April 26, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	)	
SOUTH CAROLINA	)	Docket No. 50-395 OL
ELECTRIC & GAS COMPANY,	)	
et al. (Virgil C. Summer	)	
Nuclear Station, Unit 1)	)	

APPLICANTS' RESPONSE IN OPPOSITION TO "PETITION TO INTERVENE AND REQUEST FOR HEARINGS" OF FAIRFIELD UNITED ACTION

On April 9, 1982, Fairfield United Action (FUA) filed a petition for leave to intervene as a party of record in this proceeding and requested that evidentiary hearings be conducted on the issues raised in its petition.

Almost a year ago, on June 3, 1981, the Licensing Board dismissed FUA as a party pursuant to ALAB-642 dated June 1, 1981. 1/ The Appeal Board decision was based on petitioner's unexcused lateness and lack of ability to contribute, and on the impact on the rights of the existing parties. The Appeal Board stressed the impact on the rights of the existing parties in its decision denying FUA's earlier petition.

1/ South Carolina Electric & Gas Co., et al. (Virgil C. Summer Nuclear Station, Unit 1), ALAB-642, 13 NRC 881 (1981), reversing South Carolina Electric & Gas Co., et al. (Virgil C. Summer Nuclear Station, Unit 1), LBP-81-11, 13 NRC 420 (1981), appeal pending sub nom. Fairfield United Action v. NRC, et al., No. 81-2042 (D.C. Cir.).

> "Simple fairness to [the existing parties] -- to say nothing of the public interest requirement that NRC licensing proceedings be conducted in an orderly fashion -- demand that the Board be very chary in allowing one who had slept on its rights to inject itself and new claims into the case as last-minute trial preparations were underway." ALAB-642, 13 NRC at 886. "Further, a delay in the progress of the proceeding is not merely a theoretical possibility but rather a very likely proximate result of the belated intervention [footnote omitted]." Id. at 888.

The Appeal Board recognized that granting FUA's petition at such a late date would have seriously impaired the existing parties' right to pretrial discovery and to invoke summary disposition procedures. Id. at 888-89. The Appeal Board concluded:

> "By ... remaining on the sidelines while the proceeding moved closer and closer to trial, it [FUA] voluntarily assumed the precise risk which has now materialized: that its participation in the proceeding could no longer be sanctioned without destructive damage to both the rights of other parties and the integrity of the adjudicatory process itself." Id. at 895.

Since that decision was rendered, 22 days of evidentiary hearings have been held, 6,137 pages of transcript compiled, 70 exhibits admitted, and on January 20, 1982, the record was closed as to all issues (Tr. 6137; see also Tr. 3871, 3872, 4677, 6014). We are now truly at the eleventh hour. In a briefing to the Commissioners on April 20, 1982, the Staff reported that the plant will be ready in May and that Board's initial decision is expected at the end of May.

(Briefing on Status and Assessment of Near-Term Operating Licenses, April 20, 1982, Tr. 29-30) The Applicants have an investment in the V.C. Summer Nuclear Station in excess of \$1.0 billion. It cannot reasonably be controverted that substantial carrying costs would be incurred by both owners if the plant were ready to load fuel and commence operation but was without a license; that both the State of South Carolina and the Federal Energy Regulatory Commission (FERC) provide for recovery of such costs from customers; and that South Carolina Public Service Authority would incur further additional costs for purchasing capacity and energy to replace Summer generation. Finally, it is reasonably ascertainable that energy from Summer would be cheaper from an incremental cost standpoint for SCE&G than oil. We explain later why there is no reason for a hearing on this matter; but focusing for the moment on the practicalities, no prompt hearing could be conducted without sacrificing trial preparation as discussed by the Appeal Board in Virginia Electric & Power Co. (North Anna Station, Units 1 & 2), ALAB-289, 2 NRC 395, 400 (1975). Delay would be measured in terms of months required for trial preparation, hearings, proposed findings, and decision.

Thus, the adverse impact on the rights of the parties and the very real consequences of delay are even more severe at this juncture than when the Appeal Board wrote on the matter last year.

Undaunted, FUA has filed an even more untimely petition. In its "Petition to Intervene and Request for Hearings" dated April 9, 1982, FUA again seeks to intervene and to have the hearings reopened to take up the matter of accelerated steam generator tube wear (FUA Petition, at 2). FUA urges that 1) despite the Applicants' commitment to implement preventive measures involving monitoring and inspection during initial limited power operation, the license should be either denied or unspecified conditions imposed to protect the health and safety of the public and 2) a new cost-benefit analysis should be prepared which assumes an indefinite 35% capacity factor and 57% of the projected kwh listed as benefits at page 9-1 of the Final Environmental Statement (FES). FUA also purports to incorporate by reference contention 14 on steam generators from its previously denied petition. Finally, FUA asks the Board take up these matters sua sponte should its petition be denied (FUA Petition, at 14).

At some point, there must be an end to litigation. 2/ As the Appeal Board pointed out last year in this case,

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2/ For a complete discussion see Cleveland Electric Illuminating Co., et al. (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 750 (1977); Houston Lighting and Power Co., et al. (South Texas Project Unit Nos. 1 and 2), ALAB-381, 5 NRC 582, 591 (1977); Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), ALAB-359 4 NRC 619, 620 (1976); Northern Indiana Public Service Co. (Bailly Generating Station, Nuclear-1), ALAB-227, 8

whether a subject should be treated as a hearing issue is not the same question as whether it deserves agency attention and action. ALAB-642, 13 NRC at 895-96. The NRC Staff and the Commission are well apprised of the accelerated tube wear experienced in preheater type steam generators at foreign units after periods of full power operation and are overseeing industry's program to remedy the problem. The Applicants are committed to operate the V.C. Summer facility initially at reduced power levels, to monitor and inspect for tube degradation, and to make modifications as indicated.

FUA, on the other hand, has given no indication as to what evidence it would offer, has shown no technical ability to contribute to resolution of this matter, and has

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AEC 416, 418 n.4 (1974); see Vermont Yankee Nuclear Power Corp. v. NRDC, 435 U.S. 519, 554-55 (1978); ICC v. Jersey City, 322 U.S. 503, 514, (1944); see also United States v. ICC, 396 U.S. 491, 521 (197[illegible]). In ICC v. Jersey City, supra, 322 U.S. at 514 the Supreme Court recognized:

> "Administrative consideration of evidence ... always creates a gap between the time the record is closed and the time the administrative decision is promulgated. This is especially true if the issues are difficult, the evidence intricate, and the consideration of the case deliberate and careful. If upon the coming down of the order litigants might demand rehearings as a matter of law because some new circumstance has arisen, some new trend has been observed, or some new fact discovered, there would be little hope that the administrative process could ever be consummated in an order that would not be subject to reopening."

not specified what if any, additional and/or alternative corrective actions should be taken or what license conditions it would impose. Given these circumstances, the key questions are: 1) Should the Licensing Board (as distinct from the NRC Staff and the Commission) take up this particular issue? 2) What purpose would be served by holding a hearing since the Applicants and Staff have matters well in hand? 3) What would be the impact on the rights of other parties? See discussion at pages 1-3, supra. The answers to these questions emerge from the familiar analyses of late intervention standards and are buttressed by consideration of the authorities regarding reopening.

Legal Standard Governing Petitions for Intervention

The standards governing late intervention are set forth at 10 C.F.R. § 2.714(a)(1). 3/ Five factors are balanced in deciding whether to grant or deny a late-filed petition: (1) good cause, if any, for failure to file on time; (2) availability of other means whereby the petitioner's interests will be protected; (3) extent to which the petitioner's participation may reasonably be expected to assist

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3/ Houston Lighting & Power Co. (Allens Creek Nuclear Generating Station, Unit 1), ALAB-671, slip op. (March 31, 1982); Project Management Corp., et al. (Clinch River Breeder Reactor Plant), ALAB-354, 4 NRC 383, 388-94 (1976). The Allens Creek case involves strikingly similar facts to the instant case, in that it involved a second petition by a person previously denied intervention.

in developing a sound record; (4) extent to which the petitioner's interest will be represented by existing parties; and, (5) the extent to which the petitioner's participation will broaden the issues or delay the proceeding. 10 C.F.R. § 2.714(a)(1)(i)-(v). FUA's petition refers to 10 C.F.R. § 2.714(a)(1) and touches upon the five factors but the overall showing is weak.

Good Cause. First, as to good cause, FUA argues that it only recently became aware of accelerated tube wear and leaks caused by flow-induced vibrations in Westinghouse Model D steam generators. It contends that prior to the Board Notification, BN-82-02, dated January 20, 1982, subsequent correspondence between the Applicants and the agency, NRC memoranda, and newspaper articles in the Columbia Record and The State it had no way of knowing about potential accelerated steam generator tube wear (FUA Petition, at 8-10).

In its petition to intervene filed March 23, 1981, FUA sought to raise a contention on steam generators (FUA Supplement to Petition to Intervene, Contention 14, March 23, 1981). 4/ Although that contention focused on occupational

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4/ In its latest petition to intervene, FUA attempts to incorporate by reference matters set forth in its original petition to intervene and supplement dated March 23, 1981. Such matters are not before the Board as Contention 14 was one of the group of 17 contentions initially denied by the Licensing Board (See Partial

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exposure to workers, 5/ it stated that "Westinghouse steam generators have demonstrated a generic tendency to denting, cracking, leaking and rupturing" and notes that extensive repairs, including replacement of the steam generators, have been required at some plants. (Id.) FUA's representative said the following about its contention 14 at the April 8, 1981 prehearing conference:

> "DR. RUOFF: given the history of Westinghouse steam generators, even recognizing that this D-3 Model is a model which was only in place at this plant and one in Sweden ... it is my understanding that the differences in the steam generator are not significant from other models. And given the proportion of Westinghouse steam generator models, which have had difficulty, this is something that can reasonably be expected to occur during the design life of the plant. And we would strive to get a witness to discuss those issues.

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Order Following Prehearing Conference, LBP-81-11, 13 NRC 420, 428 (1981)). FUA's inability to contribute to development of a sound record was weighed most heavily against it by the Board in rejecting these contentions. That inability to contribute has not been cured in this petition.

5/ The issue of occupational radiation exposure to nuclear plant workers has already been addressed by the Staff in the FES. The Staff determined that the Applicants are committed to design features and operating practices that will assure that individual occupational exposure is maintained within the limits of 10 C.F.R. Part 20 and ALARA. (FES, May 1981, at 4-23). Based on actual operating experience, average annual reactor doses of up to 1300 persons-rems over the life of the plant may be expected at Summer, but the actual doses vary greatly depending on factors such as the amount of required routine and special maintenance and the degree of plant operations (Id. at 4-23 to -24). Thus, the FES has already accounted for activities such as steam generator maintenance and modification in determining occupational exposure levels.

CHAIRMAN GROSSMAN: But you do not have one available now?

DR. RUOFF: No."
(Tr. 623).

As to the group of 17 contentions not admitted, including contention 14 on steam generators, the Licensing Board in its order following the prehearing conference stated, "we conclude that the good cause, delay, and ability-to-contribute to-a-sound-record factors weigh heavily against admission." LBP-81-11, 13 NRC 420, 427. As stressed by the Appeal Board in denying FUA's original petition to intervene, a petitioner who files at the 11th hour, has a very heavy burden to carry. ALAB-642, 13 NRC at 886,888. In this instance, as before, the good cause factor should be weighed against FUA.

Availability of Other Means and Representation by Existing Parties. Turning to the second factor, the availability of other means whereby the petitioner's interests would be protected, we will combine our discussion of that factor with the fourth factor, the extent to which the petitioner's interest will be represented by existing parties. FUA states at pages 10 and 12 of its petition that no existing parties will represent its interest and that no other forum exists to protect such interest. With regard to other available means or forums to protect its interest, FUA seems to assume that any and all incompletely resolved

matters regarding licensing of a nuclear plant should be resolved in adjudicatory hearings. This is not the practice of the Commission, nor is it the purpose of the hearing process. Many matters in the licensing process are handled routinely without hearing as part of the Staff's review prior to licensing or thereafter. The Appeal Board discussed this point in its decision denying FUA's earlier petition to intervene:

> "It does not follow from FUA's exclusion from the proceeding that its concerns perforce will be ignored in the licensing of this reactor.... To the extent that they go beyond the bounds of the hearings as fixed prior to the belated FUA intervention attempt, under the long-prevailing regulatory scheme these concerns fall within the province of the Staff." ALAB-642, 13 NRC at 895.
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> "As to those aspects of reactor operation not considered in an adjudicatory proceeding (if one is conducted), it is the Staff's duty to insure the existence of an adequate basis for each of the requisite Section 50.57 determinations." (Id. at 896).

This is precisely the situation we have here.

We do not claim that it is the function of the Applicants or the Staff in this proceeding to protect FUA's litigative interests. But those parties are protecting the public health and safety outside the hearing process. To the extent that FUA is interested in the public health and safety (as distinct from opposing the licensing of this plant), the NRC as a whole protects that interest outside the hearing process and FUA is free to make known its

views to the Staff or the Commissioners. As is evident from the affidavit of W.D. Fletcher, Westinghouse Corporation, the affidavit of Michael D. Quinton, SCE&G, and correspondence with the NRC Staff attached hereto, SCE&G has been diligently working with Westinghouse and the Staff to correct the problem with full appreciation of its significance. We believe that the Staff has been on top of the matter and, as evidenced by the documents attached hereto, is engaged in a program involving cooperative efforts of the industry and the utilities to guard against and remedy the tube wear problem. (See Attachment A, Steam Generator Status Report, February 1982, SECY-82-72, February 18, 1982). For these reasons, in the present circumstances, and especially given FUA's lack of demonstrated ability make a technical contribution to this matter, there is no reason why the Staff cannot protect FUA's interest outside the hearing process.

FUA argues that it cannot rely on Mr. Bursey to represent its interest in this matter. Nonetheless, FUA initially took the risk of relying on the existing intervenor. (FUA Petition to Intevene and Request for Hearings, March 23, 1981, at 3-4) On this factor, the Appeal Board in Puget Sound Power & Light Co. (Skagit Nuclear Power Project, Units 1 and 2), ALAB-559, 10 NRC 162, 172-73 (1979) wrote:

> "the promiscuous grant of intervention petitions inexcusably filed long after the prescribed deadline would pose a clear and unacceptable threat to the integrity of the entire adjudicatory process. [citations omitted]. More specifically, persons potentially affected by the licensing action under scrutiny would be encouraged simply to sit back and observe the course of the proceeding from the sidelines unless and untl they became persuaded that their interest was not being adequately represented by the existing parties..."

Ability to Contribute. As to the third factor, the extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record, FUA contends that only through its participation will the Board have a complete and sound record on this matter (FUA Petition, at 12). The question of a petitioner's ability to contribute to the development of a sound evidentiary record was most recently addressed by the Appeal Board in Allens Creek, supra March 31, 1982 slip op. at 10. The Appeal Board in this case weighed the ability to contribute factor heavily against FUA in denying its earlier petition to intervene. ALAB-642, 13 NRC at 891-93. As previously discussed, the Board, in ruling on the 17 contentions not admitted in FUA's first petition to intervene, found the ability to contribute factor "weighed most heavily against the petitioner." LBP-81-11, 13 NRC at 426. With regard to those contentions including steam generators) originally denied, the Board stated,

> "we can only contrast petitioner's familiarity with the substance of these issues [those originally admitted] with its lack of prior

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> involvement or expertise in the other areas it raised. On those other issues, it named few or no witnesses committed to testifying on its behalf, but sought mainly the opportunity to search for such witnesses. In view of the late date, we see no reason to afford that opportunity." Id.

In its most recent petition, FUA makes no showing as to any ability to contribute sound evidence; and gives no indication as to what expert witnesses or other evidence, if any, it would offer on this matter. 6/ Further, FUA does not specify what actions, if any, it believes should be taken in lieu of or in addition to those to which the Applicants are already committed. Based on FUA's petition, there is no reason to believe that it can contribute meaningfully and substantially to resolution of this matter.

When a petitioner has no ability to contribute to development of a sound record, the fourth factor (as to protection of its interest by existing parties) should not be weighed very heavily and the second and third factors, i.e., other means by which its interest may be protected and its ability to contribute, should be weighed quite heavily. See ALAB-642, 13 NRC at 895. We can well understand that the Board would want assurance that the Applicant and the Staff are aware of the need for corrective action, and that it is being implemented by the Applicant with appropriate

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6/ See Detroit Edison Co. (Greenwood Energy Center, Units 2 and 3), ALAB-476, 7 NRC 759, 764 (1978).

oversight by the regulatory staff. If that assurance is forthcoming, as is shown by the attached affidavits and documents, and we would expect by the Staff response, there is nothing to be gained by further hearings. If the matter will be resolved so as to protect the public health and safety without the petitioner's intervention, then the Board need not make any findings with regard to steam generators but should leave appropriate resolution of this matter to the Staff. This is consistent with the role of the Board in an operating license proceeding under 10 C.F.R. § 2.760a vis-a-vis that of the Staff.

Delay. The last of the five factors in 10 C.F.R. § 2.714(a)(1) is the extent to which the petitioner's participation will broaden the issues or delay the proceeding. FUA admits at page 12 of its petition that admission of these new contentions will expand the issues now before the Board, yet argues that because of their gravity, any delay is warranted. The Appeal Board, denying FUA's petition to intervene last year, stated that a late petitioner bears a heavy burden and the delay factor is extremely important. ALAB-642, 13 NRC at 888-89; see Allens Creek, supra, ALAB-671, March 31, 1982 slip op. at 6, 11. As discussed at pages 1-3 of this response, the consequences of delay and the impact on the rights of the parties is extremely crucial at this point in the proceeding. FUA urges that any delay is

justified by the gravity of the issue. We address the significance of the steam generator issue in the following section by reference to the standards for re-opening. 7/

The Legal Standard for Reopening

The Appeal Board reserved the question of whether the standards for reopening the record are applicable to a non-party in Allens Creek, supra, March 31, 1982 slip op. at 5 n.5. The reopening analysis is appropriate, however, because a non-party should not be held to a lesser standard than a party to a proceeding. This is a case in which assessment of the five factors for admitting a petition for late intervention would be aided by the "significant" issue analysis applied to reopening. FUA itself suggests the "significance" analysis is appropriate in its argument on the fifth factor for intervention when it says expanding the issues and delay are warranted by the gravity of the matters it seeks to raise.

Briefly, reopening the record is based on appraisal of three factors: (1) Is the motion timely? (2) Does it address

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7/ We briefly respond to the petitioners allegations regarding completion of the plant. FUA argues the Board "should not credit any estimates" of a fuel load readiness date (FUA Petition, at 12). Completion dates are estimates made in good faith of when the physical completion, checkout, and all of the numerous other items which go to make up readiness for fuel loading, will be completed assuming, generally, only a minor allowance for things going wrong. The most recent estimate of a fuel load readiness date is May 1982 (Briefing on Status and Assessment of Near Term Operating Licenses, April 20, 1982, Tr. 29-30).

significant safety (or environemntal) issues? (3) Might a different result have been reached or would the outcome have been affected had the newly proferred material been considered initially? 8/ Further, to justify the granting of a motion to reopen, the moving papers must be strong enough, in light of any opposing filings, to avoid summary disposition. Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-138, 6 AEC 520, 523 (1973). Thus, even though a matter might be timely raised and involve a significant safety issue, no reopening of the evidentiary hearing will be required if the affidavits submitted in response to the motion demonstrate there is no genuine unresolved issue of fact, i.e., if the undisputed facts establish that the allegedly significant safety issue does not exist, has been resolved, or for some other reason will have no effect upon the outcome of the proceeding. (Id.)

The affidavits and other attachments hereto provide the basis for summary disposition on the pleadings. As is evident from review of the Commission and Staff papers, the Applicants' affidavit, and the Westinghouse affidavit, resolution of the problem of accelerated tube wear in Model

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8/ For a complete discussion of the standards for reopening see Pacific Gas & Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-598, 11 NRC 876, 879 (1980) and Public Service Co. of Oklahoma (Black Fox Station, Units 1 and 2), ALAB-573, 10 NRC 775, 804 (1979) Kansas Gas & Electric Co., et al. (Wolf Creek Generating Station, Unit No. 1), ALAB-462, 7 NRC 320, 328 (1978).

D3 steam generators is being capably pursued. We discuss these various documents in chronological order.

On January 20, 1982, in a letter from Mr. Eisenhut to Mr. Nichols, the agency requested that SCE&G provide information concerning its reliance on Westinghouse test data, testing at operational plants, its plans for instrumentation to detect flow-induced vibrations, and the testing and start-up procedures proposed for Summer (See Attachment B).

Also on January 20, the Staff issued a Board Notification, BN-82-02, on Preheater Type Steam Generators which included information on the results of testing and inspection of steam generators at Duke Power Company plants as well as two foreign plants (See Attachment C). The McGuire testing program discussed was conducted primarily at power levels of 50% or less. No significant tube wear as indicated (Id.).

An NRC memorandum from Mr. Chesnut to Mr. Youngblood, dated February 19, summarized a meeting in Bethesda between the NRC, Westinghouse, and representatives of various utilities, including SCE&G, at which the problems with Model D steam generators were addressed (See Attachment D).

SCE&G, in a letter from Mr. Nichols to Mr. Denton dated February 19, 1982, outlined its plans to inspect for, monitor and correct problems associated with the Model D3 steam generators (See Attachment E). The letter describes

the Westinghouse testing program for Model D steam generators and proposes the following program for Summer: 1) normal low power testing, 2) complete startup testing with power escalation up to 50% power, 3) continue operation at 50% power for approximately two months, or at power levels above 50% based on information available to preclude tube damage, 4) shut down and eddy current test rows 49, 48, and 47 of one steam generator, and 5) reevaluate available data to confirm continued limited power operation until a modification can be made to resolve the matter (Id.). Following that phase of the test program, SCE&G and Westinghouse are committed to jointly establish an operating power level for Summer considering: 1) results of the eddy current inspection, 2) experience and data from other operating plants, 3) status of the Westinghouse program, and 4) status of any proposed Westinghouse modification to resolve the problem (Id.).

Further, on April 14, SCE&G informed the Commission, in a letter from Mr. Nichols to Mr. Denton, that internal diagnostic instrumentation will be installed in one of the steam generators at the Summer plant as part of its monitoring program outlined on February 19 (See Attachment F).

At the request of the Commission, the Staff is engaged in an overall review of steam generators as a generic matter. In a memorandum from Mr. Dircks to Mr. Minogue dated February 17, transmitting its Status Report on

on Steam Generators, the NRC outlined its current overall steam generator program (See Attachment A). The report defined the problem associated with tube degradation and its safety significance (which includes problems of tube wear arising from flow-induced vibration in preheater-type units); the NRC regulatory approach; current corrective actions; the NRC, industry, and foreign research and development activities; and, the long term approach for resolving the matter of steam generator tube integrity and the NRC/industry program (See Attachment A). The NRC Status Report and an agency's outline of ongoing work were transmitted to the Commissioners as SECY-82-72 on February 18, 1982. (See Attachment A).

In a memorandum dated March 25, 1982, SECY-82-72A, Mr. Dircks advised the Commissioners of the Staff's coordinated program to address steam generator problems. (Attachment G). The program will involve the efforts of the Atomic Industrial Forum, Electric Power Research Institute, Steam Generator Owners Group, the ACRS, and the Staff to coordinate and manage research, review needs, and develop short-term and long-term solutions. The program will pursue the areas of materials, water chemistry and control, design and technical considerations (which includes problems arising from excessive vibration), secondary system components, primary and secondary side inspection, repair procedures and personnel exposure,

systems interactions, quality assurance, and operating experience.

As a result of these combined efforts, both the Applicants and the Staff, with the cooperation of the industry, have been fully aware the further work which needs to be done with respect to resolving the problems with tube degradation in the Model D3 steam generators. The Applicants are committed to the Staff to correct the problem. (See Quinton Affidavit at 4).

The parties have satisfied their obligation to report new information which is relevant to the proceeding. See Georgia Power Co. (Alvin W. Vogtle Nuclear Plant, Units 1 and 2), ALAB- 291, 2 NRC 404, 411 (1975). In this instance, as in numerous others, a board notification was issued informing the Board of a potentially significant issue. That alone, of course, does not determine whether a late petition should be granted or the hearing reopened. 8/ As we stated in an earlier response to Mr. Bursey's motion for admission of a new contention, while the Board might reasonably want assurance that the Staff is "on top of" the matter, it need not, in an operating license proceeding, take up every

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8/ On April 14, 1982, Mr. Bursey filed a motion for admission of new contention and request for hearings on this and other issues to which we will be responding separately.

matter that crops up during ongoing Staff review, but may leave matters outside the hearing process to resolution by the Staff. (10 C.F.R. § 2.760a; ALAB-642, 13 NRC at 895-96; see Grossman, Tr. 6136) (See Applicants' Response in Opposition to Intervenors Motion for Admission of New Contention, March 11, 1982, at 6). The Applicants are confident that this matter can and should be so resolved.

It should be understood that the tube wear issue which FUA uses as a basis for its belated petition is only one part of the overall steam generator "picture" discussed in these NRC documents. Mr. Fletcher, of Westinghouse, states in his affidavit that significant tube wear will be precluded during interim operation and alleviated by a permanent modification (Fletcher Affidavit, at 2). FUA raises issues concerning tangent point cracking of Row 1 tubes, tube rupture events, PORV malfunction, and accident sequences. These issues are not concerns during the V.C. Summer interim program. (Fletcher Affidavit, at 5). Operation during this period is designed to minimize tube wear so that tube rupture, multiple tube rupture, actions of the PORV, and LOCA events are not relevant (Id.) All the evidence currently available shows that it is amply conservative to operate plant employing Model D preheater-type steam generators at reduced power (Fletcher Affidavit, at 2) (Quinton Affidavit, at 4). Thus, the various consequences of steam generators tube failure postulated by FUA as a

result of tube wear never arise because significant tube wear will be precluded.

For these reasons, as detailed in the attached affidavits, the "significant" safety concern alluded to in FUA's petition does not arise. (Fletcher Affidavit, at 2) (Quinton Affidavit, at 6) Therefore, FUA's petition does not meet the standards of Vermont Yankee, supra 6 AEC at 523, for avoiding summary disposition on these pleadings.

Cost-Benefit Analysis. FUA in its new contention B2 questions the favorable cost-benefit analysis reached at the construction permit phase. FUA assumes that the V.C. Summer plant will never operate at greater than 50% power and therefore "a more realistic 35% capacity factor" should be used in a new cost-benefit analysis (FUA Petition, at 6).

The Commission, on March 26, 1982, issued its final rule on the need for power and alternative energy issues in operating license proceedings. 47 Fed. Reg. 12,940 (March 26, 1982). The rule effectively eliminates consideration of cost-benefit analyses based on power production at the operating license stage. The Commission has not diminished the importance of these issues at the construction permit stage, but has recognized that at the operating license stage the plant would be needed either to meet increased energy needs or replace older, less economical, generating

capacity and that no viable alternatives to the completed nuclear plant are likely to tip the NEPA cost-benefit balance against issuance of the operating license. Id. Experience shows that completed nuclear power plants are used to their maximum availability and there has never been a finding in an NRC operating license proceeding that a viable environmentally superior alternative to operation of the nuclear facility exists. Id. at 12,942. The purpose of the amendment is to avoid unnecessary consideration of issues at the operating license stage that are not likely to tilt the cost-benefit balance. (Id.) Hence, 10 C.F.R. § 51.53(c) is amended to provide "Presiding officers shall not admit contentions proferred by any party concerning need for power or alternative energy sources for the proposed plant in operating license hearings." Id. at 12, 943. We necessarily defer to the Staff for further development of the impact of the amended rule.

On the merits of FUA's proposed contention, we believe there is no change in the favorable cost-benefit analysis reached in the construction permit proceeding. There is no basis to assume that the plant will operate at reduced power for its entire lifetime. Although we would not argue that operation at reduced power for as much as a is beyond possibility or that a period such as a year could be assumed, for purposes of discussion of the contention, it

seems obvious that even one year at fifty percent power would not result in more than a minimal adjustment in total output over the lifetime of the facility.

Sua Sponte Authority. As to the alternative relief sought in the event intervention is denied, FUA would have the Board hear these matters under its sua sponte authority pursuant to 10 C.F.R. § 2.760(a). As we have stressed earlier in this response, there is no need for the Board to consider these matters for the Commission and the NRC Staff are moving to resolve the problem. There is nothing for the Board to add or require which is not already being done.

The issue of the Licensing Board's sua sponte authority was thoroughly addressed by the Commission in Texas Utilities Generating Co., et al. (Comanche Peak Steam Electric Station, Units 1 and 2), CLI-81-36, slip op. (Dec. 30, 1981). The standard for exercise of sua sponte authority is analogous to that for reopening, i.e., that a significant safety, environmental, or common defense and security matter remains. 10 C.F.R. § 2.760a; see Consolidated Edison Co. of New York (Indian Point Station, Unit 3), CLI-74-28, 8 AEC 7, 9 (1974). In Comanche Peak, the Commission stated that "the apparent need to ... monitor the Staff's progress in identifying and/or evaluating potential safety

or environmental issues are not factors which authorize a board to exercise its sua sponte authority." Comanche Peak, supra, December 30, 1981 slip op. at 3. Based on the authorities cited, we do not believe this to be a case in which the exercise of the Board's sua sponte authority is warranted or justified.

Conclusion

For all the foregoing reasons, the Petitioner has failed to satisfy the five requirements for late intervention and has similarly failed to meet the burden required for reopening the record. Accordingly, the Board should deny the Petitioner's motion in all respects.

Respectfully submitted,

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Gas Company
P.O. Box 764
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

SOUTH CAROLINA ELECTRIC & GAS
COMPANY Docket No. 50-395

(Virgil C. Summer Nuclear
Station, Unit 1)

COUNTY OF ALLEGHENY
STATE OF PENNSYLVANIA ss

AFFIDAVIT OF W. D. FLETCHER

My name is W. D. Fletcher. I am Manager, Steam Generator Development and Performance Engineering in the Nuclear Technology Division of the Westinghouse Electric Corporation. My business address is P.O. Box 855, Pittsburgh, Pennsylvania 15230. A statement of my educational, professional qualifications and experience is attached and forms a part of my affidavit.

I have reviewed the "Petition to Intervene and Request for Hearing" filed by Fairfield United Action ("FUA") dated April 9, 1982 in this proceeding. The purpose of this affidavit is to address the safety issues raised in the contention identified as "B1" by FUA. I have reviewed the results of the data gathered from operating plants,

utilizing Model D steam generators. In my opinion, the interim operation program developed by Westinghouse and South Carolina Electric & Gas Company ("SCE&G") for the V. C. Summer plant will preclude any significant steam generator tube wear until modification to the steam generators can be implemented. Contention B1 does not present, in my opinion, significant safety issues.

I am familiar with data from the operating plants utilizing Model D3 steam generators. Wear has been observed at plants utilizing Model D3 steam generators which have operated for extended periods of time at high main feed flow. Significant tube motion has been determined from instrumented steam generators and model tests to begin at about 60-65% main feed flow. It is believed that tube wear is related to significant tube motion. Significant wear is not expected below about 60-65% power level. As explained in greater detail below, I have concluded that the interim operating program for the Summer plant within the specified parameters will minimize the potential for significant tube wear.

An interim operating program for the Summer Plant has been developed by Westinghouse and South Carolina Electric and Gas Company with the objective of precluding any significant steam generator tube wear until the permanent modification to the steam generators can be implemented. This interim program includes:

- Instrumentation installed inside steam generator
- Low power testing
- Power escalation testing to main feed flow up to 50% power
- Power operation with main feed flow at 50% power for specified time intervals (including operation above 50% power based on information available at that time)
- Tubing eddy current inspections

For present purposes, power level and main feed flow can be taken to be equal. The 50% main feed flow power conditions identified for the interim period have been based on instrumentation data obtained from three operating plants (having Model D steam generators of similar design as installed at Summer) and hydraulic, vibration test model results, analysis of operating plant eddy current inspection data and tube inspections made on operating plants before and after 50 percent power operation.

During the interim operating period, it is understood that data from installed instrumentation will be examined and inspection of the steam generator tubing will be performed at appropriate intervals as a means of verifying that significant tubing degradation has not occurred.

In D3 operating plants where tube wear was observed, instrumentation was installed to measure tube motion, which has provided data relating motion to power level. This data supports the power level which can be used during the interim operating period. Characterization of tube motion in the Model D-3 steam generators at operating conditions showed that significant motion begins to occur at about 60% to 65% feed flow through the main nozzle. It is concluded from these data that operation at 50% feed flow through the main nozzle is an acceptable power level for the interim period to minimize the potential for significant tube wear.

A pertinent comparison of operating data which follows shows the operating history of three plants with similar steam generators, in terms of hours and power levels. The data are based upon the time of operation prior to tube inspection in which wear indications were measured.

COMPARATIVE POWER HISTORIES

% Power	Plant 1 (Non-domestic) as of Oct. 1981	Plant 2 (Non-domestic) as of Nov. 1981	Plant 3 (Domestic) as of Feb. 1982
	Number of Hours of Operation		
>90 to 100	1640	1456	72
>75 to 90	540	95	252
>50 to 75	1580	537	1176[1]
>20 to 50	1390	738	(negligible)
	Wear Observed	Wear Observed	No significant eddy current indications

[1]Most of this time period was at 50% power

At Plant 1 and Plant 2, where wear indications were observed in the tubes, significant periods of operation were at high power levels, i.e., greater than 90%. Plant 3 has operated for about 55% of the time since initial startup in December, 1981. At Plant 3, where the power level has been primarily at 50%, no significant wear was indicated. The judgment derived from analysis of this data is that no significant wear is to be expected from interim time periods of operation at the lower power levels.

Additionally, the results of other plant operating experience at 50% power, equivalent to the program for the Summer D-3 steam generators is relevant. At Plant 2 (D-3 steam generators) a program of 50% power operation for 1500 hours (with short intervals of higher power operation to obtain tube instrumentation data) was completed in March of 1982. Comparing the tubing inspection data in the preheater region prior to (i.e., in Nov. 1981) and after (i.e., in March 1982) this operating interval showed no significant change, including tubes which previously exhibited indications of wear.

Experimental testing has provided a data base which also supports the results observed from the tube instrumentation and plant inspection data. The tube instrumentation data indicate that the onset of tube motion is a result of turbulent feed flow at the preheater inlet at the higher equivalent power levels. The testing results have been obtained from several hydraulic models which demonstrate levels of turbulence and associated flow fluctuation at various feed flow rates. Data from a scale model ($\sim$4/10 scale) of the D-3 preheater demonstrates tube motion which increases with the feed flow rate in a manner similar to the data from the operating plant instrumentation. These test results support the conclusions derived from the field data.

Westinghouse currently has an engineering design program to modify Model D steam generators to alleviate tube wear. The anticipated design modification can be implemented in the field. An extensive testing program is in progress to verify that the anticipated modification to the steam generator preheater section addresses and corrects tube wear. The testing program incorporates a variety of testing facilities such as 0.417, 2/3 and full scale model testing. These test results, when combined with actual plant performance data, are expected to provide verification that the design modification will perform as predicted by engineering design analysis.

The other issues raised in FUA Contention B1 are not concerns during the V. C. Summer interim program as described above. Operation during this period is designed to minimize tube wear so that tube rupture, multiple tube rupture, actions of the PORV and LOCA events are not relevant.

Notwithstanding this, a brief status of these issues is as follows:

Row 1 U-bends

The staff analysis of tangent point cracking of Row 1 tubes concludes that while there remains a potential that this issue may occur in Model D Row 1, it is not a safety concern. Westinghouse supports the technical basis for this conclusion as detailed in Section 5.4.2 of the V. C. Summer SER, Supplement 3, January, 1982.

Tube Rupture Events

Westinghouse has performed analyses for the postulated double-ended steam generator tube rupture event for all Westinghouse-designed NSSS plants. Systems installed in these plants are designed to accommodate these events. The results of these analyses and of other analyses have been used to formulate Emergency Response Guidelines (ERG) for use by utilities in writing plant-specific Emergency Operating Procedures. Westinghouse has also performed evaluations of those significant tube leakage events that have occurred and has used the results of these evaluations to further improve the guidance provided in earlier versions of the Westinghouse ERG's.

Additional evaluations of these significant tube leakage events and the manner in which those leaks developed, leads Westinghouse to believe that multiple tube rupture is very unlikely. This belief is further reinforced by the results of steam generator tube inspections that are routinely performed in operating steam generators and also as described above for the Summer interim operation program. These inspections are specifically aimed at early detection of any condition that would have the potential for tube rupture. In addition, tube conditions are continuously monitored for tube leakage during normal operation.

As noted above, Westinghouse believes that concurrent multiple tube ruptures or concurrent tube ruptures in multiple steam

generators are very unlikely. Notwithstanding this belief, Westinghouse has performed, for the purposes of developing additional ERG's, evaluations of multiple tube ruptures and of tube ruptures in multiple steam generators. These ERG's take into consideration past operating experience, post-TMI lessons learned and multiple equipment failure contingencies. These procedures have been written to minimize radioactive release from the plant. This effort was sponsored by the Westinghouse Owners Group (WOG) of which South Carolina Electric and Gas is a member.

The latest revision of these guidelines were submitted by the WOG to the NRC in November 1981 and are currently under review. In a meeting with the ACRS on 3/24/82, NRC staff expressed basic agreement with the Westinghouse guidelines.

PORV

With respect to a tube rupture, the PORV provides a means to depressurize the primary side of the steam generators. If the PORV were to stay in the open position during use, the operator could isolate the PORV with the PORV isolation (block) valves, such as used at both Ginna and Three Mile Island, to effectively isolate the PORVs. The presence of three PORVs and associated block valves in the Summer plant allows the operator sufficient flexibility to enable him to decrease and control RCS pressure. The NRC has requested certain operational data for pressurizer PORV's through NUREG-0737, Item II.K.3.2. In responding to this request, Westinghouse conducted a survey of domestic Westinghouse NSSS operating plants, and the results of such survey were provided to the NRC. Notwithstanding the experience in the past that some PORVs have in a few instances remained in the open position and did not close on demand, the results of the Westinghouse survey and operating experience subsequent to the survey, still indicates that the PORV reliability is acceptable.

Moreover, the industry has implemented certain testing. For example, the Electric Power Research Institute (EPRI) has data from such tests that are directly applicable to the PORVs at Virgil Summer, since EPRI tested the same exact model PORV as installed in the Virgil Summer nuclear facility [Copes-Vulcan PORV Model D-100-160 (316 w/Stellite plug and 17-4Ph cage)]. As reported in the "EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report," (April 7, 1982), twenty tests were conducted on this valve during which the valve fully opened and fully closed on demand.

Accident Sequences

Postulated accident sequences, such as the loss-of-coolant accident (LOCA) and the feedline break (FLB) accident, impose increased loads on the steam generator tubes. Such postulated increased loads have been evaluated by Westinghouse in an analytical program. The results of these analyses show that tubes which exhibit degradation less than that specified in the tube plugging criteria will maintain their integrity for all postulated design basis accident sequences.

These studies have further shown that for a LOCA, the maximum steam generator tube stresses, caused by rarefaction waves, blowdown and vibrational forces, occur in the U-bend region of the steam generator. The tube stresses near the tubesheet and in the preheater region are lower. In addition, the pressure force mechanism during the postulated LOCA is in the direction of the potential for tube collapse rather than tube rupture since the primary side has depressurized. A postulated tube collapse has small potential for creating a leak path between the secondary side and primary side of the steam generator.

Westinghouse studies have shown that the postulated feedline break accident produces the highest steam generator tube stresses of any of the design basis accident sequences. During the postulated feedline break accident, the primary steam generator tube stresses result from the pressure differential across the tube. The calculated stresses resulting from the postulated feedline break accident are within acceptance limits for a tube with wear even more than that limit specified in the tube plugging criteria.

Postulating a steam generator tube rupture in conjunction with another design basis accident sequence constitutes a double failure. As such, the ECCS Acceptance Criteria in 10CFR50.46 and the requirements of 10CFR50 Appendix K do not require evaluation of such an event. Accordingly the evaluation models do not apply to such postulated double failures and evaluation of such a postulated double failure has been performed using modified codes and assumptions.

Westinghouse has performed calculations of a postulated loss-of-coolant accident ("LOCA"), with and without secondary to primary steam generator tube leakage, for a typical Westinghouse plant. The calculation was performed for a double-ended cold leg break of the reactor coolant pipe using codes and assumptions modified to more realistically represent the system than is performed for design basis calculations. Contrary to FUA contention (page 4), these calculations showed that adequate core cooling was maintained. The results of these calculations were that less than a 10°F increase in peak fuel cladding temperature was calculated when a 250 gpm secondary to primary tube leak was modelled. This leak rate is consistent with a postulated double-ended break of a steam generator tube. Since the pressure force mechanism during

a LOCA is postulated to occur in the direction of the potential for tube collapse and since wear patterns on tubes removed from two operating plants exist on primarily one side of the tube, a postulated double-ended tube break resulting from a LOCA is considered highly unlikely. Moreover, the data from the two operating plants indicates that one tube tended to lead the rest by at least 10 percent of the tube wall thickness reduction, thus the participation by leakage of more than one tube is considered unlikely.

Further, affiant sayeth not.

AFFIDAVIT

Before me, the undersigned authority, personally appeared W. D. Fletcher, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this affidavit are true and correct to the best of his knowledge, information, and belief:

W D Fletcher

W. D. Fletcher, Manager
Steam Generator Development
and Performance Engineering

Sworn to and subscribed
before me this 22 day
of April 1982.

Paulette Slonska

Notary Public
PAULETTE SLONSKA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES MARCH 10, 1986
Member, Pennsylvania Association of Notaries

STATEMENT OF QUALIFICATIONS AND EXPERIENCE

W. D. FLETCHER

EXPERIENCE

My name is W.D. Fletcher; I am presently Manager, Steam Generator Development and Performance Engineering in the Nuclear Technology Division of the Westinghouse Electric Corporation.

I graduated from Hardin-Simmons University in 1950 with a Bachelor's degree in Chemistry and from Fordham University in 1960 with a Masters degree in Chemistry.

I was employed with the Vitro Laboratories from 1951 to 1955, where I performed research on organo-phosphorus compound synthesis, reaction kinetics and mechanisms of organo-phosphorus compounds, phase studies, bench scale and pilot plant production of organo-phosphites, high and low temperature kinetic studies of boron hydride synthesis, and electro-kinetic studies of electrophoretic deposition of inorganic oxides in the manufacture of reactor fuel elements.

In 1957 I began my employment with Westinghouse and have been engaged in development work on the heterogeneous catalysis of reactions between hydrogen and oxygen produced through radiolysis of reactor coolants, reaction kinetics and mechanisms, catalyst development and evaluation in high temperature and pressure aqueous solutions; evaluation and study of reactor coolant contaminants

and means of coolant purification; study of behavior of fission and corrosion products in reactor coolants; in-pile studies of reactor coolants as pertains to chemical shim technology; reactor plant chemistry control, analyses, and data collection and interpretation of all operating reactor systems designed by Westinghouse.

Since 1970, I have been directly involved in development and design activities related to Westinghouse steam generators. Under my direction, steam generator programs related to operations have been executed involving chemistry and materials as well as specific design configurations.

As Manager, Steam Generator Development and Performance Engineering, I am responsible for three design-development groups that involve steam generator thermal/hydraulics, advanced concepts design and analysis and design of field modification to steam generators.

I am a member of the American Chemical Society, the National Association of Corrosion Engineers, the American Nuclear Society, and the American Society of Mechanical Engineers.

PUBLICATIONS

"Update of Operations with Westinghouse Steam Generators" American Nuclear Society 1977, D.C. Malinowski and W.D. Fletcher.

"Operating Experience with Westinghouse Steam Generators", Nuclear Technology, 1975 W.D. Fletcher and D.D. Malinowski.

"Water Technology for Nuclear Power/PWR's", Industrial Water Engineering, 1971, W.D. Fletcher.

"Primary Coolant Chemistry of PWR's", W.D. Fletcher, the International Water Conference of the Engineers Society of Western Pennsylvania, Pittsburgh, October, 1970.

"Post Accident Iodine Cleanup by Containment Filters and Sprays." Presentation at Tampa, Florida, May 21, 1968, J.D. McAdoo and W.D. Fletcher.

"Effects of Coolant Chemistry on Corrosion and Corrosion Products", W.D. Fletcher, Am. Nuc. Soc., Seattle, June 1969.

EURAEC-1972 (WCAP-3690-4) - "Description and Evaluation of the Boron Concentration Meter Utilized at the SENA (Franco-Belge) Reactor Plant", January 1968, W.D. Fletcher.

WCAP-3269-57 - "The Post-Irradiation Examination of Saxton Fuel Cladding Corrosion Products", March 1966, L.F. Picone and W.D. Fletcher.

WCAP-3269-63 - "Fission Products from Fuel Defect Test at Saxton", April 1966, W.D. Fletcher and L.F. Picone.

WCAP-2964 - "Stability of Alkali in Reactor Coolant", 1964, W.D. Fletcher.

WCAP-2656 - "Analysis of Fission Products in Saxton Primary Coolant", August 1964, W.D. Fletcher.

"Water Technology of the Saxton Nuclear Experiment", Division of Water and Waste Chemistry, 4, 46 (1964), W.D. Fletcher and R.F. Swift.

"Flame Photometric Determination of Lithium Produced by B-10 (n,a) Li-7 to Measure Boron-10 Burnup in Reactors Utilizing Chemical Shim Control,: Presentation at Gatlinburg, Tenn., Oct. 6-8 1964, B.D. LaMont and W.D. Fletcher.

WCAP-3716 - "Ion Exchange in Boric Acid Solutions with Radioactive Decay", November 1962, W.D. Fletcher.

WCAP-1689 Rev. - "The Behavior of Stainless Steel Corrosion Products in High Temperature Boric Acid Solutions", May 1961, W.D. Fletcher, A. Krieg and P. Cohen.

WCAP-4097 - "Inorganic Ion-Exchanger Materials for Water Purification in CVTR", August 1961 (CVNA-135), N. Michael, W.D. Fletcher, et al..

WCAP-3730 - "Interactions Between Stainless Steel Corrosion Products and Boric Acid Solutions", March 1960, W.D. Fletcher.

"Some Performance Characteristics of Zirconium Phosphate and Zirconium Oxide Ion Exchange Materials", Trans. Am. Nuc. Soc., 3, 46 (1960), N. Michael and W.D. Fletcher.

WCAP-1206 - "Internal Recombination Catalyst Studies", May 4, 1959, W. D. Fletcher and D.E. Byrnes.

WCAP-1110 - "A Semi-Flow System for the Study of Catalytic Combination of Hydrogen and Oxygen in Aqueous or Slurry System", February 1959, W.D. Fletcher and W.E. Foster.

"Electrophoretic Deposition of Metallic and Composite Coatings", Plating 42, 1255 (1955).

"Post LOCA Hydrogen Generation in PWR Containments", W.D. Fletcher, M.J. Bell, R.T. Marchese, and J.L. Gallagher, American Nuclear Society.

PATENTS

U.S. Patent, "Information Storage Systems and Methods for Producing Same".

U.S. Patent, "Boron Concentration Meter".

U.S. Patent, "Electrophoretic Coating Dispersion Formulations".

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of SOUTH CAROLINA ELECTRIC & GAS COMPANY, et al. (Virgil C. Summer Nuclear Station)	))))))))	Docket No. 50-395 OL

AFFIDAVIT OF MICHAEL D. QUINTON
REGARDING THE V. C. SUMMER NUCLEAR STATION
STEAM GENERATORS

Michael D. Quinton, being first duly sworn according to law comes forward and states:

My name is Michael D. Quinton. I am employed by South Carolina Electric & Gas Company as Director of Mechanical Engineering within the Nuclear Engineering Department. In this position I have been assigned specific responsibility to monitor technical developments relative to steam generator tube wear problems associated with Westinghouse Model D steam generators and particularly Model D-3, which is the model of steam generators installed in the Summer Station three loop unit. I possess a degree in mechanical engineering, received from the University of South Carolina in 1973. I am a registered professional engineer in the State of South Carolina and I have been involved in the operation, design and construction of nuclear power plants for over 12 years. (Before entering college I was a qualified reactor operator/reactor technician in the U. S. Navy.)

I have read the "Petition to Intervene and Request for Hearings" dated April 9, 1982 and filed by Fairfield United Action in this docket. The purpose of my Affidavit is to assess the significance of safety questions which FUA seeks to raise. I have concluded the issues raised by them in their petition do not present a significant risk to the health and safety of the public. I will describe the Applicants' involvement with this problem and with utility and vendor efforts to identify the cause and develop an effective remedy. I shall also describe Applicants' commitments relative to interim operation of the Virgil C. Summer Nuclear Station and evaluation of proposed modifications leading to installation of an effective and approved "fix."

Through industry publications, information from Westinghouse, and information exchange systems such as NUCLEAR NOTEPAD, we became aware of Westinghouse designed steam generator operating problems at Ringhals (Sweden) and Almaraz (Spain) (which both utilize Model D3 steam generators). However, it was not until late December, 1981 that we were advised that these problems had direct application to the Summer Plant (also having Model D3 steam generators).

Upon being so informed, we initiated an information gathering and evaluation effort involving frequent contact with Westinghouse as well as other utilities similarly affected to discuss the nature of the mechanisms observed, interim protective measures, and the possible remedies.

By letter dated January 20, 1982 the Director of the Division of Licensing of NRC's Office of Nuclear Reactor Regulation (NRR) requested that we provide them with our

plans to address the problem as it relates to the Summer Plant. By a letter of the same date, the NRC Staff gave notification to the ASLB of the problem.

By letter dated February 19, 1982, Applicants responded to the request for our position on the matter. This letter was sent to all parties in this proceeding. In that letter, we committed to an interim operating program for the Summer Plant with the objective of precluding any significant steam generator tube wear pending permanent modification. Our response outlined our plans to proceed with initial core loading, low power testing, and escalation up to and continued operation at the 50% level (or a higher level if justified by the information available at the time). Additionally, we committed to shut down after approximately two months of operation at the escalated level and inspect (eddy current testing) tube rows 47, 48, and 49 for indications of tube wear. At the time of that response, there were no plans for providing internal or external instrumentation as installed at some of the other plants with similar problems. However, since that time we have committed to install internal instrumentation in two tubes in one steam generator. This commitment is contained in a Nichols to Denton letter dated April 14, 1982, which has been sent to all parties. Westinghouse has indicated that a modification to correct the tube wear problem may be available by late summer, 1982. It is estimated that implementation of the modification in

all three (3) steam generators will take approximately three to four months.

It is my opinion that the operation of the V.C. Summer Nuclear Station Unit No. 1 under the conditions set forth in the Nichols to Denton letter of February 19, 1982 as supplemented by the instrumentation commitment in the Nichols to Denton letter of April 14, 1982 presents no significant risk to the health and safety of the public. This is based upon the results to date of the Westinghouse analysis and test programs at the two operating foreign reactors and Duke Power Company's McGuire unit as well as other Westinghouse studies. (See Fletcher Affidavit.) In addition to health and safety considerations, which are of primary concern, the economic incentive for Applicants to avoid operations at power levels posing any significant risk of steam generator damage with the cost penalties attendant to such damage provides every reason for us to adhere to the monitoring and testing programs outlined in our February 19, 1982 letter.

To address FUA's proposed Contentions (B1 and B2) specifically, I note first that an underlying premise in those contentions is that the Summer unit will be operated at a power level at which flow-induced vibrations in the preheater region will act to cause tube wear. This premise is incorrect. The Applicants' commitment is to limit operation of the unit to 50% power (or to an appropriate level of power above 50%) which precludes significant tube damage.

The mechanism for the inducement of tube wear in the Model D steam generator cited by FUA is in agreement with our current information on the subject. FUA has referenced the Chesnut to Youngblood Memorandum statement that the increased turbulence is experienced at feed flow rates of approximately 50% in Model D3 steam generators. Our commitment to limit operation to 50% at this time is consistent with the NRC memorandum. Based on available information, operation at this level precludes the tube wear problem.

Since the accelerated tube wear problem will not arise during interim operation (or after a permanent modification is made to preclude the problem), there is no basis to postulate, as FUA does, tube rupture or multiple tube rupture, possibly in combination with PORV failure or possibly leading to LOCA events, as a consequence of accelerated tube wear. Nevertheless, a few comments are in order on those matters to correct the impression that might otherwise be left by FUA's statements.

Westinghouse has performed analyses for postulated double end steam generator tube rupture events for all Westinghouse designed nuclear steam supply system plants. (See Fletcher Affidavit, page 6.) While FUA properly points out that the FSAR (5.2-16) gives the design basis tube failure as a double ended rupture of single tube, it is also true that this accident will result in a transient which is no more severe than that associated with a reactor trip from full power and thus requires no special treatment insofar as fatigue evaluation is concerned. (Id.)

FUA's discussion of potential problems with the Power

Operated Relief Valve (PORV) on the pressurizer is misplaced. The PORVs at the Summer Plant are of the specific model tested by the Electric Power Research Institute (EPRI). The results of those tests conclusively demonstrate the PORV's operability ("EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, April 7, 1982). The "anomalies" in safety valve (not PORV) performance referred to in the Nichols to Denton letter are in no way related to PORV operability.

Applicants are well aware of the potential for premature tube wear problems in Westinghouse Model D steam generators. We recognize the need for corrective action and are working with the vendor and the NRC to develop, verify and install a suitable modification to eliminate the problem. Based upon our understanding of the mechanisms involved, we have developed and committed to an interim operating program to insure significant steam generator tube wear does not occur. By virtue of our actions in this matter, operation of the V.C. Summer Nuclear Station does not present undue risk to the public health and safety.

Michael D. Quinton

Michael D. Quinton

SWORN to before this

23rd day of April, 1982

Darlene G Stowe (L.S.)

Notary Public for South Carolina

My Commission expires: 12-22-88

ATTACHMENT A

February 18, 1982

SECY-82-72

POLICY ISSUE
(Information)

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: OVERALL STEAM GENERATOR PROGRAM

Purpose: To transmit the enclosed report, "Steam Generator Status Report."

Discussion: As a result of concerns expressed by the Chairman regarding the overall program addressing steam generator problems, I have requested the staff to review NRC and industry programs related to this subject and to develop a comprehensive, integrated program for addressing the issue. Furthermore, I have emphasized the need for and asked the staff to consider joint NRC/industry programs that will promote resolution of the problem. I have assigned the lead responsibility to RES to work with NRR to pull together a plan of action. (See February 17, 1982 memorandum to Robert B. Minogue from William J. Dircks, Subject: Overall Steam Generator Program.)

As a starting point, RES, with input from NRR and I&E, has prepared the enclosed report, "Steam Generator Status Report." The report summarizes steam generator degradation experience and its safety and economic significance; discusses the current regulatory approach for ensuring safe steam generator operation; describes current corrective actions and their limitations; summarizes NRC, industry, and foreign research and development activities; and discusses the long-term approach for addressing steam generator problems. The report is intended to provide a "snapshot" of the steam generator issue as it exists today and to serve as a background and foundation for development of the proposed integrated program. The staff is currently working to develop such a program and will discuss its feasibility and form in a later report.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Dircks' Memo
2. Status Report

Contact:
J. Strosnider, RES
443-5903

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 17, 1982

MEMORANDUM FOR: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: OVERALL STEAM GENERATOR PROGRAM

As you know, as a result of concerns raised by the Chairman associated with the overall program to address the steam generator problems, I discussed this issue with the various Office Directors involved. As a result, I am assigning you the lead responsibility working jointly with NRR to pull together a plan of action to address the steam generator concerns. You should consult with and get input from other offices, particularly IE, as necessary.

I envision the development of two papers for the Commission's information and consideration:

1. The first paper, to be developed in a short time frame, should address principally the question of "What is going on now?", and "What has happened to date?" The general framework of this paper would include:

   a. A discussion of the problems (both regulatory and economic) and their safety significance;

   b. What programs are now going on within the NRC, DOE, USN, industry (including EPRI and AIF), and international programs. The NRC efforts include the NRR efforts on Unresolved Safety Issues A-3, 4, 5, as well as ongoing RES efforts;

   c. The short term fixes that have been proposed and/or implemented for individual problems over the years and some insight into difficulties;

   d. Staff recommendations on the short term actions (for all affected reactors) believed to be prudent to ensure that public health and safety are protected; and

   e. A general idea of possible direction to attack the problem in the long term.

2. Recognizing that current efforts (NRC and regulated industry) are probably not nearly enough, a second paper, again working jointly with NRR, should be developed. This would include what additionally

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STEAM GENERATOR STATUS REPORT

FEBRUARY 1982

U.S. NUCLEAR REGULATORY COMMISSION

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must be done, and a plan of action for a joint research effort to provide a definitive resolution. To assure a creditable product, close coordination with industry will be required. Perhaps a lead group representing the industry, such as EPRI, can, together with NRC staff, ACRS, and other consultants as necessary, form a steering group to identify what must be done, who should do what to assure complete coverage without overlap and the priorities of work - to include the inspection program.

I would like you to report to me at the weekly meetings regarding progress, problems, or action I can take to move this program along in an expeditious manner.

William J. Dircks
Executive Director
for Operations

CC: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
H.R. Denton/NRR
R.C. DeYoung/IE

I. Problem Defintion

A. Summary of Tube Degradation

Degradation of steam generators (SG) manufactured by each of the three pressurized water reactor (PWR) vendors has resulted due to a combination of steam generator mechanical design, thermal hydraulics, materials selection, fabrication techniques, and secondary system design and operation. In the early and mid-1970s, Westinghouse (W) S.G. experienced caustic stress corrosion cracking, and W and Combustion Engineering (CE) S.G.s experienced tube thinning (wastage). These modes of degradation were due to difficulties encountered with phosphate secondary water chemistry. Because of these difficulties, most W and all CE plants converted to an all volatile (AVT) secondary water treatment. Although this conversion greatly reduced the occurence of stress corrosion cracking and wastage, other degradation modes including denting (deformation of the S.G. tubes due to corrosion of the carbon steel support plates) began to occur.

Babcock and Wilcox S.G., which have a significantly different design from W or CE and have operated exclusively with AVT water chemistry, had relatively good operating experience in their early years of operation. Nevertheless, they have experienced numerous tube leaks. The principal modes of degradation in B&W units have been fatigue crack growth, confined primarily to limited sets of tubes located on the open inspection lane, and more recently erosion-corrosion and primary side intergranular attack.

To date, many different forms of steam generator degradation have been identified including: stress corrosion cracking, wastage, intergranular attack, denting, erosion-corrosion, fatigue cracking, pitting, fretting, and support plate degradation. One or more of these forms of degradation have affected at least 40 operating PWRs and have resulted in extensive S.G. inspections, tube plugging, repair, or replacement. Recently, foreign Westinghouse S.G.s of the same design as McGuire have experienced tube wear associated with flow induced vibration due to a new integral preheater design. References 1, 2, and 3 present detailed discussions of domestic S.G. operating experience.

The economic impact of steam generator degradation has been significant. Approximately 23% of non-refueling outage time has been attributed to steam generator degradation. The cost of such outages in terms of replacement power alone is very high. However, perhaps the greatest financial costs incurred to date are those associated with steam generator replacement. Replacement of the Surry Unit 1 and Unit 2 S.G.s cost approximately $200 million, including cost of makeup power. Replacement of the Turkey Point S.G.s, currently in progress, will cost an estimated $460 million. NRC staff time involved with these activities is estimated at 6000 manhours for Turkey Point (which included time for a hearing) and 3000 manhours for Surry. Less radical operations also incur significant costs. Recent tube sleeving operations at San Onofre involved repair of approximately 7000 degraded tubes at a cost of $70 million. Proposed sleeving of 3000 tubes at R.E. Ginna has an estimated cost of $20 million.

B. Safety Significance

The safety significance of S.G. tube integrity can be divided

into three categories: tube failures under normal operating conditions; tube failures concurrent with postulated accident conditions; and personnel exposure associated with S.G. inservice inspection (ISI), repair, and replacement.

The majority of the S.G. tube failures that have occurred under normal operating conditions were small stable leaks sometimes requiring plant shutdown, inspection, and corrective actions, but for the most part small enough (e.g., below technical specification leak rate limit) that operations continued until a scheduled shutdown. However, four significant S.G. tube ruptures have occurred in domestic PWRs since 1975. These events occurred on February 26, 1975, at Point Beach Unit 1, September 15, 1976, at Surry Unit 2, October 2, 1979, at Prairie Island Unit 1 and on January 25, 1982, at R. E. Ginna. The first three of these events were evaluated in NUREG-0651, "Evaluation of Steam Generator Tube Rupture Events." The report includes an evaluation of system response, operator action, and radiological consequences during the three events. The leak rate associated with these events ranged from about 80 gpm to 390 gpm. The conclusion of the report is that no significant offsite doses or systems inadequacies occurred during the tube rupture events analyzed. However, the potential for more significant consequences was recognized and a number of procedural recommendations were made to correct the deficiencies that were noted. The present disposition of each of the recommendations is discussed in a recent memo to Commissioner Bradford from W. Dircks (Ref. 4). The present design basis for assuring that plants are acceptably protected against S.G. tube rupture events is a postulated double-ended rupture of a single S.G. tube. This assumption is intended to provide a bounding leak rate for a spectrum of rupture geometries in a single tube and a spectrum of smaller leaks in multiple tubes within a single S.G. The consequences of multiple tube failures, in excess of the design base, have not yet been rigorously studied. Rapid degradation between inspections of a large number of tubes could create the potential for multiple tube failures in the event of a plant transient or failure of a single tube and the accompanying jet impingement and tube whip could cause failure of additional tubes. Furthermore, the potential for complicating circumstances involving multiple equipment failures such as the stuck open PORV during the Ginna incident and possible steam bubble formation in the primary system have not been evaluated. Another concern is ruptures in multiple S.G.s. In this event, unless the plant can be rapidly depressurized and brought onto Residual Heat Removal, there is the potential to continuously lose emergency core cooling water outside of containment. The above concerns are being addressed as part of the TMI Action Plan. Item I.C.1 in the TMI Action Plan addresses S.G. tube failures coupled with other failures (such as a stuck open safety relief valve in the secondary system), ruptures of multiple tubes, and simultaneous ruptures in multiple S.G.s. The purpose of this effort is not to expand the plant design basis but to assure that operator emergency procedures provide proper guidance for safely controlling the plant during these types of events. Although rigorous analyses of many of the scenarios postulated above have not been completed, ISI, leak rate limits, and tube plugging requirements are intended to guard against such occurrences (See Section II). In summary, the consequences of S.G. tube ruptures under normal operating conditions have been small; however, such events can present a significant challenge to plant operators and safety systems.

During postulated accident conditions, such as main steam line break (MSLB), feedwater line break, or LOCA, the S.G. tubes are subject to increased pressure differentials and possible pressure waves (e.g., subcooled decompression phenomena) and vibrational loadings. These loads increase the potential for failure of degraded S.G. tubes which could exacerbate the accident sequence. In the event of MSLB, failed S.G. tubes would provide a leakage path from the primary to secondary system and several potential leak paths for radioactivity to the environment would then exist. In the event of a LOCA, the core reflood rate could be retarded by steam binding. This phenomenon is associated with a cold leg break, in which reflood of the core requires displacing steam generated in the core through the hot leg, the affected steam generator, and out of the cold leg break. S.G. tube failures would create a secondary to primary leak path which aggravates the steam binding effect and could lead to ineffective reflooding of the core. Analytical and experimental evaluations of this phenomenon are contained in References 4 and 5. Large MSLBs and LOCAs are considered extremely low probability events, but are postulated as bounding conditions. More realistic events might include small and intermediate size MSLBs or LOCAs. Although these postulated accidents pose a less severe challenge to S.G. tube integrity, tube rupture(s) leading to or following such events could have serious consequences. This is particularly true if fuel damage has occurred as in the case of Three Mile Island.

The final area of concern is the radiation exposure of personnel involved in S.G. inspection, repair, and replacement. Reference 3 presents a summary of data on S.G. related personnel exposure for selected plants from 1974 to 1980. In recent years, as much as 25% of some plants annual occupational exposure has resulted from routine S.G. inspection and maintenance and as high as 60% for S.G. replacement. Recent tube sleeving operations at San Onofre incurred 3500 man rem exposure and similar operations are planned for other plants.

II. Regulatory Approach

The NRC approach to assuring S.G. tube integrity under all operating conditions is based on <u>inservice inspection (ISI)</u>, primary to secondary leakage rate limits, and preventive tube plugging requirements. Guidance for performing ISI is provided in R.G. 1.83, "Inservice Inspection of S.G. Tubes," and plant technical specifications include requirements for ISI. Typical plant specifications require periodic inspections of 3% of the S.G. tubes in the plant and augmented ISI in the event tube degradation is detected. Required frequency of inspection is generally flexible enough to allow inspections to be performed concurrent with refueling outages. Certain incidents such as tube leakage require unscheduled ISIs. Furthermore, many plants with extensive degradation problems have licensing amendments imposing higher frequency and larger size inspections. The ISI requirements were developed largely through a combination of engineering judgement and operating experience. More rigorous statistically based ISI programs have been developed as part of Unresolved Safety Issues A-3, A-4, and A-5 (see Section V). The purpose of the required ISIs is to determine if tube degradation is occurring in the S.G., assess the rate of tube degradation based on results of successive inspections, and identify those tubes requiring plugging or repair.

Primary to secondary leak rate limits are an extremely important requirement for ensuring safe S.G. operation. Some forms of tube degradation have been observed to degrade tubes beyond the prescribed plugging limit during the interval between inspections. Technical Specification primary to secondary leak rate limits requiring shutdown, ISI, and corrective actions provide protection against unacceptable levels of degradation between inspections. Many serious conditions of tube degradation have been detected by monitoring of primary to secondary leakage and subsequent inspection. Primary to secondary leak rate limits exist in each plant's technical specifications. The bases for these limits are twofold. First, the leak rate limit ensures that the calculated dosage contribution from tube leakage will be limited to a small fraction of the allowable limits in the event of a S.G. tube rupture or MSLB. Second, the leak rate limit is intended to correspond to a defect size that would not be expected to result in tube rupture under normal or postulated accident conditions.

Finally, degradation limits for tube plugging exist in the plant Technical Specifications. Criteria for establishing the tube plugging limits are presented in R.G. 1.121, "Basis for Plugging Degraded Pressurized Water Reactor Steam Generator Tubes." These criteria require that the plugging limit include margins for eddy current testing error and continued degradation between inspections. Thus, it is important to have a good estimate of the rate of degradation based on successive ISI results and an understanding of the degradation phenomena.

The primary focus of the current NRC philosophy is directed at maintaining primary system integrity. This is accomplished primarily through the requirements described above for ISI, leak rate monitoring, and tube plugging. In a sense, it is directed at treating the symptoms and not the cause of S.G. degradation, which lies primarily in secondary system design and operations. This philosophy has been debated extensively, but the current position regards eliminating the problem at its source as an industry responsibility.

III. Current Corrective Actions

An effective solution to S.G. tube degradation problems would require major changes in S.G. mechanical design, thermal-hydraulics, materials selection, fabrication techniques, and changes in the secondary system design and operation. Elimination of S.G. degradation requires a systems approach integrating all of these considerations. There are no simple corrective actions. This is particularly true for those plants which have significant operating time and have experienced S.G. degradation. Design changes in operating S.G.s that would be necessary to eliminate degradation problems are virtually impossible. For example, tube to tubesheet crevices already contaminated with corrosive environments are virtually impossible to clean, carbon steel support plates cannot be replaced with more corrosion resistant materials, and residual fabrication stresses cannot be removed. Thus, corrective actions may prolong S.G. life, but tube degradation is expected to continue in operating plants. Once the secondary system is contaminated by an aggressive environment it is difficult to reverse the adverse affects. For example, caustic stress corrosion cracking and wastage, due to residual phosphate water chemistry conditions, still continue in some plants long after conversion to AVT water chemistry.

are in use. These fixes include such actions as tube sleeving, sludge lancing, soaking and flushing, reduced operating temperatures to slow corrosion, boric acid injection to arrest denting, support plate modifications to retard denting, S.G. replacement, and improvements in secondary system design and operation. Secondary system improvements include prompt correction of condenser in-leakage, condenser retubing, removal of copper based alloys from the secondary system, and addition of demineralizing systems. An industry constituted secondary water chemistry guidelines committee, under chairmanship of EPRI, is developing generic chemistry limits and operating guidelines. NRR has been in contact with this committee for the past year and will review a copy of the draft reports prior to issue. Chemical cleaning has also been proposed but has not been implemented due to uncertainties regarding its longer-term affect on S.G. integrity. Industry efforts are currently underway to eliminate these uncertainties and chemical cleaning may become a viable option in the near future. These fixes have met with varying degrees of success, but none of them is a panacea. Furthermore, short term solutions to one problem may create other problems. Conversion from phosphate to AVT water chemistry, which minimized wastage and stress corrosion cracking but was followed by denting, is a case in point.

Finally it should be noted that the majority of the plants under review for operating licenses have S.G.s of similar design to those currently in operation, so that the potential for S.G. tube degradation exists in these plants as well.

IV. NRC, Industry, and Foreign Research and Development Activities

NRC's steam generator research program addresses improved eddy current inspection techniques for steam generator tubing, stress corrosion cracking of steam generator tubing and evaluation of tube integrity.

The objective of the eddy-current program is to upgrade and improve eddy-current inspection probes, techniques and associated instrumentatio for inservice inspection of steam generator tubing to improve the ability to identify and characterize tube defects. Specific objectives include improving defect detection and characterization as affected by tube diameter and thickness variations, tube denting, probe wobble, tubesheet and tube support interference, and defect location and type.

The stress corrosion cracking program is developing data and models which will be used to predict the stress corrosion cracking initiation and service life of Inconel 600 steam generator tubing. The testing program includes variables which influence stress corrosion cracking such as temperature, stress, strain and strain rate, metallurgical structures and processing, and ingredients in the primary and secondary coolant.

A steam generator, with service induced degradation will be used for the validation of the accuracy and confidence limits of nondestructive inspection instrumentation and techniques; burst and collapse tests on field degraded tubes to validate tube integrity models; and for developing data for validation of previously developed stress corrosion cracking predictive models, chemical cleaning and decontamination, dose-rate reduction and secondary side characterization. In addition, statistically based sampling models for inservice inspection programs will be confirmed and/or improved utilizing the first ever confirmed data base.

There are many ongoing programs addressing S.G. issues a EPRI, most of which are sponsored by the S.G. Owner's Group, and t.. rest by EPRI itself. The programs address the following areas: (1) chemistry and corrosion, (2) materials selection and testing, (3) thermal hydraulic and structural testing and analysis, and (4) nondestructive examination (NDE). Efforts in the chemistry and corrosion area are directed at examining the causes of corrosion related degradation such as denting, intergranular attack, and stress corrosion cracking, and identifying potential fixes such as alternative secondary water chemistry treatments. Materials selection and testing efforts are directed at characterizing and evaluating the suitability of alternative tubing and S.G. materials. This includes consideration of new heat treatments for tubing and compatability of S.G. tubing with structural materials. Testing and analysis in thermal hydraulics and structures is directed at secondary side S.G. design and performance and their effect on S.G. tube integrity. The EPRI nondestructive examination programs focus on development of improved inspection techniques. These techniques include multiple frequency/multiparameter eddy current testing, automatic eddy current signal analysis, profilometry for quantifying dent configuration and strain levels in dented tubes, and methods for evaluating the condition of the tube support plates. In addition, EPRI has established the NDE Center in Charlotte, NC, dedicated to providing good NDE techniques, and effectively transferring research and development results to the industry.

Research and development activities underway on steam generators outside the USA are being funded at high levels in several countries. The Japanese are conducting a very large program with emphasis on thermal/hydraulics, and also on water chemistry and tube testing. To date, we have received little information on the progress or results of their programs. The French have work underway on eddy current NDE, crevice chemistry, and decontamination. There is work underway in Sweden on water chemistry. The Germans have work underway in eddy current NDE, and at KWU on primary side decontamination and secondary side cleaning; however, German steam generators are tubed with Incolloy 800 so much of their research is less relevant to ours. Finally, the Italians have underway a large program which will allow them to make new designs to avoid current and possible future problems.

V. Long Term Approach

A. Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity

In 1978, the NRC established Unresolved Safety Issues A-3, A-4, and A-5 (USI) regarding degradation in W, CE, and B&W steam generators, respectively. A draft report, NUREG-0844, presenting the proposed NRC staff resolution of these generic safety issues has been prepared and is currently being reviewed by NRR management prior to transmittal to the Committee for Review of Generic Requirements and the Commission and publication for public comment. The report integrates technical studies in the areas of systems analyses, inservice inspection (ISI), and tube integrity to establish improved criteria for ensuring adequate tube integrity and safe steam generator operation under all conditions.

In the systems analyses portion of the report, the consequences of steam generator tube failures during normal operation and postulated loss-of-coolant and main steam line break accidents are evaluated. The evaluation considers predicted fuel behavior, emergency core cooling system performance, radiological consequences, and containment response. The results of the systems analyses lead to proposed criteria for establishing a tolerable level of steam generator leakage during postulated accidents. ISI techniques are then evaluated, and statistically based ISI programs presented which, if implemented, would provide additional assurance that no more than the tolerable level of tube leakage, defined by the systems analyses, would occur during normal or postulated accident conditions.

In the tube integrity portion of the report, the behavior of degraded tubes during normal and postulated accident conditions and tube plugging criteria are evaluated. Proposed changes in operating procedures and design changes to minimize tube degradation are also identified.

Implementation of the proposed requirements and criteria developed in the program for resolution of the USI are not expected to totally eliminate S.G. degradation. The intent of the proposed requirements is to establish a logical approach to evaluating steam generator tube integrity and ensuring safe steam generator operation. The draft NUREG-0844 recommends criteria and requirements that can be used to evaluate current and future degradation programs in steam generators. The establishment of maximum allowable steam generator tube leak rates during postulated accident conditions and associated tolerable number of defective tubes is a major contribution to the evaluation of steam generator tube degradation problems. It provides objective criteria against which steam generator tube integrity can be evaluated. Similarly, the development of statistical ISI programs provides a rational, scientific basis that can be used to establish and evaluate ISI requirements that will ensure the above criteria are satisfied. Results from NRC S.G. research programs are expected to lay the experimental basis for many of these criteria.

In keeping with the NRC's current and past philosophy on this issue, the proposed regulatory requirements developed in the draft report focus on ISI programs and techniques and tube plugging criteria. The primary responsibility for attacking the problem at its source and eliminating S.G. degradation is the industry's. However, several of the requirements proposed in NUREG-0844 are intended to promote industry efforts in this area. For example, one requirement is to ensure that all operating plants have implemented an approved secondary water chemistry monitoring and control program. This is a requirement in the most recent version of the NRR standard review plan for licensing of new plants. In addition, this type of program has been implemented at some but not all operating plants. Under this requirement, it is the industry's responsibility to establish specific water chemistry limits and effective monitoring techniques. This will ensure that each utility at least considers the importance of secondary system water chemistry and puts in the effort to develop a comprehensive water chemistry program. Similarly, ISI requirements for condensers are proposed. These requirements will hopefully reduce the frequency of condenser in-leakage and encourage

utilities to improve condenser performance. Use of noncopper based alloys when retubing condensers and feedwater heaters is also a requirem. Additional requirements are proposed for plants in the preoperating license stage and many recommendations for operating and future plants are made. The intent of the proposed requirements as stated in the report is to leave primary responsibility for correcting the S.G. problem in the hands of the industry, to allow the industry flexibility in addressing the issue, but at the same time, to strongly encourage proper industry actions.

B. Comprehensive NRC/Industry Program

The preceeding review has attempted to summarize the status of the S.G. issue at this time. As indicated, the NRC has many ongoing efforts to address this multifaceted problem. However, to date, joint NRC and industry cooperative efforts on this issue have not been extensive. This is due largely to the different focuses on the issue. NRC is primarily concerned with requiring adequate ISI and corrective actions to ensure primary system integrity, while the industry has been concerned with developing fixes to prolong S.G. service life and reliability. NRC and industry efforts have been primarily complementary in nature. However, to the extent that reliability implies safety and vice-versa the NRC and industry efforts are synonomous. Therefore, the staff is pursuing the development of a joint NRC and industry program to address both near-term and long-term actions required for continued safe operation of steam generators and ultimate resolution of the S.G. degradation problem. The intent is to evaluate the degree to which the NRC can expand its role in prevention of tube degradation and work with the industry to solve this problem. Efforts to determine the feasibility of this type of cooperative program have been initiated and proposals for a joint NRC and industry program will be presented in a later document.

REFERENCES

1. Eisenhut, Liaw, Strosnider, "Summary of Operating Experience with Recirculating Steam Generators," NUREG-0523, January 1979.

2. Liaw, Strosnider, "Summary of Tube Integrity Operating Experience with Once-Through Steam Generator," NUREG-0571, March 1980.

3. SECY-81-664, "Information Report - Steam Generator Tube Experience," from W. J. Dircks to the Commissioners, November 24, 1981.

4. Memorandum for Commissioner Bradford from W. J. Dircks, Status of Recommendations Made in NUREG-0651, "Evaluation of Steam Generator Tube Rupture Events," to be transmitted.

5. EG&G Idaho, Inc. Report TREE-NUREG-1213 (NUREG/CR-0175), "Investigation of the Influence of Simulated Steam Generator Tube Ruptures During Loss-of-Coolant Experiments in Semiscale MOD-1 Systems," May 1978.

6. EG&G Idaho, Inc. Report CAAP-TR-032, "Steam Generator Tube Rupture - Effects on a LOCA," November 1978.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ATTACHMENT B

JAN 2 0 1982

Docket No.: 50-395

Mr. T. C. Nichols
Vice President and Group
Executive-Nuclear Operations
South Carolina Electric & Gas Company
P. O. Box 764
Columbia, South Carolina 29218

Dear Mr. Nichols:

Subject: Plans for Use of Westinghouse Model D Steam Generators

Within the last few months, there has been considerable interest paid to operating nuclear power plants with pre-heater steam generators (Model D) manufactured by Westinghouse Electric Corporation. This interest was initiated by tube failures and degradation in steam generators of this model at non-domestic nuclear power plants. We understand that you propose to use Westinghouse Model D pre-heater type steam generators at your facility.

Because of the safety concerns relative to steam generator tube damage, we consider the potential for such damage to be of safety significance. Therefore, we request that you provide us with your plans to address this problem at your facility within 30 days of receipt of this letter. We are especially interested to know whether or not you are relying on the results of the Westinghouse test program or testing at operating plants, what instrumentation you may propose for detection of flow-induced vibrations, and what testing and start-up procedures you propose for you own facility. Your full cooperation in this matter is appreciated.

The reporting and/or record keeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

cc: See next page

8202120047

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101 Marietta Street
Suite 3100
Atlanta, Georgia 30303

ATTACHMENT C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 20 1982

MEMORANDUM FOR: The Atomic Safety and Licensing Board
for Virgil C. Summer Nuclear Station

FROM: Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

SUBJECT: BOARD NOTIFICATION - PREHEATER TYPE STEAM GENERATOR (BN-82-02)

Enclosed is a memorandum dated December 7, 1981, providing a summary of a meeting held on November 20, 1981, with the Duke Power Company and Westinghouse Corporation. Information was presented regarding the results of testing of the Model D steam generator (similar to those at the Virgil C. Summer Nuclear Station) at two foreign reactors. Also enclosed is a letter from the Duke Power Company dated December 29, 1981, describing the results of its sequence of plant operation and steam generator inspection. To date, no indication of steam generator tube wear has been observed at the McGuire plant. The staff is closely monitoring the McGuire operation and steam generator test program and is evaluating information from the plant and the Westinghouse test program as it relates to the Virgil C. Summer Nuclear Station. We will keep the Board informed.

RLTedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosures:
1. Summary of Meeting held on November 20, 1981, dated December 7, 1981
2. Letter from Duke Power Company to NRC dated December 29, 1981

8202160079

Mr. T. C. Nichols, Jr.
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DISTRIBUTION OF BOARD NOTIFICATION

V.C. Summer
Docket No. 50-395

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From: Skip Copp
Duke Power
(2 pages)

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

December 29, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: McGuire Nuclear Station
Docket No. 50-369

Dear Mr. Denton:

On October 31, 1981 Duke Power Company notified the NRC Staff of tube degradation on a non-domestic Westinghouse plant with Model D steam generators similar to those at McGuire Nuclear Station. Subsequently a meeting was held on November 20, 1981 in Bethesda to brief the Staff on the details of the problem as well as Duke Power Company's plans for operation of McGuire. In particular Mr. H. B. Tucker outlined a planned sequence of plant operation, steam generator tube inspection and instrumentation installation. The purpose of this letter is to update the NRC Staff on the status of this effort to date and on plans for future operation.

Eddy current testing was performed on the 'A' Steam Generator to determine if a threshold power level existed at which tube vibration in the preheater was initiated. Rows 49, 48 and 47 were examined. This testing was conducted after two weeks operation at approximately 50% power and again after one week operation at 75% power. This testing was conducted by Babcock and Wilcox Company personnel utilizing a .590" diameter differential probe. A Zetec MIZ-12 multi-frequency apparatus was employed at frequencies of 130 khz, 200 khz, 400 khz and 550 khz. Since this examination was looking for wear damage at tube support plate locations both 130-550 khz and 200-400 khz mixed outputs were used to eliminate the support plate signal leaving only defect signals for analysis. An ASME Section XI type calibration standard was used.

Results of both of these inspections (i.e. after operation at 50% and 75% power) were reviewed by Babcock and Wilcox, Duke Power Company, Westinghouse and EPRI NDT personnel. A comparison with the results of the preservice inspection was made. No wear type indications were observed.

During the November outage, three transducers were mounted around the feedwater nozzle on each steam generator. These transducers are intended to provide an early indication of any gross mechanical vibration inside the preheater and will eventuall be used in conjunction with the internal instrumentation when installed. To date no signals have been noted which correlate to preheater/tube vibration. Resonance peak which have been observed were caused by flow turbulence rather than any mechanical vibration phenomenon.

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Currently, the unit is in the startup phase. Plans are to increase power to 90%, hold at that level for up to 4 days then increase to 100% for one day. Power operation would then continue for up to 6 weeks at power levels up to approximately 75%. The unit will then be shutdown, eddy current examination performed on all 4 steam generators and internal instrumentation mounted in one steam generator as described in our November 20, 1981 meeting in Bethesda. This operating plan represents our best efforts to balance testing and operational needs with a prudent course of actio: to assure the integrity of the steam generators. Minor changes to this planned sequ ence may occur due to unforseen circumstances; however, we will keep you advised of any significant departure from this plan.

Please advise if you have any questions regarding this matter.

Very truly yours,

William O. Parker, Jr.

GAC/jfw

cc: Mr. P. R. Bemis
Senior Resident Inspector
McGuire Nuclear Station

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

DEC 7 1981

Docket Nos: 50-369
and 50-370

LICENSEE: DUKE POWER COMPANY

FACILITY: McGuire Nuclear Station, Units 1 and 2

SUBJECT: SUMMARY OF MEETING HELD ON NOVEMBER 20, 1981

A meeting was held with the licensee on November 20, 1981, to discuss preheater-type steam generator tube problems in foreign reactors as related to the operation of McGuire Unit No. 1. A list of attendees is shown on Enclosure 1.

The major briefing presentation was made by Westinghouse and included the results to date of testing of the Model D steam generator at two foreign reactors. These steam generators are of similar design (Model D) to those in the McGuire Unit 1 plant. Eddy Current testing has revealed that there are indications in the outer rows of tubes in essentially all steam generators so tested. Westinghouse has initiated a program of testing and tube examination along with analytical evaluations to determine the initiator of this tube degradation phenomena. Westinghouse believes at this time that tube degradation can be attributed to excitation of the tubes from high fluid velocities and/or non-uniform velocity distribution.

The licensee stated that McGuire Unit 1 was shut down on November 16, 1981 and that an operating plan for Unit 1 had been developed (see Enclosure 2). Rows 47, 48 and 49 in "A" Steam Generator (S/G) were Eddy Current tested. The results were negative with possibly one slight indication. The licensee has installed external vibration monitors (3 transducers per S/G) on each S/G. Upon completion of this inspection effort, the unit will be restarted and a power level of 75% established, approx. November 23. Upon completion of the traditional 75% plateau power ascension testing (2 weeks), the unit will be shutdown and S/G Eddy Current testing repeated on S/G "A". During unit operation, monitor instrumentation will be evaluated. Interim operation on the above basis appears appropriate at this time.

The licensee indicated that further operation at 90% power is contemplated to complete some ascension testing (1 week) provided no indications above 20% are discovered in S/G "A". Following this week of testing, evaluation of the monitoring, and evaluation of 50-90% power data, they would decide on escalation to 100% power for one day. Data evaluation at this time would determine whether or not the licensee would plan to continue operation at an acceptable power level or shut down at that time for further EC testing.

8112300206

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Since the information presented by Westinghouse was proprietary, the licensee agreed to document the information pursuant to 10 CFR 2.790.

/s/

Ralph A. Birkel, Project Manager
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Enclosures:
1. Attendance List
2. McGuire Nuclear Station, Unit 1, Operating Plan

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November 20, 1981

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McGUIRE NUCLEAR STATION - UNIT 1

OPERATING PLAN

- o CONDUCT EC EXAMINATION

  "A" S/G

  ROWS 49, 48, 47, [redacted]

- o INSTALL EXTERNAL INSTRUMENTATION

  "A" S/G

- o EVALUATE EC EXAMINATION RESULTS

  IF INDICATIONS <20% CONTINUE OPERATION

  IF INDICATIONS >20%, INSPECT 4 S/G, EVALUATE

- o ESCALATE TO 75% POWER, COMPLETE TESTS, (~2 WEEKS)

  - MONITOR INSTRUMENTATION, EVALUATE

- o SHUTDOWN - CONDUCT EC EXAMINATION

  "A" S/G

  ROWS 49, 48, 47, [redacted]

- o EVALUATE EC EXAMINATION RESULTS

  IF INDICATIONS <20% CONTINUE OPERATION

  IF INDICATIONS >20%, INSPECT 4 S/G, EVALUATE

MC GUIRE NUCLEAR STATION - UNIT 1

OPERATING PLAM

- ESCALATE TO 90%, COMPLETE TESTS (1WEEK)
- MONITOR INSTRUMENTATION, EVALUATE
- EVALUATE 50-90% POWER DATA
  - IF SATISFACTORY, CONTINUE OPERATION
  - IF UNSATISFACTORY, OPERATE AT REDUCED POWER
- ESCALATE POWER TO 100% (MAXIMUM 1 DAY)
- AT THIS POINT EITHER:
  1) OPERATE AT AN ACCEPTABLE POWER LEVEL BASED ON DATA AVAILABLE AT THIS TIME
  2) SHUTDOWN FOR EC INSPECTION AND INSTALLATION OF ADDITIONAL INSTRUMENTATION
- WHILE SHUTDOWN

  ECT INSPECTION 4 S/G

  INSTALL INTERNAL INSTRUMENTATION 2 TUBES, 1 S/G

  EVALUATE 50 - 100% POWER OPERATION PLUS OTHER DATA
- DETERMINE APPROPRIATE NEAR TERM OPERATING CONDITIONS

  USE OF COMBINED MAIN, AUXILIARY FEED NOZZLES

  POWER LEVEL
- RETURN TO POWER

ATTACHMENT D

MEETING SUMMARY DISTRIBUTION

MAR 1 ^ 1982

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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50-369, 50-370, 50-390, 50-391, STN 50-546,
STN 50-547, 50-445, 50-446, 50-395, 50-413,
50-414, 50-498, 50-499, 50-400, 50-401, 50-402
and 50-403

MEMORANDUM FOR: B. J. Youngblood, Chief, Licensing Branch No. 1, DL

FROM: S. Chesnut, Project Manager, Licensing Branch No. 1, DL

SUBJECT: SUMMARY OF MEETING ON WESTINGHOUSE MODEL D STEAM GENERATORS

A meeting was held February 19, 1982 in Bethesda, Maryland with representatives from Westinghouse and various utilities which had purchased Westinghouse Model D steam generators. A list of attendees is given in Enclosure (1).

Westinghouse representatives presented the results of their initial evaluations of accelerated tube wear on Model D steam generators noted at two non-domestic plants in Ringhals, Sweden (Model D3) and Almaraz, Spain (Model D3). Eddy current data and information derived from internal and/or external flow monitoring instrumentation of scale models and other Model D steam generators at McGuire (D2) and Krsko, Yugoslavia (D4) was also discussed.

Westinghouse representatives reported the the accelerated tube wear was flow related and had only been observed at two non-domestic plants. The accelerated wear was attributed to turbulence in the preheater region caused by the feed inlet impingement plate and flow limiter. Initial data from the instrumented steam generators showed that the onset of the increased turbulence occured at high feed flow rates (approximately 50% for Models D2 and D3, 70% for D4, D5).

Westinghouse indicated that increased instrumentation and additional analyses would continue until a resolution was reached. Westinghouse also discussed several preliminary design change possibilities which would decrease the flow-induced vibrations and turbulence. Westinghouse indicated that it would make recommendations to utilities but that each utility would respond to the staff as to appropriate measures to prevent the accelerated tube wear.

S. H. Chesnut

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Enclosure:
As stated

cc: See next page

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February 19, 1982

Mr. Harold R. Denton, Director
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Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Westinghouse Model D-3
Steam Generators

Dear Mr. Denton:

On January 22, 1982, South Carolina Electric and Gas Company (SCE&G) received Mr. D. G. Eisenhut's January 20, 1982 letter requesting a response within thirty days regarding plants to address problems associated with the preheater section of Westinghouse Model D steam generators. SCE&G's present plan is provided in this letter.

The Virgil C. Summer Nuclear Station utilizes three (3) Westinghouse Model D-3 steam generators. The Model D-3 incorporates a split flow design where feedwater flow enters at a mid section of the preheater section and splits into an upward and downward flow around the tubes and baffles.

The tube wear mechanism was first identified in the Model D-3 steam generator preheater area at a non-domestic plant. After 113 effective full power days of operation, this plant experienced a primary to secondary leak in one tube in the preheater section. Inspections revealed wear on other tubes in the outer three rows in this area.

Another non-domestic plant which had a similar operating history found similar indications but of less magnitude.

SCE&G is aware of other operating plants having Model D-2 and D-4 steam generators, and are following their progress as they proceed through their startup testing and operation.

As follow-up to previous meetings and frequent telephone discussions, SCE&G met with Westinghouse on February 10, 1982 to discuss the nature of the mechanisms observed. We discussed with Westinghouse the program that is in place to understand the nature of the phenomena to permit operation of all preheat steam generators. The Westinghouse program includes the use of scale model testing,

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analytical studies, wear testing, periodic eddy current inspection of operating units and the use of diagnostic internal and external instrumentation in selected operating plants.

As stated, installation of internal acceleometers has been used by Westinghouse as a diagnostic tool for verification of the analytical studies and concurrent lab testing. At the present time, diagnostic internal instrumentation is installed or being installed in a Model D-2 unit domestic plant and in Model D-3 and D-4 in non-domestic plants. Field data collection from the diagnostic instrumentation has adequately met the verification needs of the Westinghouse evaluation program and the diverse type Model D units instrumented makes it unnecessary to further instrument Model D units at this time. Therefore, Westinghouse has concluded that in-plant, on-line, data availability is not needed for safe operation of the Virgil C. Summer Nuclear. The conclusion for not installing internal instrumentation at Virgil C. Summer Nuclear Station is based on the following:

1) The similarities with the Model D-3 unit currently instrumented and the adequacy of diagnostic instrument data currently collected.

2) The results to date of the Westinghouse evaluation program including analyses, lab testing and operation plant data, support the start-up program and safe level of power operation defined below.

In our review of the Westinghouse program, we have determined that the results of the Westinghouse anaylsis and test programs and the experience and data gained from these two operating plants appear to be adequate. Accordingly, our current program is outlined below:

* Conduct normal low power testing.

* Conduct power escalation testing up to 50% power to complete the startup testing up to and including that level.

* Continue operation at 50% power for approximately 2 months or at a power level above 50% that has been evaluated, based upon information available at that time, to preclude significant tube damage.

* Shut down and eddy current test rows 49, 48, and 47 of one steam generator.

* Reevaluate available data to confirm continued limited power operation capability until a modification can be accomplished to resolve the problem.

* The proposed schedule may be altered at any time to perform modifications.

Following completion of this phase of the program SCE&G and Westinghouse will jointly establish an operating power level for operation of the Virgil C. Summer Nuclear Station prior to modification. In determining this power level consideration will be given to:

* Results of the eddy current inspection.

* The experience and data from other operating plants.

* The status of the Westinghouse analysis and test program.

* The status of any proposed Westinghouse modification to alleviate or resolve the problem.

In summary SCE&G has reviewed the Westinghouse program to approach this problem and will proceed with the outline of activities as described above. Additional details and scheduling information will be provided to the NRC as they become available.

If you have any questions, please let us know.

Very truly yours,

T. C. Nichols, Jr.

T. C. Nichols, Jr.

RBC:TCN:lkb

cc: Page 4

cc: V. C. Summer
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ATTACHMENT F

SOUTH CAROLINA ELECTRIC & GAS COMPANY
POST OFFICE BOX 764
COLUMBIA, SOUTH CAROLINA 29218

T. C. NICHOLS, JR.
VICE PRESIDENT AND GROUP EXECUTIVE
NUCLEAR OPERATIONS

April 14, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Steam Generator Instrumentation

Dear Mr. Denton:

This is to inform you that South Carolina Electric and Gas Company (SCE&G) will install internal instrumentation in the A steam generator at the Virgil C. Summer Nuclear Station. The instrumentation will consist primarily of biaxial accelerometers and will be installed in two tubes as selected by Westinghouse and SCE&G. We expect that work will begin on installation of this equipment in early May and will be complete within a few weeks.

If you require further information, please contact us.

Very truly yours,

T. C. Nichols, Jr.

T. C. Nichols, Jr.

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