fashion with a high regard for safety. The licensee is responsive and cooperative to NRC initiatives.

b. Conclusion

Category 1

c. Board Comments

A high level of performance with respect to licensing activities has been achieved.

9. Quality Assurance Program

a. Analysis

One corporate QA inspection and five onsite QA inspections were performed during the assessment period by regional based inspectors. Additionally, routine inspections of the QA program and organization were made by the resident inspector. Four violations were identified as follows:

(1) Severity Level IV violation for failure to take prompt corrective action. During April 1981, the licensee identified in a QA audit that controls were inadequate to maintain the protective cover integrity for a chemical and volume control system (CVCS) pump. Another QA audit in January 1982 identified that similar in-place storage control appeared inadequate. In May 1982, an inspector found the condition still existed for the pump identified in April 1981 and four additional CVCS pumps were inadequately protected.

(2) Severity Level V violation for failure to provide protection required to prevent damage or deterioration of 14 pieces of equipment installed or stored in the Unit 2 auxiliary building.

(3) Severity Level V violation for failure to perform quarterly maintenance on two safety related components during the first and second quarters of calendar year 1982.

(4) Severity Level VI violation for a failure to obtain a statement of receipt from all procedure manual holders for controlled procedure changes.

Violation (4) was limited to the corporate office. Violations (1), (2), and (3) listed above involve failure to implement storage and maintenance requirements which indicates a need for continuing licensee and inspector attention.

The licensee designated Southern Company Services Inc. (SCS) as the lead organization to audit the Bechtel design and procurement supplier quality assurance activities. Becnte audits Westinghouse, the nuclear steam supply system supplier, and other vendors for the licensee. The licensee and Alabama Power Company conduct joint audits of the SCS QA department. The licensee's QA staff has been actively involved, and routinely audits, site construction activity. The audits were generally complete and thorough. Records were generally complete, well maintained, and retrievable. The corrective action systems generally recognized and addressed nonreportable concerns. Procurement activities were generally well controlled and documented.

The licensee's and contractor's approach to the resolution of technical safety issues was generally viable, conservative, and thorough in approach, and usually provided timely resolution.

The licensee's responsiveness to NRC initiatives, such as bulletins, circulars, and notices, is considered viable and generally sound. Bechtel, as the architect engineer, reviewed all such NRC initiatives for design input and transmits their response to the licensee. The licensee also reviews these initiatives and tracked the status of applicable correspondence. Corporate management maintained a continuing involvement in QA matters and routinely assessed the effectiveness of the program during quarterly committee meetings. The Senior Vice President for Project Management visited the Vogtle site approximately every six weeks to review construction status, inspection and audit activities, and site trend analyses.

There was evidence of prior planning, assignment of priorities, and defined procedures used to control activities. QA policies were adequately stated and understood. The system of audits was effective, but construction site followup and corrective action were not always effective as evidenced in item (1) above.

Staffing of QA positions appears to be adequate. Key positions were identified, and authorities and responsibilities were defined. Management independence was retained. QA staffing has increased with the expanded workload.

The training and qualification program contributed to an understanding of work and a reasonable adherence to procedures. A defined program is being implemented.

Staffing has been adequate for the status of construction. Positions were clearly identified with authorities and

responsibilities well defined. Training and qualifications were well defined and implemented. Personnel were adequately trained to understand the authority and responsibilities of their positions.

During this reporting period the QA staff has been increased from 11 to 14 persons. The Quality Control department has 223 inspectors and 20 supervisors. The above figures do not include the contractor's QA staff or internal auditors.

b. Conclusion

Category 2

c. Board Comments

None

B. Supporting Data

1. Reports Data

a. Construction Deficiency Reports

During the period, eight CDRs were reported. Four were caused by errors associated with the design of the component; two were due to manufacturing/fabrication errors; and, the cause for two of the deficiencies has not yet been determined.

2. Investigation and Allegation Review

No major investigations were conducted during the reporting period.

3. Enforcement Action

a. Violations

Severity Level I, II, III - No violations
Severity Level IV - 2 violations
Severity Level V - 8 violations
Severity Level VI - 5 violations

b. Civil Penalties

None

c. Orders

None

d. Administrative Actions

(1) Confirmation of Action Letters

None

(2) Management Conferences

None

Attachment 28

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

MAR 06 1984

Georgia Power Company
ATTN: Mr. R. J. Kelly
Executive Vice President
P. O. Box 4545
Atlanta, GA 30302

Gentlemen:

SUBJECT: REPORT NOS. 50-321/84-01, 50-366/84-01, 50-424/84-01, AND 50-425/84-01

The NRC Systematic Assessment of Licensee Performance (SALP) Board has completed its periodic evaluation of the performance of the subject facilities. The Hatch and Vogtle facilities were evaluated for the period November 1, 1982 through October 31, 1983. The results of the evaluation are documented in the enclosed SALP Board Assessment. In the past, as a part of the SALP program, we have routinely met with Georgia Power Company officials to discuss the results of the SALP Board's evaluation. The format of the SALP program has recently been revised so that this meeting is no longer a requirement, but may be held as a public meeting at the discretion of the licensee or NRC. In consideration of this revised format, please contact this office within ten days of the date of this letter to discuss the need for a meeting.

The performance of your Hatch facility was evaluated in the functional areas of plant operations, radiological controls, maintenance, surveillance, emergency preparedness, security and safeguards, licensing activities, and the quality assurance program.

Construction performance at the Vogtle facility was evaluated in the functional areas of soils and foundations, containment and other safety related structures, piping systems and supports, safety related components, electrical power supply and distribution, licensing activities, and the quality assurance program.

The SALP Board's evaluation of your performance in these functional areas is contained in the SALP Board Assessment which is enclosed with this letter.

The SALP Board evaluation process consists of categorizing performance in each functional area. The categories which we have used to evaluate the performance of your facilities are defined in section II of the enclosed SALP Board Assessment. Any comments which you have concerning our evaluation of the performance of your facilities should be submitted to this office within thirty days following the date of this letter.

Your comments, if any, and the SALP Board Assessment, will both appear as enclosures to the Region II Administrator's letter which issues the SALP Board Assessment as an NRC Report. In addition to the issuance of the assessment, this letter will, if appropriate, state the NRC position on matters relating to the status of your safety programs.

In accordance with 10 CFR 2.790 (a), a copy of this letter, the enclosure and your response, if any, will be placed in the NRC's Public Document Room unless you notify this office, by telephone, within ten days following the date of this letter and submit written application to withhold information contained therein within thirty days of the date of the letter. Such application must be consistent with the requirements of 10 CFR 2.790 (b)(1).

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

David M. Verrelli for

Richard C. Lewis, Director
Division of Project and
Resident Programs
Region II SALP Board Chairman

Enclosure:
SALP Board Assessment for
Georgia Power Company

cc w/encl:
J. T. Beckham, Vice President
and General Manager, Nuclear
Generation
H. C. Nix, Site General Manager
C. E. Belflower, Site QA Supervisor

U. S. NUCLEAR REGULATORY COMMISSION
REGION II

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE BOARD ASSESSMENT

GEORGIA POWER COMPANY
EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2
DOCKET NUMBERS 50-321 AND 50-366

VOGTLE ELECTRIC GENERATING PLANT UNITS 1 AND 2
DOCKET NUMBERS 50-424 AND 50-425

NOVEMBER 1, 1982 THROUGH OCTOBER 31, 1983

INSPECTION
REPORT NUMBERS
50-321/84-01; 50-366/84-01
50-424/84-01; 50-425/84-01

CONTENTS

I. INTRODUCTION

A formal licensee performance assessment program has been implemented in accordance with the procedures discussed in the Federal Register Notice of March 22, 1982. This program, the Systematic Assessment of Licensee Performance (SALP), is applicable to each operator of a power reactor or holder of a construction permit (hereinafter referred to as licensee). The SALP program is an integrated NRC staff effort to collect available observations of licensee performance on a periodic basis and evaluate performance based on these observations. Positive and negative attributes of licensee performance are considered with emphasis placed on understanding the reasons for a licensee's performance in important functional areas, and sharing this understanding with the licensee. The SALP process is oriented toward furthering NRC's understanding of the manner in which: (1) the licensee directs, guides, and provides resources for assuring plant safety; and (2) such resources are used and applied. The integrated SALP assessment is intended to be sufficiently diagnostic to provide meaningful guidance to the licensee. The SALP program supplements the normal regulatory processes used to ensure compliance with NRC rules and regulations.

II. CRITERIA

Licensee performance is assessed in certain functional areas depending on whether the facility has been in the construction, preoperational, or operating phase during the SALP period. These functional areas encompass a wide spectrum of regulatory programs and represent significant nuclear safety and environmental activities. Functional areas may not be assessed because of little or no licensee activities in these areas, or lack of meaningful NRC observations.

One or more of the following evaluation criteria were used to assess each functional area:

- Management involvement in assuring quality
- Approach to the resolution of technical issues from a safety standpoint
- Responsiveness to NRC initiatives
- Enforcement history
- Reporting and analysis of reportable events
- Staffing (including management)
- Training effectiveness and qualification

The SALP Board has categorized functional area performance at one of three performance levels. These levels are defined as follows:

Category 1: Reduced NRC attention may be appropriate. Licensee management attention and involvement are aggressive and oriented toward nuclear safety; licensee resources are ample and effectively used such that a high level of performance with respect to operational safety or construction is being achieved.

Category 2: NRC attention should be maintained at normal levels. Licensee management attention and involvement are evident and are concerned with nuclear safety; licensee resources are adequate and are reasonably effective such that satisfactory performance with respect to operational safety or construction is being achieved.

Category 3: Both NRC and licensee attention should be increased. Licensee management attention or involvement is acceptable and considers nuclear safety, but weaknesses are evident; licensee resources appear to be strained or not effectively used such that minimally satisfactory performance with respect to operational safety or construction is being achieved.

III. SUMMARY OF RESULTS

A. Overall Utility Evaluation

The Georgia Power Company corporate management has taken a number of measures to improve the effectiveness of its nuclear operations and construction activities. These measures have been responsive to NRC concerns and in general have provided resources needed to resolve technical issues aggressively. Particularly noteworthy has been the technical support and management direction provided at Vogtle to investigate and correct pipe weld and hanger deficiencies, vendor radiograph inadequacies, and poor electrical equipment storage practices.

In some areas at Hatch the same aggressive approach has been evident and effective; namely security, emergency preparedness, and inservice inspection and testing. However, in other areas there continues to exist significant deficiencies which indicate that corporate management efforts have not been totally effective. It is recognized that corrective measures are in progress and that a major effort has been made to strengthen the quality assurance (QA) corporate organization and its control over site related QA activities. However, to date quality related concerns continue to exist in the operations and maintenance areas.

B. Overall Facility Evaluation - Hatch 1 and 2

The licensee, as discussed in the previous SALP assessment, has continued to demonstrate strong corporate management support for, and commitment to, the improvement of overall plant performance; however, weaknesses were still evident. Major strengths identified during this assessment were in the areas of emergency preparedness, and security and safeguards. Major weaknesses were in the areas of plant operations, maintenance, and the quality assurance program. Corporate management involvement, management application of resources to resolve technical issues, and site management organizational and personnel changes have not been completely effective in correctly evaluating and solving

several problem areas (e.g., adherence to procedures, and identification and elimination of the "root cause" of problems). Overall plant operations have not continued to improve due to lack of attention to details and failure to follow procedures. Performance in the maintenance area has deteriorated, apparently due to inadequate procedures for performing maintenance and specifying appropriate post maintenance testing. The lack of effectiveness in quality assurance is exemplified by the failure to identify weaknesses in plant operations and maintenance.

C. Facility Performance - Hatch 1 and 2

Tabulation of ratings for each functional area:

Operations (Units 1 and 2)

1. Plant Operations - Category 3
2. Radiological Controls - Category 2
3. Maintenance - Category 3
4. Surveillance - Category 2
5. Fire Protection - Not Rated
6. Emergency Preparedness - Category 1
7. Security and Safeguards - Category 1
8. Refueling - Not Rated
9. Licensing Activities - Category 2
10. Quality Assurance Program - Category 3

D. Overall Facility Evaluation - Vogtle 1 and 2

The licensee continues to implement a vigorous construction project management effort with well qualified and experienced personnel. Major strengths were noted in the areas of safety related components and the quality assurance program. No major weaknesses were identified.

Although the number and severity levels of the violations in the areas inspected were significant, they do not indicate a programmatic breakdown; they appear to be a result of a failure to prepare adequate procedures to implement NRC requirements and licensee commitments.

E. Facility Performance - Vogtle 1 and 2

Tabulation of rating for each functional area:

Construction (Units 1 and 2)

1. Soils and Foundations - Category 2
2. Containment and Other Safety Related Structures - Category 2
3. Piping Systems and Supports - Category 2
4. Safety Related Components - Category 1
5. Support Systems - Not Rated

6. Electrical Power Supply and Distribution - Category 2
7. Instrumentation and Control Systems - Not Rated
8. Licensing Activities - Category 2
9. Quality Assurance Program - Category 1

F. SALP Board Members:

R. C. Lewis, Director, Division of Project and Resident Programs (DPRP) (Chairman), Region II (RII)
J. A. Olshinski, Director, Division of Engineering and Operational Programs (DEOP), RII
J. P. Stohr, Director, Division of Emergency Preparedness and Materials Safety Programs (DEPMSP), RII
H. C. Dance, Chief, Project Branch 2, DPRP, RII
E. G. Adensam, Chief, Licensing Branch 4, Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR)

G. SALP Board Attendees:

V. W. Panciera, Chief, Project Section 2B, DPRP, RII
M. V. Sinkule, Chief, Operational Support Section (OSS), DPRP, RII
R. V. Crlenjak, Senior Resident Inspector, Hatch, DPRP, RII
W. F. Sanders, Senior Resident Inspector, Vogtle, DPRP, RII
G. W. Rivenbark, Project Manager, Operating Reactors Branch 4, DL, NRR
M. A. Miller, Project Manager, Licensing Branch 4, DL, NRR
P. Holmes-Ray, Resident Inspector, Hatch, DPRP, RII
J. F. Rogge, Project Engineer, Project Section 2B, DPRP, RII
D. S. Price, Reactor Inspector, OSS, DPRP, RII
V. L. Brownlee, Chief, Project Section 2A, DPRP, RII
G. R. Jenkins, Chief, Emergency Preparedness Section, DEPMSP, RII
D. R. McGuire, Chief, Physical Security Section, DEPMSP, RII
F. Jape, Chief, Test Programs Section (TPS), DEOP, RII
H. L. Whitener, Reactor Inspector, TPS, DEOP, RII
T. C. MacArthur, Radiation Specialist, OSS, DPRP, RII
K. P. Barr, Chief, Facilities Radiation Protection Section, DEOP, RII
J. J. Blake, Chief, Materials and Mechanical Section, DEOP, RII
T. E. Conlon, Chief, Plant Systems Section, DEOP, RII
C. M. Upright, Chief, Management Programs Section, DEOP, RII

IV. PERFORMANCE ANALYSIS FOR HATCH UNITS 1 AND 2

...nal Area Evaluations

Licensee Activities

During the assessment period, major licensee activities at Hatch included refueling, torus modifications, recirculation piping weld overlay (units 1 and 2), and restoration of safety related cable trays.

Inspection Activities

During the assessment period, the routine inspection program was conducted by resident and regional inspection staffs. Special inspections were conducted of the full-scale emergency preparedness exercise, safety related cable tray restoration, an event involving the shut down of the unit 2 reactor by an unapproved method, and the monitoring of weld overlaying of units 1 and 2 recirculation piping.

1. Plant Operations

   a. Analysis

      During this evaluation period, inspections of plant operations were performed by the resident and regional inspection staffs.

      Events have occurred which indicated a need for improvement in the operations area. A significant event which illustrates the need for improvement was prompted by the licensee staff conducting a fast power reduction to preclude a reactor scram due to decreasing condenser vacuum. The on-shift operators and their supervisors recognized that the normal method of reducing power would not achieve a sufficiently timely power reduction to avoid a scram. These individuals and two shift technical advisers made a "consensus decision" to achieve a rapid power reduction by bypassing both the Rod Worth Minimizer and the Rod Sequence Control System, and by selectively scramming individual control rods, without an approved procedure, from the Scram Time Test Panel which is out of sight and out of normal voice communications with the reactor control console. The "consensus decision" and the resulting actions resulted in a control rod configuration that had not been analyzed from a reactor safety viewpoint.

      Additionally, poor performance in other plant activities has reflected negatively on the performance of operations. For instance, the long term deterioration of cable trays as a result of maintenance activities detected early in this assessment period and the misalignment of systems following a efueling outage indicate that operating personnel have not monstrated a control of activities involving maintenance.

Management involvement and control was increased during this assessment period. The licensee has implemented programs to improve procedural compliance and retrain operators in proper operational philosophy. An Operations Manager was added to the plant staff providing an additional level of management. Corporate management was frequently involved in site activities. The above listed measures have resulted in positive improvements; however, additional attention is required. Areas have been identified which require continued efforts to improve performance. Philosophy of operations should become a formal part of the Hatch operator training and requalification programs. Management effort in ensuring procedural compliance should be strengthened.

Decision making was usually at a level which ensured adequate management review. Resolution of technical issues was generally performed in a timely manner. Responsiveness to NRC initiatives was generally sound and thorough. Generally, acceptable resolutions were proposed.

Events were reported in a timely manner and contained a narrative section which provided a greater amount of information than was possible on the licensee event report form. However, as noted in the maintenance section of this assessment, the licensee's review of the event did not always identify the root cause. Thus, corrective action to prevent recurrence was often incomplete.

Several other areas have improved since the previous SALP assessment. New management techniques have been implemented which have improved shift-to-shift communications. Traffic flow through the control room has been reduced by the addition of an additional air lock. Outage management and control has improved through the implementation of a better and more comprehensive outage scheduling and tracking system.

Twenty-two senior reactor operator exams, twenty-one reactor operator exams and eleven instructor certification exams were administered. Seventy-seven percent of the senior reactor operator, forty-eight percent of the reactor operator and forty-eight percent of the instructor certification exams were passed. A special NRC training assessment identified some weakness in lesson plans, problem research techniques, and simulator lesson plans for initial training. These weaknesses have contributed to the significant failure rates identified above for reactor operators and instructors. Strong areas identified were in non-licensed operator training, lesson plans, simulator lesson plans for requalification training, and health physics and chemistry training.

Procedure violations continued to be a concern as shown by several of the violations listed below.

Seven violations were identified during the assessment period:

(1) Severity Level II violation for a reactor power reduction by means not analyzed in the Final Safety Analysis Report.

(2) Severity Level IV violation for failure to notify the NRC Operations Center of an event which required the initiation of the licensee's emergency plan or any section of that plan.

(3) Severity Level IV violation for failure to follow procedures concerning the use of adjustable wrenches to override spring loaded key switches.

(4) Severity Level IV violation for failure to follow procedures concerning valves not locked in position.

(5) Severity Level V violation for failure to follow procedures concerning a valve out of position and a valve not locked in position.

(6) Severity Level V violation for failure to follow procedures concerning an improper valve alignment.

(7) Severity Level V violation for failure to report within twenty-four hours to NRC when both fire suppression water supplies were below technical specification limits.

b. Conclusion

Category 3

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. While the board recognizes that management has been responsive to correcting deficiencies to the significant events that have occurred at Hatch, additional licensee management attention should be directed to this area and NRC inspection effort should also be increased.

_. Radiological Controls

a. Analysis

During this evaluation period inspections were performed by the regional inspection staff. This included a confirmatory measurements inspection using the Region II mobile laboratory and an environmental protection inspection.

Senior site management and corporate officials have exhibited a responsive awareness to radiological concerns and have initiated actions for improvements.

To strengthen the radiological safety program a Health Physics/Chemistry Manager position was established during this past SALP assessment period. Also, a technical group to support the health physics/chemistry operation was established and partially filled. The number of health physics personnel has been increased with emphasis being placed on company hires as opposed to contract personnel. Rigid qualifications were established for the hiring of contract personnel. The licensee was negotiating with a contractor to establish and coordinate the radiation protection controls and coverage of the forthcoming unit 2 outage to replace the recirculation piping.

A radioactive waste sorting and storage facility was completed during this assessment period. Assignment of personnel in the waste management area appears to be adequate to ensure proper management of radioactive wastes. Because of leaking fuel the unit 1 power level was reduced to approximately 70 percent of full power. A resequencing of the control rods was used to maintain radioactivity levels of gaseous effluents at acceptable values. The leaking fuel was replaced in November 1983. Effluent releases were considered normal for an operating plant of this type and rated capacity.

Licensee audits of the radiological control program have identified weaknesses and problem areas that needed attention. Audit reports showed that the licensee has been responsive and taken corrective actions to improve the program.

Radiation exposures to individuals have not exceeded regulatory limits. Total radiation exposure doses were directly related to, and consistent with, the outage work performed during the assessment period. Management controls and ALARA considerations have been effective in reducing exposures to individuals.

One quality control (QC) and confirmatory measurement inspection was performed during the evaluation period using the Region II Mobile Laboratory. With the exception of one violation (3 below), analyses between the licensee and NRC were in agreement. The inspection identified a need to determine the cause of a high systematic bias in tritium analysis of ground water samples. All other aspects of the laboratory program were satisfactory.

Three violations and one deviation were identified during the evaluation period. These were not indicative of significant programmatic deficiencies. Licensee management was adequately involved in radiological controls and was generally responsive to NRC concerns.

The following violations and deviation were identified:

(1) Severity Level IV violation for failure to conduct annual audits of the radioanalytical program to ensure conformance of facility operation to all provisions of the environmental technical specifications.

(2) Severity Level IV violation concerning the disposal of radioactive waste by an unauthorized method in that a container of compacted radioactive waste shipped to the Chem-Nuclear burial site contained free standing liquid.

(3) Severity Level IV violation concerning the use of an inaccurate computer software program for computing gaseous effluents.

(4) Deviation for failure to implement a commitment to develop and conduct annual QA audits of contractor activities related to the environmental monitoring program.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. No decrease in licensee or NRC attention in this area is recommended.

3. Maintenance

a. Analysis

During this evaluation period, inspections were performed by the resident and regional inspection staffs. Two special inspections were also conducted. One concerned the failure to restore safety related cable trays properly after maintenance, and a second involved the failure to restore systems to an operational status.

Overall performance in the maintenance area has deteriorated since the previous SALP assessment. Management review and attention should be increased to improve performance in adherence to procedures, problem area identification, proper assignment of functional testing following maintenance, proper identification of "root causes", and QC involvement.

Adherence to procedures was considered weak during the licensee's actions to repair reactor coolant recirculation piping cracking. Site personnel made the decision to deviate from procedures but did not change the procedures because the personnel considered the deviations to be technically acceptable.

In other cases, poorly stated or ill understood procedures for the control and maintenance of safety related equipment resulted in violations (2, 3, 4, 5, and 6).

During this evaluation period, the licensee demonstrated a slowness to identify problem areas with regard to cable tray restoration. This problem was identified during a resident inspection and was at first attributed by the licensee to be a localized housekeeping problem. Later, after additional concerns were expressed by the staff, the licensee realized the problem to be widespread and indicative of faulty maintenance practices. Once appropriate management attention was focused on the cable tray deficiencies the licensee was able to resolve the concerns with regard to operability of the cables during seismic or fire threatening events.

Inadequate functional testing was also illustrated by the condition of the cable trays as described above. In addition, functional testing was not adequate during startup of the plant following an extensive outage in that systems were returned to service without being fully restored to an operable condition (violations 2 and 3).

Recent reviews of LERs, related to maintenance activities have revealed problems in identifying specific "root causes". The corrective action for an event was usually limited to the actions taken to effect repairs and not to the root cause of the failure.

Involvement of the QA and QC organizations in maintenance activities was at times, inadequate, as evidenced by the cable tray restoration. QC involvement tended to be limited during the actual performance of maintenance.

Six violations were identified:

(1) Severity Level III violation and associated civil penalty for improper cable tray restoration.

(2) Severity Level IV violation for improper return to service of systems prior to restart of Unit 2.

(3) Severity Level IV violation for the improper return to service following maintenance on main steam isolation valve leakage detection system.

(4) Severity Level IV violation for failure to follow maintenance procedures during repair of the residual heat removal system.

(5) Severity Level V violation for improper procedure change being used for qualification of class 1 pipe welders.

(6) Severity Level V violation for failure to color code cables as required by maintenance procedures.

b. Conclusion

Category 3

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. Licensee management involvement in this area was acceptable, however, weaknesses were evident. Increased licensee management attention should be directed to this area and NRC inspection effort should also be increased.

4. Surveillance

a. Analysis

During this evaluation period inspections were performed by the resident and regional inspection staffs.

General Surveillance Activities

Overall, the surveillance program appeared to be an effective and smoothly operating program.

A major strength in the surveillance program was the effective use of a computer tracking system to ensure that surveillances were scheduled and performed as required.

Although the number of violations in this area has doubled since the last SALP assessment this is not indicative of a degradation of the overall program at Hatch. Two of the violations (2 and 3) were caused by failures to follow procedures or by personnel error, and were not attributable to an inadequate surveillance program or a programmatic problem. Corrective actions taken by the licensee should be effective in preventing recurrence of the violations.

In general, decision making was usually at a level which ensured adequate management review, audits were complete and thorough, and reviews were timely and technically sound.

One exception, however, was the changing of the surveillance test method for the automatic depressurization system (ADS). This change was processed by the engineering support staff and was reviewed and approved without the basic verification that it complied with the technical specifications (item 4).

Inservice Inspection

Licensee management involvement in inservice inspection (ISI) and inservice testing (IST) activities appeared to be adequate. Corporate management was usually involved in site activities. Reviews were adequate in timeliness, thoroughness and technical content. Records were generally complete, well maintained, and available. Procedures and policies were occasionally violated as evidenced by the violations below. Corrective action systems recognized and addressed most non-reportable concerns.

The licensee's performance involving IST was satisfactory as indicated by the small number of regulatory issues that were attributed to the licensee. This satisfactory performance is attributed to the timeliness and technically sound solutions offered by the licensee to NRC initiatives.

Resolution of technical issues sometimes lacked thoroughness or depth, and resolution was sometimes delayed as illustrated by one violation (5 below). This problem was first identified in March 1982, and was not resolved until May 1983. This was not repetitive and was not indicative of a programmatic breakdown. While key positions were identified, and authorities and responsibilities defined, the licensee's level III examiners need to be more involved in disposition of ISI findings.

Containment Leak Rate Testing

During the performance of local leak rate testing an inadequate surveillance procedure was used. The procedure failed to identify position of some valves involved in Type C tests as well as having several other inadequacies.

The weaknesses in this local leak rate procedure indicated at least in this instance, that the level of management involvement and control was not sufficient to assure a quality product. Further, the licensee's response to the violation resolving the quality of the Type C tests was marginally acceptable. The resolution was accepted after further communications with the licensee and consideration that an integrated leak rate test had been performed during the outage after the completion of Type C testing. The licensee's resolution to the technical issue did not indicate a sound, thorough, conservative, and timely approach to the issue.

The licensee and consultant (Bechtel) performed an adequate integrated leak rate test using an acceptable procedure. The licensee was two hours into the four hour stabilization period when increasing temperature in containment forced the licensee to adjust the containment back to test pressure. Concern was expressed during the test by the NRC inspector regarding a decision to not restart the four hour stabilization period. While the licensee did restart the stabilization period, the initial decision was not indicative of a sound and conservative technical decision.

Four violations were identified as follows:

(1) Severity Level IV violation for use of unapproved valve lineups for local leak rate testing.

(2) Severity Level IV violation for failure to take control room ventilation system filter samples when required

(3) Severity Level IV violation for failure to properly change a surveillance procedure.

(4) Severity Level IV violation for improper plant review board review which changed the ADS surveillance to a method different from that required by technical specifications.

(5) Severity Level V violation concerning failure to follow procedure for evaluation of ISI non-destructive examinations (liquid penetrant) indications.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. No decrease in licensee or NRC attention in this area is recommended.

5. Fire Protection

a. Analysis

During this assessment period, inspections were conducted by the resident inspection staff. During this period, certain weaknesses were noted as discussed in the maintenance section. These involved a number of cable trays which were not adequately protected from fire because of improper restoration following maintenance in that protective jackets and fire stops were not reinstalled. The large number of deficiencies found should have been identified by site personnel who were tasked with fire protection responsibilities.

One procedure adherence problem led to a violation when fire protection personnel did not provide a deviation report concerning the inoperability of a fire protection system. This report would have caused the operations staff to post a fire watch as required by technical specifications.

One violation was identified as follows:

Severity Level IV violation for failure to establish a fire watch within one hour.

b. Conclusion

Not Rated

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. There was insufficient inspection activity in this area, during this assessment period, to justify a rating. It should be noted that certain post-maintenance fire protection restorations were not adequate which resulted in system inoperability and that this condition was not detected by the plant staff. No decrease in licensee or NRC attention in this area is recommended.

6. Emergency Preparedness

a. Analysis

During the evaluation period, inspections were performed by the resident and regional inspection staffs. These inspections included evaluation of two full-scale exercises (December 1982 and October 1983), and one routine inspection (July 1983). There were no violations or deviations identified.

Subsequent to the 1982 emergency exercise, the NRC issued a letter to Georgia Power Company acknowledging the substantial improvement in the Hatch emergency preparedness program as demonstrated by that exercise, as well as the innovative use of the plant simulator in the exercise.

An emergency plan deficiency was identified during the routine inspection in July 1983. The plan deficiency was significant in that the licensee's emergency plan and implementing procedures did not adequately address general emergency protective action recommendations based on plant accident conditions prior to any substantial release of radioactivity.

Of four inspector follow-up items identified during the exercise in October 1983, one involved the readability of meteorological instruments in the control room. The licensee had identified to NRC a prior open item on the issue as complete and ready for inspection. However, the NRC found that the problem had not been solved, and supported the finding by requesting a trained operator to make readings. The operator made several errors which were attributed to the equipment.

With the exceptions noted above regarding the meteorological instrumentation, the licensee's approach to the resolution of technical issues was quite thorough. The licensee generally evaluated each NRC identified problem from the perspective of identifying and resolving the underlying cause.

As in the previous year, the 1983 Hatch exercise was fully successful. In the months prior to the exercise, the licensee devoted special effort through meetings, training, and resource assistance to the state and counties to assist in resolving off-site issues that were identified during the 1982 exercise.

The licensee has been responsive to expressed NRC concerns and has taken generally prompt action on all open items. There was definite evidence of management commitment to the emergency preparedness program. For example, for the October 1983 exercise, the scenario was exceptionally detailed and contained contingencies for possible unplanned events. The licensee also made a large commitment to training and to providing personnel for control and evaluation of the exercise.

b. Conclusion

Category 1

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. Licensee management attention and involvement in this area were aggressive. No decrease in licensee or NRC attention is recommended.

7. Security and Safeguards

a. Analysis

Inspections were performed by the resident and regional inspection staffs.

Corporate and site managements' support and security awareness was positive as indicated by their professional approach to providing a safe and secure onsite environment; their responsiveness to all NRC comments and discussions; and the non-adversary relation with onsite personnel. The proprietary security guard force was adequately staffed to meet all commitments of the security plan and of the contingency plan. Review of the training and qualification plan, observations of on-the-job training, and interviews with security force personnel indicated that security training, as programmed,

was being efficiently and effectively implemented. This was also demonstrated by the positive morale of the security force.

The licensee had instituted an intensive drug and alcohol abuse prevention program for all employees, with initial attention given to employees and contractors at the nuclear facility. This self-intiated effort exceeded proposed NRC criteria.

While the identified violations reflected continued deficiencies in the area of access controls to the facility, the licensee has taken strong measures to prevent security personnel error, provide improved procedures, and renovate access portal hardware. During the most recent security inspection the issue of access controls received extensive review and resulted in no violations being identified.

The licensee provided prompt and thorough corrective actions to the violations and all identified technical issues raised during security inspections. These violations were not indicative of the total effectiveness and proficiency of the security program at the Hatch plant.

Two violations were identified:

(1) Severity Level IV violation for an authorized employee wearing the wrong picture badge inside the protected area.

(2) Severity Level IV violation for escorts not maintaining contact with visitors while within the protected area.

b. Conclusion

Category 1

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. Licensee resources were effectively used in this area such that a high level of performance was achieved. No decrease in licensee or NRC attention is recommended.

8. Refueling

a. Analysis

Unit 1 and unit 2 refueling operations were inspected by the resident inspection staff, however, no indepth review of the refueling program was conducted. No violations or deviations were identified.

During the unit 2 refueling operating problems were discovered by the licensee with the refueling bridge, grapple, and mast. Corrective action was prompt and adequate. Problems were also encountered when rigging the steam separator back into the reactor vessel, causing some damage to the separator and the vessel. Repairs were made and reassembly of the vessel was completed with no further problems.

Refueling personnel demonstrated appropriate response during a fuel misorientation problem. Refueling activity was stopped at the first point which allowed for identification. Management review identified the root cause to be a lack of QC coverage in the area of computer generated loading patterns.

b. Conclusion

Not Rated

c. Board Comments

There was insufficient inspection activity in this area to justify a rating.

9. Licensing Activities

a. Analysis

The assessment was based on an appraisal of the following significant licensing activities:

- Appendix R activities
- MK 1 torus modifications
- Equipment environmental qualification
- Safety Relief (S/R) valve failure evaluation
- Reload reviews
- Scram discharge volume system modifications
- Appendix I activities
- Purge and vent system modifications
- Pipe crack inspection and repair

- Low low set points modifications
- Inservice inspection program
- Control of heavy loads activities
- Missing pipe whip restraints
- Temporary technical specification change to lower reactor low water limit (emergency basis)
- Temporary technical specification change to extend allowable time to inert containment (emergency basis)
- Temporary order change related to leak detection requirements (emergency basis)

Management involvement in assuring quality varied. For the licensing actions in the above list, while there were some exceptions, there was usually evidence of prior planning and assignment of priorities, and the reviews were generally timely, thorough, and technically sound. An area that needed improvement, however, was in the preparation of evaluations supporting the no significant hazards consideration determination. These determinations have been required only since May 1983 and everyone is on a learning curve. However, the quality and thoroughness of these determination evaluations are lower than the quality of the balance of the submittals.

The licensee has demonstrated an understanding of the safety consequences of technical issues and has generally provided acceptable and conservative resolutions to problems. Resolutions were sometimes slowed by the need to include the inputs of Southern Company Services and Bechtel, whose offices are geographically separate from the licensee's corporate headquarters, and the scheduling of reviews of items by the site review committee.

The licensee's responsiveness to NRC initiatives was adequate. Generally timely, viable, sound, and thorough responses were provided.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 1 during the previous SALP assessment. No decrease in licensee or NRC attention in this area is recommended.

10. Quality Assurance Program

a. Analysis

During this evaluation period routine inspections were performed by the resident and regional inspection staff:

Significant corporate QA organizational changes were implemented during the assessment period. The licensee created and filled a new position, General Manager, Quality Assurance and Radiological Health and Safety (GMQA). The GMQA is responsible to the Executive Vice President Power Supply for the overall control of the licensee's QA program. Further reorganization details depict the following corporate level QA personnel reporting directly to GMQA: the Plant Hatch QA Manager (HQAM), the QA Engineering/Support Manager (QAE/SM), the QA Coordinator for Fossil and Hydro Projects, the QA Special Projects Assistant (QASPA), and a Radiological Health and Safety Representative (RHSR). The RHSR has no line QA functions. The QA Site Manager (QASM) located at the plant site reports to the HQAM.

The new QAE/SM position was established to support the Hatch QA programs by providing increased participation in solving quality-related problems, increased oversight of architectural/engineer (A/E) and contractor QA activities, regulatory and associated document review, assessment of trends, and procurement QA activities.

The QASPA position was created to develop and direct QA training programs for organizations performing quality-related activities. Additionally, the QASPA will direct or accomplish special QA projects as directed by the GMQA.

The above mentioned organizational changes have been beneficial to the QA program. Under this revised organization, identical QA manager positions have been established for both Vogtle and Hatch making them solely responsible for the QA programs of their respective projects subordinate to the GMQA. The creation of an QAE/SM and staff appears to be a major improvement in that it strengthens the licensee's QA capability to assess their program and should provide increased QA oversight of A/E design activities (oriented more towards technical/engineering review versus the usual program compliance verification approach), suppliers, and vendors. Additionally, it appears that the licensee has appointed a strong GMQA who possesses valuable and broad experience in nuclear and QA activities. The GMQA appears dedicated to strengthening and upgrading the licensee's overall QA program and has necessary management support and attention in this endeavor.

Licensee quality assurance policies were adequately stated and understood. The licensee revised the QA procedures for pursuing and resolving problems identified by NRC and QA procedures related to audit finding categorization. The licensee also issued new procedures for conducting QA surveillance activities and for annual QA Department assessments of line organization performance.

Decision making appeared to be at a level that ensured appropriate management review. Corporate management was closely involved in site activities. Audits were generally complete and timely; however, corrective actions for audit findings were inadequate. This problem had been recognized by upper licensee management and newly created QA staff positions were implemented to provide timely resolution of audit findings. Establishing this new system received total upper management support. The licensee QA staff categorized all previously identified audit findings according to their safety significance and firm commitments for their resolution were established.

Records were generally complete and available for review. A new record vault has been proposed and is due for construction during 1984. Procurement activities were generally well controlled and documented.

Design change activities met regulatory requirements; however, design change procedures are difficult to use. A backlog of design changes has accumulated for which documentation was being gathered to permit closeout. The engineering staff responsible for design changes was being supplemented by consultants to assist in design change closure. Progress had been achieved. Licensee management should continue their efforts to close out completed design changes and consider upgrading existing design control procedures.

The licensee's responsiveness to NRC initiatives was technically sound. Their response to inadequate corrective actions on previously identified QA staff audit findings was vigorous and well managed. Corrective actions were handled by the regulatory compliance group; however, this system had not been totally tested under heavy workload conditions. Staffing appeared to be adequate, although some QA personnel on site were relatively new. These personnel were being effectively trained but it will take time for positive results. Plant personnel were being rotated into the QA department to provide needed expertise in various areas. One qualified senior reactor operator on loan to the QA Department was being trained as a lead auditor.

While improvements have been made in the QA corporate structure and in strengthening the on site QA organizations and functions, quality related deficiencies continue to be noted in the operations and maintenance areas. The extent of these deficiencies as described in the operations and maintenance sections of this report indicates that the Quality Assurance Program has been ineffective in identifying weaknesses in these areas, and as a result quality related deficiencies have been allowed to remain unnoticed by plant management until a significant event has occurred. The QA program should increase the scope and depth of audits in these areas to identify problems with the effectiveness of QC, operations, and maintenance organizations, and provide suitable corrective actions. Management effort will be necessary to increase the level of thoroughness of the audits.

The following violations were identified:

(1) Severity Level IV violation for failure of the QA Department to assure conditions adverse to quality are promptly corrected. This was a repeat violation.

(2) Severity Level IV violation for failure to respond to audit finding within required timeframes.

(3) Severity Level V violation for failure to annually update qualification records of lead auditors.

b. Conclusion

Category 3

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. It appears, on the surface from a review of the QA program and its implementation at the Hatch facility, that the program is effective. However, when an overall evaluation of the facility's history for this period is conducted, it becomes readily apparent that the implementation of the QA program at Hatch is inadequate in identifying problems and/or ineffective in bringing about adequate corrective actions. Specific issues which are examples of these problems includes inadequate procedures, Type C testing, a high operator examination failure rate, independent verification, cable tray restoration, and a high component failure rate. Increased licensee management attention is required in this area to assure that licensee

personnel are effective in performing the QA functions as required by NRC regulations. Increased NRC inspection effort should be directed to this area.

B. Supporting Data

1. Reports Data

a. Licensee Event Reports (LERs)

During the assessment period, there were 111 LERs reported on unit 1 and 134 on unit 2. The distribution by Licensee Cause Code and SALP Functional Area is as follows:

Cause Code	Unit 1	Unit 2
Personnel Error	12	24
Design, Manufacturing, Construction/Installation	5	5
External Cause	0	1
Defective Procedures	6	3
Component Failure	64	77
Other	24	24
Total	111	134

Functional Area	Unit 1	Unit 2
Plant Operations	50	64
Radiological Controls	2	1
Maintenance	4	3
Surveillance	44	43
Fire Protection	2	3
Security	1	0
Refueling	3	2
Quality Assurance	4	15
Other	1	3
Total	111	134

Of the 111 LERs submitted on unit 1, 58% were due to some kind of component failure. Of these failures, 42% were mechanical, 45% were electrical, and the remaining were attributed to other miscellaneous causes. There were 134 LERs submitted for unit 2 during the evaluation period, of which 57% were due to component failure. Of these failures, 37% were mechanical and 45% were electrical. The remainder were assigned various miscellaneous causes. Although not a regulatory requirement, it was noted that the licensee does not report component failures to the nuclear plant reliability data system.

b. Part 21 Reports

None

2. Investigation and Allegation Review

One allegation involving radiological sampling techniques was closed during the assessment period. The allegation was not substantiated.

3. Enforcement Actions

a. Violations

Severity Level I - 0 violations
Severity Level II - 1 violation
Severity Level III - 1 violation
Severity Level IV - 18 violations
Severity Level V - 7 violations

b. Civil Penalties

Three violations classified as a single Severity Level II event with a civil penalty were assessed on December 27, 1983, for a July 14, 1983 improper shutdown of the unit 2 plant.

One violation classified as a Severity Level III with a civil penalty was assessed on June 2, 1983, for failure to provide required quality controls following modification and maintenance relating to electrical cable trays.

c. Orders

No orders relating to enforcement matters were issued.

d. Administrative Actions - Confirmation of Action Letters

No Confirmation of Action Letters were issued during this review period.

4. Management Conferences

Two management meetings were conducted on March 18 and 22, 1983, to discuss the quality control problems associated with the modification and maintenance of electrical cable trays.

A management meeting was held on April 29, 1983, to discuss the status of the recirculation system weld inspections.

An enforcement conference was held on July 21, 1983, to discuss the management control over reactor operation pertaining to the July 14, 1983 improper shutdown of unit 2. Subsequent enforcement conferences were held by the Director of Inspection and Enforcement on November 2 and 14, 1983 to discuss the significance of the event with all personnel involved.

An enforcement conference was held on August 10, 1983, to discuss the adequacy of startup preparations for unit 2.

A management meeting was held on September 28, 1983, to brief NRC on the scheduling of the Hatch outage to support the replacement of recirculation system piping on unit 2.

V. PERFORMANCE ANALYSIS FOR VOGTLE UNITS 1 AND 2

A. Functional Area Evaluations

Licensee Activities

Between November 1, 1982 and October 31, 1983, the construction project progressed from 37% to 52% completion. Unit 1 and common progressed from 43% to 61%. Unit 2 was 20% complete. Staffing for the project has gradually increased to the present level of manpower.

Work on the containment buildings continued to progress. In the unit 1 containment building, the first dome placement was completed on October 29, 1983, and continued to be erected at the required rate. In the unit 2 containment building, interior concrete placement continued, with the secondary shield and refueling canal walls being completed to the 217 foot elevation.

Progress in the other power block buildings continued. In the auxiliary building, large bore piping and hanger installation progressed on levels A, B, C, and 1, and the erection of small bore piping and hanger continued in all available areas. Work in the fuel handling building continued, with heating, ventilation and air-conditioning (HVAC) duct installation progressing on levels 1 and 3 of the center section and on level 1 of the east wing. In the control building, civil activities continued to progress with the greater emphasis on work on the west wing level 1.

Turbine building work continued to progress. In the unit 1 turbine building, large piping was being erected on levels A, 1, and 2, and small bore pipe erection continued on level A. Work continued on the unit 1 turbine-generator installation schedule; the General Electric Company is expected to start the main turbine work in February 1984.

Inspection Activities

Routine inspection programs were performed during this evaluation period. The Regional Construction Assessment Team conducted an indepth review of the site project management.

1. Soils and Foundations

   a. Analysis

      Inspections were performed by the resident and regional inspection staffs. The NRC reviewed quality assurance implementing procedures, observed back fill operations, examined calibration controls on soil testing equipment, and made observations of concrete placement.

Work activities were performed in accordance with procedure requirements and testing was being done with equipment having current calibration data. Discussions with QC inspectors indicated that the inspectors were knowledegable in specification and procedure requirements and were documenting their inspections on applicable documents.

Management involvement, resolution of technical issues, and staffing were adequate for the level of activity involved.

No violations or deviations were identified.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 2 during the pervious SALP assessment. No decrease in licensee or NRC attention in this area is recommended.

2. Containment and Other Safety Related Structures

a. Analysis

Inspections were performed by the resident and regional inspections staffs. The inspections involved review of QA implementing procedures; observation of work activities including containment structural steel; containment concrete; rebar installation; layout of walls and piping lines; grounding cable; embed plates; review of quality records; observation of polar crane support installation; lifting and setting of containment dome; and review of spare penetrations.

Management involvement, resolution of technical issues, staffing, and training were adequate for the level of activity involved. Violations involving concrete placement and installation of structural steel were identified. The violations were not indicative of a programmatic breakdown, but were a result of failure to prepare adequate procedures to implement licensee commitments. A similar problem regarding inadequate procedures was identified in the previous SALP evaluation. With the exception of the identified violations, QA and QC procedures and controls were found to meet NRC requirements and work activities were found to have been performed in accordance with QA and QC procedure requirements. QA records were generally complete, well maintained, and retrievable. The licensee was responsive in correcting the violation concerning the concrete placement.

Response to the violation concerning structural steel was inadequate in that it did not address all of the examples cited in the violation. The licensee was preparing a supplemental response on this item at the close of the assessment period.

The following violations were identified:

(1) Severity Level IV violation for failure to implement procedures and drawings or provide acceptance criteria for bolting, modification or removal of structural steel, and verify that expansion anchors and embed plates complied with design requirements.

(2) Severity Level V violation for failure to provide proper cold weather protection on a concrete placement.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 1 during the previous SALP assessment. No decrease in licensee or NRC attention in this area is recommended.

3. Piping Systems and Supports

a. Analysis

Inspections were performed by the resident and regional inspection staffs. Additionally, a major inspection by the Regional Construction Assessment Team was also performed.

The licensee was in the early construction phases for the installation of safety-related supports. The NRC inspection identified a weakness in that instructions were not adequate with respect to weld size, dimensions, and tolerances. These problems were identified for four pipe hangers which had been inspected and accepted by the hanger QC inspectors. The licensee was very responsive to this concern and has taken appropriate action to correct the problem. In addition to correcting the hardware, the licensee conducted an extensive retraining program for QC inspectors, craft personnel, engineers, and supervisors involved with support work. This corrective action demonstrated a thoroughness on the part of the licensee to correct the problem while still in the early stages of construction.

Another activity relating to piping systems was the reinspection of approximately 15,000 piping spool pieces that had been fabricated by Pullman Power Products. The reinspection was needed to ascertain the acceptability of fabrication welds after code rejectable deficiencies had been found in a sampling of spool pieces stored at the plant site. The extensive reinspection program was handled in a thorough manner, and resulted in the satisfactory resolution of a generic quality problem.

A strength of the licensee's program appeared to be in the arrangement for the use of the Westinghouse Vogtle Structural Analysis Mobile Unit (V-SAMU). The V-SAMU was to be used for analysis and design of class 2, 3 and non-nuclear small bore piping systems. This on-site capability will expedite resolution of many engineering or construction problems as they are identified.

The procedures for control of construction activities appeared to be adequate to insure a quality product in the area of supports and restraints.

Practices used in welding large diameter reactor coolant loop piping appear excellent as are the quality of the welds being produced. The practices and procedures for welding on other piping and piping supports are adequate.

One violation was identified:

> Severity Level IV violation concerning a failure to provide adequate instructions, procedures and drawings with respect to welding, dimensions, and tolerances for pipe supports.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. No decrease in licensee or NRC attention in this area is recommended.

4. Safety Related Components

a. Analysis

Inspections were conducted by the resident and regional inspection staffs. These inspections involved the preparation and setting of the reactor pressure vessel and steam generators; storage of safety-related equipment; load test of the containment dome lifting equipment, and non-destructive examination of the reactor pressure vessel closure head cladding.

The procedures and controls utilized to perform the movement and placing of various large pieces of safety-related components provided evidence of prior planning and assignment of priorities. The proper precautions, commensurate with the potential damage, were in place at the time of the move. The licensee incorporated the lessons learned from another construction site which had dropped a piece of equipment being moved.

The licensee demonstrated a technically sound and thorough response to the violation involving the reactor vessel closure head cladding. A complete reinspection of the head cladding has been completed and the repairs are planned. The resolution of this problem has been timely due to the adequate staff attention which the licensee placed on this item.

The violations were not considered to be indicative of programmatic breakdowns due to licensee evaluations that these were isolated cases.

Two violations were identified:

(1) Severity Level IV violation concerning failure to provide for code required penetrant examination of a completed weld.

(2) Severity Level V violation concerning the proper performance of liquid penetrant examination on the reactor vessel closure head.

b. Conclusion

Category 1

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. Licensee resources were effectively used such that a high level of performance was achieved. No decrease in licensee or NRC attention in this area is recommended.

5. Support Systems

a. Analysis

During this evaluation period, resident monitoring of the activities in correcting the licensee identified problem with the Heating, Ventilation and Cooling Systems (HVAC) concerning the duct work supports was conducted.

Licensee management has demonstrated excellent involvement and control in resolving the HVAC problem where the contractor was not following the prescribed installation procedures, and in ambiguities between the engineering drawings and fabrication drawings. The licensee identified this problem during a corporate quality assurance audit and has since determined the root cause to be a quality assurance program breakdown at the fabrication shop for the duct supports. The licensee has also taken appropriate action to prevent a recurrence by designating the architect/engineer to perform a review of shop detail drawings for conformance to design drawings. The licensee completed an evaluation and finalized a report in a timely fashion which will prevent duct support as-built conditions, shop drawings, and engineering design drawings from being in disagreement.

No violations were identified in this area.

b. Conclusion

Not Rated

c. Board Comments

There was insufficient inspection activity in this area to justify a rating.

6. Electrical Power Supply and Distribution

a. Analysis

During this evaluation period, inspections were performed by resident and regional inspection staffs. The areas inspected included electrical component receipt, storage and handling; in-place storage of electrical equipment; quality assurance; followup on the licensee's inspection of control panel weld problems; and training, qualification and certification of technical inspectors.

Management involvement to assure quality has been evident on two issues regarding electrical components. The first issue, pertaining to welds in the control panels, was quickly expanded in scope when the inspection identified problems with component identity, incorrect wiring, and questionable qualification of connectors. It should also be noted that concern for the welding on control panels was transmitted in IE Notice 82-34, Revision 1. In addition to this review the licensee inspected electrical terminations. This resulted in a full inspection, a considerable amount of field correction of equipment by the vendor, and the issuance of a Part 21 report.

The second issue was in the area of in-plant storage of electrical components. NRC issued a violation concerning the storage of equipment with little protection. The licensee response to this issue was very prompt and extensive. The corrective action consisted of a complete inspection of the equipment, the installation of protective covering and in some instances, the erection of cages to prevent unauthorized access. The one weakness on the part of the licensee was in the area of self-identification of the problem. In this regard, the indications of the problem had been identified by QC inspectors, but the licensee did not recognize the magnitude of the problem. The storage of electrical equipment in the warehouse was found to not have the same problems as identified for the in-plant storage.

Electrical installation of cable was in the very early stages and inspections have been correspondingly limited. The licensee appeared to be well organized and prepared to commence cable pulling. The start of cable pulling has been delayed due to improper Uni-strut support locking fasteners. The extensiveness of the fastener problem was indicative of a failure of the QC inspection procedures for the Uni-strut fasteners. The licensee identified the problem and is now performing complete reinspection.

The following violation was identified:

- Severity Level IV violation concerning the in-place electrical cabinets not being adequately protected for dirt, moisture, vandalism, and rodents.

b. Conclusion

Category 2

c. Board Comments

No decrease in licensee or NRC attention in this area is recommended.

7. Instrumentation and Control Systems

a. Analysis

No inspections were performed in this area during the evaluation period due to the early stage of facility completion.

b. Conclusion

Not Rated

c. Board Comments

There was insufficient inspection activity in this area to justify a rating.

8. Licensing Activities

a. Analysis

The evaluation was based on NRC evaluation of the following licensing activities:

- Category 1 Compaction
- Caseload Forecast
- Content of the Final Safety Analysis Report
- Content of the Environmental Report

Management involvement was evident in assuring quality based on a very favorable impression made by licensee management at the Caseload Forecast Panel (CFP) site visit and subsequent meetings with NRC staff. High levels of management were represented at the CFP visit. The individuals in attendance were very knowledgeable about the Vogtle project and they

appeared to place appropriate emphasis on assuring quality at the plant.

The licensee's approach to the resolution of technical issues from a safety standpoint appeared adequate. This conclusion was based on resolution of compaction of Category 1 backfill around safety-related piping. The licensee, once staff concerns were identified, addressed them in a timely manner. After discussions on the compaction issue, the licensee proposed a satisfactory solution which accounted for staff safety concerns.

The licensee was prompt and very responsive to NRC inquiries, particularily offering cooperation and information on the compaction issue when the review required several telephone conversations and supplemental submittals. However, this approach was typical of the licensee's response on most licensing issues.

The licensee appeared to be technically competent based on the involvement with the licensee's staff at the Caseload Forecast Panel visit and on the compaction issue. The staff appeared technically competent with the appropriate persons involved on both issues.

The NRC staff review of the content of the Final Safety Analysis Report (FSAR) and the Environmental Report (ER) indicated that the information provided did not provide enough detail, in some instances, to address a topic adequately. In the ER, for example, the licensee's frequent reference to the FSAR sometimes hindered the review.

This assessment area was limited due to the early licensing review stage of Vogtle. On the selected activities, the contact and involvement has been very slight and does not provide a basis for an overall detailed evaluation. For typical licensing activities, such as the caseload and the compaction issue, the licensee's performance has been more than adequate. However, the content of the FSAR and ER needs upgrading before the staff can adequately review the plant.

b. Conclusion

Category 2

c. Board Comments

Performance in this area was evaluated as Category 1 during the previous SALP assessment. The rating during the present SALP assessment period was based upon limited activity in the area.

9. Quality Assurance Program

a. Analysis

Inspections were performed by the resident and regional inspection staffs. A special team inspection was conducted to assess overall management control of the Vogtle project.

Significant corporate QA organizational changes were implemented during the assessment period. The licensee filled a new position, General Manager, Quality Assurance and Radiological Health and Safety (GMQA). The GMQA is responsible to the Executive Vice President Power Supply for the overall control of the licensee QA program. Further reorganization details depict the following corporate level QA personnel reporting directly to GMQA: the Plant Vogtle QA Manager (VQAM), the QA Engineering/Support Manager (QAE/SM), the QA Coordinator for Fossil and Hydro Projects, the QA Special Projects Assistant (QASPA), and a Radiological Health and Safety Representative (RHSR). The RHSR has no line QA functions.

The QA Site Manager (QASM) located at the plant site and a Project QA Engineer report to the VQAM. The VQAM has three QA engineers assigned to perform engineering evaluations on QA activities.

The new QA Engineering/Support Manager position was established to support the Vogtle QA programs by providing increased participation in solving quality-related problems, increased oversight of architectural/engineer (A/E) and contractor QA activities, regulatory and associated document review, assessment of trends, and procurement QA activities.

The QASPA position was created to develop and direct QA training programs for organizations performing quality-related activities. Additionally, the QASPA will direct or accomplish special QA projects as directed by the GMQA.

The above mentioned organizational changes have been beneficial to the QA program. The job title of Manager to the GMQA, VQAM, and QASM has upgraded the QA organization image in the eyes of construction and licensee management. Under this revised organization, identical QA manager positions have been established for both Vogtle and Hatch making them solely responsible for the QA programs of their respective projects subordinate to the GMQA. The creation of a QAE/SM and staff appears to be a major improvement in that it strengthens the licensee's QA capability to assess their program and should provide increased QA oversight of A/E

design activities (oriented more towards technical/ engineering review versus the usual program compliance verification approach), suppliers, and vendors. Additionally, it appears the licensee has appointed a strong GMQA who possesses valuable broad experience in nuclear and QA activities. He appears dedicated to strengthening and upgrading the licensee's overall QA program and has necessary management support and attention in this endeavor.

There was evidence that licensee management has re-examined the QA program, worked toward upgrading standards, obtained better qualified personnel, and in general promoted QA acceptance at all levels. The GMQA presented to the licensee's management an assessment audit of the QA Department operations identifying particular concerns and needed improvement areas for which resolutions were proposed and management responded with affirmative support.

Management periodically reviewed the QA program. The design assurance audits, the audit plan, the audit followup of corrective actions for the audit findings, and the tracking of Bechtel, licensee and NRC audit findings were reviewed and were generally complete and thorough.

The licensee's responsiveness to NRC initiatives was considered adequate. The construction quality assurance program update was submitted as required by 10 CFR 50.55(f), a new regulation.

There was evidence of prior planning, assignment of priorities, and defined procedures used to control activities. QA policies were adequately stated and understood.

The procurement activities were reviewed and found well controlled and documented. There has been no indications of QA programmatic breakdowns.

Staffing of QA positions appeared to be adequate. Key positions were identified, and authorities and responsibilities were defined. Management independence was retained. QA staffing has increased with the expanded workload.

The training and qualification program contributed to an understanding of work and a reasonable adherence to procedures. A defined program was being implemented.

Staffing has been adequate for the status of construction. Positions were clearly identified with authorities and responsibilities well defined. Personnel were adequately trained to understand the authority and responsibilities of their positions.

During this reporting period, the onsite QA staff has been increased from 14 to 19 persons.

The licensee has demonstrated sound technical decision making commensurate with quality assurance concerns. This was best exemplified by the licensee response to IE Bulletin 82-01. This bulletin pertained to problems where two vendors knowingly supplied altered radiographs. The licensee expanded the scope of the bulletin by performing an independent review of the radiographs for all of the shop fabricated welds for the components furnished by six other vendors. This inspection has identified numerous nonconformances with these radiographs. The licensee was working with the vendors to resolve the issue. This action was typical for the Vogtle project, where the specifics of a problem were expanded in a generic fashion to assure that a problem did not exist in related areas.

Two violations were identified:

(1) Severity Level V violation concerning the failure to properly store radiographic records.

(2) Severity Level V violation concerning the failure to review and approve weld acceptance criteria.

b. Conclusion

Category 1

c. Board Comments

Performance in this area was evaluated as Category 2 during the previous SALP assessment. Licensee mangement involvement in this area was aggressive. No decrease in licensee or NRC attention is recommended.

B. Supporting Data

1. Reports Data

a. Construction Deficiency Reports (CDR)

During the period, seventeen CDRs were reported. Eight were caused by errors associated with the design of the component; nine were due to manufacturing/fabrication errors. Three additional items were considered and later determined to be not reportable.

The reports were reported in a timely manner. The telephone reports provided sufficient information; however, the initial written reports could provide a more detailed description of the problem. This will provide NRC with sufficient information to allow evaluation. The events are properly identified and analyzed. Corrective action was effective as indicated by a lack of repetition.

2. Investigation and Allegation Review

Five allegations were closed during this assessment period. Three involved facility construction deficiencies and were unsubstantiated; one involved equipment storage and resulted in the issuance of a violation; and one involved the use of uncertified construction materials and, although substantiated, did not involve a violation of NRC regulation.

3. Enforcement Action

a. Violations

Severity Level I, II, III - No violations
Severity Level IV - 4 violations
Severity Level V - 4 violations

b. Civil Penalties

None

c. Orders

None

d. Administrative Action

No Confirmation of Action Letters were issued during this review period.

4. Management Conferences

A management meeting was held on January 14, 1983, to discuss the results of the Institute of Nuclear Power Operations related utility Self Evaluation.

A management meeting was held on March 2, 1983, to discuss field change controls.

A management meeting was held on June 24, 1983, to discuss quality workmanship by field contractors.

A management meeting was held on August 22, 1983, to discuss the licensee findings concerning quality workmanship by field contractors.

No enforcement meetings were conducted.