J. Barnie Beasley, Jr., P.E.
Vice President
Vogtle Project

Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway
P.O. Box 1295
Birmingham, Alabama 35201

Tel 205.992.7110
Fax 205.992.0403

SOUTHERN COMPANY
Energy to Serve Your World℠

April 26, 2000

LCV-1016-P

Docket No.: 50-424

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

Ladies and Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
REQUEST FOR RELIEF - STEAM GENERATOR
NOZZLE INNER RADIUS EXAMINATION

Enclosed for NRC review and approval is Request for Relief RR-32 that pertains to the VT-1 visual examination of the Vogtle Electric Generating Plant, Unit 1 (VEGP-1), steam generator nozzles. Relief is requested from the scheduling requirements of the 1989 Edition of the ASME Code, Section XI, IWB-2412, and is applicable to VEGP-1 only. The NRC mandated in its December 31, 1998, letter to Southern Nuclear Operating Company (SNC) that a VT-1 visual examination of the steam generator nozzles inner radius be performed in lieu of the ultrasonic examinations that the ASME Code, Section XI, normally requires. Relief from performing the ultrasonic examinations of the nozzle inner radii was granted by the NRC in response to our Request for Relief RR-6.

Subsequent to the submittal of the second ten-year interval request for relief (RR-6), industry has pursued with the NRC, through the ASME Code, the elimination of nozzle inner radius examinations on Class 1 vessels (steam generators and pressurizers). Information has been presented by the industry to the NRC indicating that no evidence of cracking or other deterioration of the nozzle inner radius area of these particular vessels has been observed. Further, fracture mechanics work performed by an industry representative has shown that these areas should not experience any degradation as a result of plant operation. Based on these findings, the industry has supported ASME Code, Section XI, Code Case N-619, which authorizes elimination of these examinations for Class 1 vessels (steam generators and pressurizers). The provisions of that Code Case have been incorporated into the 1999 Addenda to the ASME Code, Section XI, and the requirements for the nozzle inner radius examinations for the steam generators and

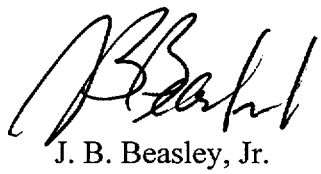
A047

pressurizers have been removed from the Code tables, although not yet approved by the NRC. Although positive information has been provided to justify elimination of the nozzle inner radius examinations for the steam generators and pressurizers, additional information has been requested of the industry by the NRC and involves the scope of examinations, quality, cost, and personnel radiation exposure of past examinations.

Southern Nuclear Operating Company requests NRC approval of the enclosed request for relief that would allow re-scheduling of the VEGP-1 steam generator nozzle inner radius examinations to such a time that the nozzle inner radius examination issue is resolved between the industry and the NRC. Your approval is requested by August 1, 2000, in order to support the upcoming VEGP-1 Ninth Maintenance/Refueling Outage (1R9) currently scheduled to begin September 17, 2000. The courtesy of a response by August 1 would be appreciated in the event that relief is denied and other plans have to be made.

Should there be any questions in this regard, please contact this office.

Sincerely,



J. B. Beasley, Jr.

JBB/JAE/

Enclosure: Vogtle Electric Generating Plant, Unit 1 - Request for Relief RR-32 - Steam Generator Nozzle Inner Radius Examination

xc: Southern Nuclear Operating Company
Mr. W. L. Burmeister (w/o enclosure)
Mr. J. T. Gasser (w/o enclosure)
Mr. M. Sheibani (w/enclosure)
SNC Document Management (w/enclosure)

U. S. Nuclear Regulatory Commission
Mr. R. R. Assa, Project Manager, NRR (w/o enclosure)
Mr. L. A. Reyes, Regional Administrator (w/enclosure)
Mr. J. Zeiler, Senior Resident Inspector (w/enclosure)

ENCLOSURE
TO
SOUTHERN NUCLEAR OPERATING COMPANY
LETTER LCV-1016-P

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

REQUEST FOR RELIEF RR-32 - STEAM GENERATOR
NOZZLE INNER RADIUS EXAMINATION

SOUTHERN NUCLEAR OPERATING COMPANY
VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2
SECOND TEN-YEAR INTERVAL
REQUEST FOR RELIEF NO. RR-32

I. System/Component(s) for Which Relief is Requested:

Class 1 steam generator nozzle inner radius (IR) examinations. This request for relief applies *only* to VEGP-1 and includes the following examinations:

11201-B6-002-IR-01 (Steam Generator #2 Inlet Nozzle Inner Radius)
11201-B6-002-IR-02 (Steam Generator #2 Outlet Nozzle Inner Radius)

II. Code Requirements:

Table IWB-2500-1, Category B-D, Item Number B3.140, of the 1989 Edition of the ASME Section XI Code requires that the inner radius volume of the steam generator inlet and outlet nozzles be volumetrically examined. Additionally, Table IWB-2412-1 requires that 16% to 34% of the total Category B-D examinations (except where deferral to the end of the interval is permitted) must be completed during the first period.

III. Code Requirement from Which Relief is Requested:

The steam generator inlet and outlet nozzle configuration and surface conditions effectively prohibit the performance of meaningful ultrasonic examinations. This position was established during the First Ten-Year (FTY) interval and documented in Relief Request RR-42 for that interval. As an alternative, visual examinations of the inner radii surface areas were performed during the FTY interval in lieu of the required ultrasonic examinations. Second Ten-Year (STY) interval relief request RR-6 requested approval to delete the ultrasonic examinations and was granted in the Safety Evaluation Report, provided in the NRC letter dated December 31, 1998. Although, the NRC approved RR-6, they added the requirement that a VT-1 visual examination of the inner radii surface areas be performed during the STY interval.

Southern Nuclear requests authorization to reschedule the performance of the STY interval, VEGP-1 steam generator inlet and outlet nozzle VT-1 examinations as shown in Table 1 to this request for relief. The industry is currently interfacing with the NRC in an effort to optimize vessel inner radius examinations. The proposed alternative schedule would allow VEGP-1 to defer further VT-1 examinations until the second period, by which time the industry is expected to resolve the issue of nozzle inner radius examination optimization with the NRC. It also takes into consideration the VEGP-1 eddy-current examination schedule, which affects steam generators to be opened for access.

(continued)

IV. Basis for Relief

Performing these examinations per the current schedule identified in Table 1 to meet the IWB-2412 Code requirements would require performing the VT-1 examinations of the nozzle inner radii for Steam Generator 2 during the Fall 2000 maintenance/refueling outage. Performance of the VT-1 examinations, during the period of time discussions are being held between the industry and the NRC to optimize the examination requirements, constitutes a hardship to SNC without a compensating increase in the level of quality and safety. The bases for this determination is shown below:

Determination of Hardship

Performance of the required VT-1 examinations requires that SNC perform either a direct VT-1 examination from the steam generator bowl area or a remote VT-1 examination using a camera or other suitable equipment. During its seventh maintenance/refueling outage in Fall 1999, the measured dose rate on the sister unit to VEGP-1, VEGP-2, was 5000 to 8000 mRem/hr in the center of the steam generator bowl and 8000 to 12000 mRem/hr in contact with the bowl surface. In addition, the measured dose rate was 1500 to 2000 mRem/hr at the plane of the manway and 800 mRem/hr on the platform just outside the manway. Similar, if not greater dose rates, are expected on VEGP-1 which has been in operation one cycle longer than VEGP-2. Using the dose rates observed on VEGP-2, the expected doses to perform the examination on the two nozzles of one steam generator (contingent upon whether the examination is performed by direct or remote means) are:

Direct VT-1 – Direct VT-1 of the inlet and outlet nozzle inner radius areas would require ingress/egress into both sides of the steam generator bowl and performance of the VT-1. With only 3 minutes in each side of the bowl to perform the examination, the total dose for examination of one steam generator (both nozzles) would be in excess of 800 mRem.

Remote VT-1 – During the time the eddy current equipment/camera is in a steam generator bowl, the nozzle dams are in place and the nozzle inner radius cannot be VT-1 examined. Therefore, a camera setup would be required outside of the timeframe in which the nozzle dams are installed. Setting up a camera to perform a remote VT-1 examination in one steam generator requires approximately 10 minutes at each manway, resulting in a total dose (both nozzles) in excess of 600 mRem.

(continued)

IV. Basis for Relief (continued)

Determination of Safety

Typical steam generator primary nozzle inner radii were designed such that fatigue usage factors are low; therefore, it is unlikely that fatigue cracking would initiate in these areas. In the unlikely event that a crack did initiate, industry fracture mechanics evaluations, based on conservative assumptions, have demonstrated that these nozzles have a large tolerance for flaws. In addition, probabilistic risk assessment calculations, performed with or without the required inservice inspections, gave such small probabilities of failure that the gain from inspection is considered to be meaningless.

The steam generator nozzle inner radii surface areas for Steam Generators 1, 2, 3, and 4 were visually examined during the FTY interval and no cracking or flaws (fabrication or service-induced) were observed. Therefore, there were no observable fabrication flaws that could serve as a localized crack initiation site. With the low propensity for fatigue cracking to initiate and propagate to a critical size in these nozzles and with the absence of observable ID crack initiation sites, the proposed examination schedule should continue to provide reasonable assurance that the structural integrity of the nozzles will be maintained.

V. Alternate Examination:

Examinations will be conducted in accordance with the proposed alternative schedule identified in Table 1 to this request for relief.

VI. Justification for Granting Relief:

Complying with the requirements of the NRC's December 31, 1998 Safety Evaluation Report to perform VT-1 visual examinations of the inner radii of the VEGP-1 steam generator nozzles, as currently scheduled in accordance with Table IWB-2412-1, is a hardship due to the associated radiation dose. Taking into consideration the favorable FTY interval visual examination results and the large tolerance for flaws in this region of the nozzle, it is concluded that no compensating increase in safety would be achieved by performing the examinations in accordance with the requirements of the Safety Evaluation Report in lieu of the proposed alternative examination schedule. Therefore, the proposed alternative examination schedule should be granted per 10CFR50.55a(a)(3)(ii).

(continued)

VII. Implementation Schedule:

The proposed schedule for examining the nozzle inner radii of the VEGP-1 steam generators will be implemented during the STY interval that commenced May 31, 1997.

(continued)

Table 1

**Nozzle Inner Radii Examination Schedule
for
VEGP-1 Steam Generators**

Steam Generator Number	Date/Period FTY Interval Visual Exam Performed	Current STY IWB-2412-1 Schedule for Performance of VT-1 Exam	Proposed STY Alternative Schedule for Performing VT-1 Exam
1	Fall 1994/ Period 3	Period 3	Period 3
2	Spring 1990/ Period 1	Period 1	Period 2
3	Spring 1993/ Period 2	Period 2	Period 2
4	Fall 1994/ Period 3	Period 3	Period 3