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Vogtle Project

April 29, 1985

Director of Nuclear Reactor Regulation
Attention: Ms. Elinor G. Adensam, Chief
Licensing Branch #4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

File: X7BC35
Log: GN-588

NRC DOCKET NUMBERS 50-424 AND 50-425
CONSTRUCTION PERMIT NUMBERS CPPR-108 AND CPPR-109
VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2
REQUESTS FOR ADDITIONAL INFORMATION: DSER OPEN ITEMS

Dear Mr. Denton:

Your staff has requested additional information as part of the VEGP review process. Attached is a listing of the DSER open items, the enclosures where they are addressed and the source of the request.

If your staff requires any additional information, please do not hesitate to contact me.

Sincerely,

J. A. Bailey

J. A. Bailey
Project Licensing Manager

JAB/sm
Enclosure
xc: D. O. Foster
R. A. Thomas
J. E. Joiner, Esquire
B. W. Churchill, Esquire
M. A. Miller
B. Jones, Esquire (w/o enclosure)
L. T. Gucwa
G. Bockhold, Jr.
T. Johnson (w/o enclosure)
D. C. Teper (w/o enclosure)
L. Fowler
Vogtle Project File

0205m

B001
1/1

ATTACHMENT

OPEN ITEMS INDEX

Open Item	Enclosure	Remarks
105	A	Response to NRC teleconference on 4/22/84 concerning Regulatory Guide 1.94 and position C.2 of Regulatory Guide 1.58. This information will appear in Amendment 17.
51	B	Clarification of GN-514, 4/3/85, submittal and includes information request telecopied to SCS on 4/16/85.
ASB	C	Additional information requested by NRC letter dated 2/14/85. This submittal completes response to letter request and will be included in Amendment 16.
24	D	Revision to Q210.48 on allowable leakage rate for pressure isolation valves. This change will appear in Amendment 17.
97	E	Clarification of change concerning STA's absent from duty for a period of thirty days.
103	F	Additional information concerning steam generator tube rupture issue. This information will appear in Amendment 16.

0205m

ENCLOSURE A

1.9.93.2 VEGP Position

VEGP will conform with this guide by implementing the appropriate NRC approved standard Technical Specifications.

Refer to the Technical Specifications for further discussion.

1.9.94 REGULATORY GUIDE 1.94, REVISION 1, APRIL 1976, QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

1.9.94.1 Regulatory Guide 1.94 Position

This guide describes a method acceptable to the NRC for complying with the quality assurance requirements for installation, inspection, and testing of structural concrete and structural steel during the construction phase of nuclear power plants. This guide endorses ANSI N45.2.5-1974 as generally acceptable to the NRC as a basis for complying with Appendix B to 10 CFR 50.

1.9.94.2 VEGP Position

INSERT 1.9.94.2

~~The extent of conformance with ANSI N45.2.5-1974 for both the operational and construction phases is discussed in paragraph 3.8.3.6.2.C.~~ 15

Refer to Regulatory Guide 1.55 comparison for a discussion of the standards being used in the placement of concrete in Category 1 structures.

1.9.95 REGULATORY GUIDE 1.95, REVISION 1, JANUARY 1977, PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE 3

1.9.95.1 Regulatory Guide 1.95 Position

This guide describes design features and procedures that are acceptable to the NRC for the protection of nuclear plant control room operators against an accidental chlorine release.

Amend. 3 1/84
Amend. 9 8/84
Amend. 15 3/85

Insert 1.9.94.2

VEGP conforms with requirements of ANSI N45.2.5-1974 as it is endorsed by Regulatory Guide 1.94, Revision 1, for both the construction and operations phases with the following clarifications:

1. Paragraph 1.5, Referenced Documents.
   This paragrah states that for documents referred to in ANSI N45.2.5-1974, that the specific issue or edition of these documents that is required is specified in the Appendix to ANSI N45.2.5. Also, Paragraph 8 states that when standards referred to in ANSI N45.,2.5-1974 are superseded by an approved revision, then the revision shall apply for the VEGP, the specific issue or edition of documents and standards referenced in ANSI N45.2.5-1974 that shall apply is addressed within the appropriate sections of the VEGP FSAR.

2. Paragraph 3.2, Materials Suitability (3.2.1).
   This paragraph, Table A, states reinforcement shall be tested for physical properties per (ASIM A615) ASTM A370. VEGP conforms to this guidance in that the certification of compliance provided by the vender ensures materials have been tested for physical properties (ASTM A615) per ASTM A370.

3. Paragraph 4.8, In-process Tests on Concrete and Reinforcing Steel.
   This paragraph, Table B, requires in-process compressive strength tests shall be performed daily on grout in accordance with ASTM C109. VEGP tests batch plant grout daily in accordance with ASTM C109, while non-shrink grout is tested in accordance with CRD621-82A prior to its initial use on the job site.

4. Paragraph 4.8, In-process Tests on Concrete and Reinforcing Steel.
   This paragraph, Table B, requires that fly ash and pozzolans be checked for physical properties (ASTM C618) in accordance with ASTM C311 every 200 tons. VEGP specifications require this testing to be performed every 1000 tons except for loss on ignition and sieve No. 325 which are performed every 200 tons.

5. Paragraph 4.9, Mechanical (Caldweld) Spliced Testing.
   (4.9.2) This paragraph requires visual inspection on completed splices shall be performed only after the splices have cooled to ambient temperatures. The VEGP specification does not require splices to be cooled to ambient temperature prior to inspection. Caldwelds are inspected after they have cooled such that it will allow the inspection to be performed without any danger of burns.

6. Paragraph 5.5., Welding.
   This paragraph requires inspection of structural steel welding to be performed in accordance with the provisions of AWS D1.1, section 6 entitled "Structural Welding Code." The visual acceptance criteria is in accordance with AWS D1.1-75 with the clarifications and modifications described on the next page.

In order to designate the specific set of visual acceptance criteria applicable for a weld, structural steel weld joints within the jurisdiction of AWS D1.1 are classified into the following categories:

- o Category A - Structural steel joints which are a part of the main building frame and those joints which connect miscellaneous structural steel with the main building frame.

- o Category B - Miscellaneous structural steel joints, not covered in Category A, but provide auxiliary support or framing for systems, components, and equipment (e.g., supports for cable tray and HVAC ductwork, miscellaneous stiffeners and bracing, etc.).

The base metal adjacent to a weld joint of dissimilar categories shall meet the visual inspection acceptance criteria of the category under which that base metal falls.

A. AWS D1.1, paragraph 3.1.4, is clarified as follows:

For category A and B joints, the fillet leg dimension may underrun the nominal fillet size by 1/16 inch provided the undersize length does not exceed 10 percent of the weld length. For flange to web joints, the undersize may not be within two (2) flange thicknesses of the weld end.

For Category A and B joints, where an intermittent weld is specified, a continuous weld of the same size is acceptable.

B. AWS D1.1, paragraph 3.6.1, is modified as follows:

For Category A and B joints, the faces fo fillet welds may be slightly convex, flat, or slightly concave as shown in Figure 3.6A, B, and C, with none of the unacceptable profiles shown in Figure 3.6.D. Convexity height may not exceed 1/8 inch.

C. AWS D1.1, paragraph 3.6.4, is modified as follows:

For Category A joints, undercut not exceeding 1/32 inch for the full length of the weld is acceptable. For members welded from both sides, the cumulative undercut depth includes the sum of both sides if the undercuts on both sides are directly opposite to each other.

For Category B joints, undercut not exceeding 1/32 inch for the full length of the weld is acceptable. Undercut greater than 1/32 inch but not exceeding 1/18 inch is acceptable provided the width is greater than the depth, and the undercut does not exhibit an acute intersection at its root. The cumulative length of 1/16 inch undercut shall not exceed 50 percent of the total weld length. For members welded from both sides, the cumulative undercut depth or length includes the sum of both sides of the undercuts on both sides are directly opposite to each other.

D. AWS D1.1, paragraph 3.6.6, is modified as follows:

For Category A and B joints, overlap/rollover may not exceed 1/8 inch.

E. AWS D1.1, paragraph 3.10.1, is clarified as follows:

For Category A and B joints, "Trap" slag on the root and/or end of fillet and partial penetration welds is acceptable.

For Category A joints, loose spatter is not acceptable. Tightly adhering spatter is acceptable, if it will not interfere with subsequent NDE (when required).

For Category B joints, spatter is not a cause for rejection.

F. AWS D1.1, paragraph 4.4, is modified as follows:

For Category A joints, arc strikes shall be blended with the surface.

For Cateogry B Joints, arc strikes are acceptable provided the craters do not contain cracks and the material specification minimum gauge thickness is not violated.

G. AWS D1.1, paragraph 8.15.1.3, is modified as follows:

Underfill of a weld is not acceptable. A crater is not considered as an underfill.

For Category A joints, underfilled grove weld craters are acceptable provided the depth of underfill is 1/32 inch or less. Underfilled single-pass fillet weld craters are acceptable provided the crater length is less than 10 percent of the weld length. On multi-pass fillet weld, crater depth of 1/32 inch or less is acceptable.

For Category B joints, underfilled groove weld craters are acceptable provided the depth of underfill is 1/16 inch or less. Underfilled single-pass fillet weld craters are acceptable provided the crater length is less than 10 percent of the weld length. On multi-pass fillet welds, crater depth of 1/16 inch or less are acceptable.

H. AWS D1.1, paragraph 8.15.1.5 , is clarified as follows:

For Category A joints, the sum diameters of porosity shall not exceed 3/8 inch in one (1) linear inch of weld and 3/4 inch in 12 linear inches of weld.

For Category B joints, piping porosity is not a cause for rejection if the major axis is equal to or less than 1/16 inch. For porosity greater than 1/16 inch, the sum of the diameters shall not exceed 3/8 inch in any linear inch of weld nor 3/4 inch in any 12 inches of weld.

C. The extent of compliance with ANSI N45.2.5-1974 is as ~~follows~~: DESCRIBED IN PARAGRAPH 1.9.94.2.

1. ~~Preplacement inspection of concrete forms.~~

2. Inspection of concrete placement.

3. Inspection of concrete curing.

4. Inspection and finishing of repairs.

5. Records kept per ANSI N45.2-1973.

6. Definitions (per ANSI N45.2.10-1973).

7. Concrete quality and acceptance criteria conformance with ACI 318-71.

8. Cadweld splices conformance with Regulatory Guide 1.10.

9. ANSI N45.2-1971, N45.2.2-1972, N45.2.3-1973, N45.2.9-1973, and N45.2.10-1973 complied with as referenced.

10. ~~Inspection personnel qualified per ANSI~~ N45.2.6-1973.

3.8.3.6.3 Special Construction Techniques

There are no special construction techniques used in the construction of the internal structures.

3.8.3.7 Testing and Inservice Inspection Requirements

A formal program of testing and inservice surveillance is not required for the internal structures. Tests and inspections for the RCS component supports are discussed in subsection 5.4.14.

3.8.3.8 Standard Review Plan Evaluation

The Standard Review Plan specifies ACI 349, augmented by Regulatory Guide 1.142, as the acceptable code for design of 1
concrete structures. The Standard Review Plan also specifies the load combinations that would result from the use of ACI 349, as modified by Regulatory Guide 1.142. VEGP design is based on ACI 318-71 and is in conformance with the load combinations

Clarifications are as follows: 9

REPLACE WITH ATTACHMENT

1. Paragraph 1.2 of ANSI N45.2.6-1978, Applicability. VEGP personnel who approve preoperational, startup, and test results and who direct or supervise the conduct of individual preoperational, startup, and operational tests shall be qualified in accordance with the VEGP position to Regulatory Guide 1.8 in lieu of being qualified to ANSI N45.2.6 as allowed by Regulatory Position C.1 of Regulatory Guide 1.58 Rev. 1. VEGP personnel who perform NDEs shall meet the requirements of ANST "Recommended Practice No. SNT-TC-1A" in 15 accordance with regulatory position C.2 of Regulatory Guide 1.58 Rev. 1. For nuclear operating personnel, VEGP shall apply the requirements of this guide to quality control inspection personnal; however, for personnel performing calibration, installation checkouts, or routine surveillance, the requirements of this guide shall not apply since, as stated in Section 1.2 of ANSI N45.2.6, the requirements of this guide are optional for these personnel. 9

2. Paragraph 2.5 of ANSI N45.2.6-1978, Physical. VEGP will implement the requirements of this section with the stipulation that, where no special physical characteristics are required, none will be specified. The converse is also true; if no special physical requirements are stipulated by VEGP, none are considered necessary. GPC employees receive an initial physical examination to assure satisfactory physical condition; GPC management shall determine which personnel are required to receive an annual examination.

3. Paragraph 3 of ANSI N45.2.6-1978, Qualification. Same clarification as 1. 15

9

INSERT Pg. 1.9-51

1. Paragraph 1.2 of ANSI N45.2.6 - 1978, Applicability. VEGP personnel who approve preoperational, startup, and test results and who direct or supervise the conduct of individual preoperational, startup, and operational tasks shall be qualified in accordance with the VEGP position to Reg. Guide 1.8 in lieu of being qualified to ANSI N45.2.6 as allowed by regulatory position C.1 of Reg. guide 1.58, Rev. 1. For Nuclear Operations, VEGP elects to apply the requirements of this guide to quality control inspection personnel. Regulatory position C.2 of Reg. Guide 1.58, Rev. 1, states that the 1975 version of SNT-TC-1A is acceptable for the qualification of personnel performing nondestructive examinations. In lieu of this, VEGP shall use the requirements of the 1980 version of SNT-TC-1A for qualifying personnel performing nondestructive inspection, examination, or testing, and in addition VEGP shall supplement these requirements by replacing the "shoulds" contained in the 1980 version with "shalls" except as follows:

   (a) The "should" in the last sentence of Section 4.3 (3) of SNT-TC-1A - 1980, which refers to a NDE level III being qualified to instruct and examine NDT personnel, is not replaced with a "shall". For the VEGP, a NDE level III individual shall be responsible for the training and examination of NDT personnel; while conducting of actual training and examination may be delegated to qualified individuals appointed by the level III individual.

   (b) The "should" in the second sentence of Section 8.4.3 of SNT-TC-1A - 1980 remains as a "should", as allowed by Section 8.4.2. VEGP elects to use a composite grade based on the simple average of the examinations; this shall be prescribed in VEGP's written practice.

For personnel performing calibration, installation checkouts, or routine surveillances the requirements of this guide will not be applied, as allowed by Section 1.2 of ANSI N45.2.6 - 1978; personnel performing these functions shall either meet the minimum education and experience recommendations of ANSI N18.1 - 1971 or will complete a qualification program which will demonstrate their ability to perform their job functions. FSAR Table 13.1.3-1 designates the minimum education and experience recommendations for plant personnel, while FSAR Section 13.2.2 describes the training programs which demonstrate the ability of plant personnel to perform their job functions.

Enclosure B

CLARIFICATION OF PREVIOUS VEGP RESPONSE TO
NRC STAFF QUESTIONS ON VOGTLE FSAR SECTION 6.4,
CONTROL ROOM HABITABILITY (OPEN ITEM 51)

1.0 BACKGROUND

In a telephone conversation between VEGP and the NRC staff on April 11, 1985, the NRC expressed concern that the assumptions used in VEGP's control room toxic gas hazard evaluation were inconsistent with the recommendations of USNRC Regulatory Guide 1.78. Specifically, the staff's technical objections to VEGP's modeling approach appeared to focus on two issues:

1. VEGP utilized a toxic gas infiltration model that took credit for the dilution of Cl_2 gas in a portion of the control building before infiltrating into the control room. The staff stated that an unpublished NRC document limits the credit that can be taken for dilution of gas in a volume that surrounds the control room to a maximum factor of two. VEGP's model provides a control building dilution factor of approximately 100.

2. VEGP's model assumes that all of the contaminated outside air entering the control building is forced in by the control building HVAC intake fans. These fans continue operating following Cl_2 gas detection and control room isolation. The staff's contention is that VEGP should have also assumed additional inleakage through the control building outer walls associated with a 1/8 inch water gauge (w. g.) differential pressure across these walls.

2.0 CLARIFICATION OF VEGP's PREVIOUS RESPONSE

VEGP's earlier written response to the NRC staff's questions regarding the control room habitability analysis outlined the various sources of conservatism in the VEGP model that we believe resulted in conservatively high predictions of the Cl_2 gas concentration inside the control room. That opinion remains unaltered. However, to more thoroughly address the staff's specific concerns, the following clarifying remarks are provided.

Issue 1: Dilution of Chlorine Gas Within the Control Building

The dilution which occurs as Cl_2 gas passes through rooms in the control building was modeled in exactly the same manner as dilution within the control room. If Cl_2 entering the control room directly from outside air becomes diluted within the free volume of the control room, then the same phenomenon will occur if the gas were introduced into some other room. We do not understand the technical basis underlying the staff's position that dilution within the control building (or any

portion thereof) should be limited to a factor of two. We surmise that the staff may be referring to the 2-fold dilution credit allowed by Standard Review Plan Section 6.4 for control rooms with physically separated outside air intakes. If this is in fact the source of the confusion, the following points should help to clarify the issue:

1. VEGP does have two physically separated control room outside air intakes. They are situated approximately 150 feet apart. Under the postulated worst case Cl_2 release scenario (catastrophic rupture of a 1-ton Cl_2 cylinder 187 meters from the closest control room outside air intake; wind blowing directly from the release point towards that intake; Pasquill G stability), one of the control room outside air intakes would be exposed to peak centerline concentrations of the Cl_2 cloud. Simultaneously, the second intake would see virtually zero ppm Cl_2. For this scenario, the intake exposed to the peak centerline concentration of Cl_2 would isolate. The other intake would not isolate because its chlorine detector would not be exposed to an activating concentration (2 ppm by volume) of the gas.

2. If this postulated worst case accident scenario were to actually occur, Cl_2 gas entering the control room through the contaminated outside air intake (prior to isolation) would undergo a two-fold dilution by the uncontaminated air from the second intake. After isolation of the first intake, the second intake would continue to draw in 1500 cfm of uncontaminated outside air. The important points to note regarding this sequence of events are as follows:

   a. VEGP's model does not take any credit for the two-fold dilution of Cl_2 gas that would occur by virtue of the dual intake configuration. Such credit would have produced a substantially smaller predicted Cl_2 concentration inside the control room at the time when operators are assumed to have donned protective breathing equipment (120 seconds following an alarm from the chlorine detector).

   b. Because one outside air intake would continue to deliver 1500 cfm of uncontaminated outside air to the control room following isolation of the contaminated outside air intake, the control room would exist at a positive pressure relative to the adjoining rooms within the control building. Consequently, the net flow of Cl_2-contaminated air after control room isolation would be outward; i.e., from the

control room to the control building, rather than the reverse. VEGP's model conservatively assumes a 1/8 inch w.g. differential pressure driving contaminated air <u>into</u> the control room from the adjoining rooms -- exactly the opposite of what would actually occur. Additionally, VEGP took no credit for dilution of the presumed inleaking air by the 1500 cfm of uncontaminated air supplied by the remaining (unisolated) control room outside air intake.

<u>Issue 2</u>: Inleakage of Contaminated Outside Air Through the Outer Walls of the Control Building.

Another of the staff's criticisms of the VEGP Cl_2 gas infiltration model is that VEGP omitted consideration of the additional inleakage of contaminated outside air through the outer walls of the control building caused by a 1/8 inch w.g. differential pressure across those walls. Keeping all other modeling assumptions fixed, adoption of this assumption would produce higher predicted control building Cl_2 concentrations versus time, and correspondingly higher control room concentrations.

While we concede that it may have been more realistic to assume a 1/8 inch w.g. differential pressure across outer walls of the control building rather across the control room outer walls, we continue to believe that our current analysis provides a conservative upper bound on Cl_2 concentrations that would actually occur within the control room for the postulated worst case onsite Cl_2 gas release. Our reasons are as follows:

1. At any instant in time, a maximum of only two outer walls of the control building (those facing the oncoming wind) could be under a positive pressure. The remaining walls would be under a negative pressure. Hence, differential pressure could not be a driving force causing entry of contaminated outside air into the control building except on a maximum of two of the control building outer walls.

2. The windspeed necessary to create a 1/8 inch w.g. differential pressure across the outer walls of the control building is in excess of 7 meters per second (~ 15 1/2 miles per hour). Windspeeds of 7 meters per second or greater occur less than 5 percent of the time at the Vogtle site. At the 0.5 meter per second windspeed producing the highest predicted 2-minute integrated Cl_2 concentration inside the control room, the differential pressure across the control building outer walls is on the order of 10^{-5} inch w.g. At this windspeed, the only significant mode of entry of contaminated outside air into the control building would be via the control building HVAC intake system.

3. Any Cl_2 gas entering the control building through outer wall inleakage would have to pass through multiple layers of rooms prior to reaching the control room. Dilution would occur within every layer, yielding a negligible addition to the peak centerline Cl_2 gas concentration assumed to be supplied directly to rooms adjoining the control room by the control building HVAC system.

4. For the postulated worst case Cl_2 release scenario, after isolation dampers in the contaminated outside air intake vent for the control room have closed, the control room exists at a positive pressure relative to adjoining rooms within the control building (see the earlier discussion). Thus, Cl_2 infiltration into the control room would effectively cease upon isolation. The Cl_2 concentration inside the control room at the time of isolation would be approximately 1×10^{-3} ppm. The concentration would then begin to decline.

3.0 SUMMARY OF VEGP's POSITION

VEGP takes exception to the NRC position that the results of our current analysis are nonconservative and non-conforming to Regulatory Guide 1.78. We do not understand the technical basis for the NRC staff's apparent position that no more than a two-fold dilution credit can be taken for initial dilution of Cl_2 within the control building. In addition, we do not believe that our analysis results are nonconservative because we did not assume the existence of a 1/8 inch w.g. differential pressure across all outer walls of the control building. At the 0.5 meter per second windspeed which maximizes the predicted Cl_2 concentration within the control room, the contribution of control building outer wall inleakage is clearly negligible.

At higher windspeeds, where a measurable positive differential pressure could exist across walls of the control building (but on no more than two outer walls at any one time), the control room would still be at a positive pressure relative to the adjoining rooms, resulting in essentially zero infiltration. Considering these factors and the numerous conservatisms in our analysis (e.g., no credit for the dual outside air intake configuration as allowed by Standard Review Plan Section 6.4), we believe that we have adequately demonstrated conformance with the USNRC Regulatory Guides 1.78 and 1.95 by providing an acceptable alternative.

Enclosure C

Question 410.55

In FSAR subsection 9.4.9, you state that the piping penetration ventilation system will maintain the concrete surrounding the piping restraints for the main steam and feedwater systems below 200°F. Verify that the ambient air temperatures in these areas, including the valve rooms, will be maintained at a low enough temperature to allow personnel to inspect equipment during normal plant operation. If there is another heaving, ventilation, and air conditioning (HVAC) system that performs this function, identify the system.

Response

During normal plant operation, the maximum temperature has been calculated to be 115°F in the main steam/feedwater isolation valve areas. This is adequate to allow personnel entry for inspection during normal plant operation. No other HVAC system performs this function.

ADD INSERT

INSERT TO Q410.55

The piping penetration ventilation system is not safety-related because loss of this system would not limit operation of the plant or constitute a safety hazard. The ventilation system is required to keep the long term concrete temperature below 200°F. This is not a safety-related function.

An analysis has been performed for loss of the ventilation system and for continued normal operation of the plant. The analysis indicates that the main steam isolation valve room ambient temperature will not exceed the equipment environmental qualification temperature of 126°F. The room would be cooled by natural circulation instead of forced circulation from the fans. Low flow alarms have been provided in the control room to alert the operator to the loss of forced ventilation.

The temperature of the concrete surrounding the main steam line and main feedwater line restraints would increase due to the heat transmitted from the main steam and main feedwater pipe restraints. ACI Standard 349-76 permits concrete temperatures up to 350°F for any short-term period. A finite element heat transfer analysis was performed to design the forced ventilation system. This calculation also envelopes the natural convection condition with no forced ventilation, indicating that the concrete temperature would not exceed 320°F.

ENCLOSURE D

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of piping and instrument diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

Response

A list of pressure isolation valves and their function is provided in table 210.48-1. The piping and instrumentation diagrams which show the layout of these valves are shown in figures 210.48-1 through 210.48-5. The valves listed in table 210.48-1 correspond to those valves discussed in paragraphs 5.2.5.2.1.A through 5.2.5.2.1.E of the FSAR with the exception of check valves 1204-026 through 1204-029 and 1204-013. As discussed in paragraph 5.2.5.2.1.E of the FSAR, leakage past valves 1204-026 through 1204-029 and 1204-013 will be isolated by motor-operated valves HV-8801 A/B and manual valve 1204-007. Leakage past valves HV-8801 A/B and manual valve 1204-007 is not possible since the valve inlets are maintained at a pressure greater than reactor coolant system pressure by the operating charging pump.

The valves listed in table 210.48-1 will be included in the VEGP Technical Specifications with Limiting Conditions for Operation which will require system isolation or shutdown in the event that leakage limits are not met. A draft version of the VEGP Technical Specifications is scheduled for submittal to the Nuclear Regulatory Commission (NRC) at least 12 months prior to fuel load.

Periodic leak testing of each valve listed in table 210.48-1 will be performed after each refueling outage; prior to returning the valve to service following maintenance, repair, or replacement work on the valve; and for those systems rated at less than 50 percent of reactor coolant system design pressure following valve actuation due to automatic or manual action or flow through the valve.

~~Concerning the maximum allowable leak rate for each pressure isolation valve, VEGP proposes to use an allowable leak rate of 1/2 gal/min for each inch of valve size with a maximum upper limit of 5 gal/min. The justification for this leakage limit follows.~~

INSERT "A"

Insert "A"

Concerning the maximum allowable leak rate for pressure isolation valves, VEGP's leak rate limiting conditions for operation will be equal to or less than one gpm for each pressure isolation valve. Pursuant to favorable resolution by the Committee to Review Generic Response's (CRGR) the maximum allowable leak rate for pressure isolation valves will be 1/2 gpm per inch of nominal valve size with a maximum upper limit of 5 gpm.

A. In a study which was sponsored by the NRC staff (EGG Report EGG-NTAP-6175, February 1983, In Service Leak Testing of Primary Pressure Isolation Valves, R. A. Livingston), it was concluded that allowable leak rates based on valve size were superior to a single allowable value because a single allowable value imposes an unjustified penalty on larger valves without providing information on potential valve degradation. Also, the larger valves must be repaired in place which subjects plant personnel to radiation exposure in order to meet an overly conservative standard.

In addition, an indexing criterion to account for gross increases in leakage from one test to a later test, as found in the ASME code, paragraph IWV-3427 (b) is a direct indicator of potential valve degradation. Since such an indexing criterion will be used by Georgia Power Company, this will provide at least as good, if not better, an indication of valve deterioration as the 1-gal/min criterion.

B. The original 1-gal/min. criterion was based on a very conservative estimate of the pressure relief system capacity for a plant. The 1-gal/min criteria is not an indicator of imminent accelerated deterioration or potential valve failure.

ENCLOSURE E

Curriculum Outline	Approximate Duration
Electrical, pressure vessel, and piping codes and standards (including nondestructive testing review)	3 days

G. Incumbents and New Employees

Personnel with experience that exceeds NRC commitments may fill a position in the electrical career path provided that the maintenance superintendent certifies that the employee's experience qualifications exceed the position requirements. The training department may also accept prior training or experience to fill specific course requirements.

13.2.2.1.5 Shift Technical Advisor Training Program

A. Education Requirements

Shift technical advisors will have a bachelor's degree in a scientific or engineering discipline.

B. Training Program

The candidate holds or has held an NRC senior reactor operators license for that type of reactor, or the candidate completes a Georgia Power Company (GPC) shift technical advisor training program described in table 13.2.2-1. In either case, the shift technical advisor shall receive specific training in the response and analysis of the plant for transients and accidents.

C. Experience Requirements

The candidate will have 1 year of power plant experience and will have performed reactor operator or senior reactor operator duties for that type of reactor, or the candidate will receive 1 month of on-the-job training as an extra shift technical advisor.

D. Requalification Training for Shift Technical Advisors

Shift technical advisors will attend the same requalification program as NRC-licensed operators. ~~Persons not actively performing the shift technical advisor functions for a period of 4 months or longer shall, prior to assuming responsibilities of the position, as a minimum receive training to ensure they~~

INSERT "A"

Amend. 4 2/84

Insert "A"

Shift Technical Advisors which do not stand shift for a period of 30 days or longer will receive on-the-job instruction from the duty STA concerning plant changes which have occured during their absences.

~~are cognizant of facility/procedure changes that occurred during their absences.~~

Persons not performing the shift technical advisor function for a period of 6 months or longer shall, prior to assuming the responsibilities of the position, undergo an individual requalification program.

13.2.2.1.6 Nonlicensed Operator Training Program

A. Initial Training

After the start of fuel load, all personnel assigned to perform independent plant equipment manipulations will either complete this initial training program, be qualified to the shift technical advisor level or certified to the senior reactor operator level, or have experience which is equivalent to the following program.

Curriculum Outline	Approximate Duration
General employee badge and health physics training	6-10 h[a]
Industrial safety	4 h
New employee fire training	2 h
Nuclear power plant fundamentals	1 week
Power plant components	1 week
VEGP systems	4 weeks
On-the-job training	1 week

B. Continuing Training

After completing initial qualifications, the nonlicensed operator will complete qualification on the plant systems on which he was not initially qualified. Normally, the nonlicensed operator will qualify on all systems outside the control room and containment during the individual's first 3 years in the plant operations department.

C. Annual Requalification Training or Exemption Testing

Nonlicensed operators will complete annual requalification training or exemption testing to make

a. Duration depends on whether the individual attends additional radiation worker training as described in paragraph 13.2.2.1.8.A.

~~Amend. 3 1/84~~
Amend. 4 2/84
Amend. 16 4/85

Enclosure F

Question 440.131

Figures 15.6.3-1 and 15.6.3-4 show a differential pressure of about 1000 psi between the primary and faulted steam generator at 30 min. Figure 15.6.3-11 shows an increasing water volume due to the break flowrate as shown in figure 15.6.3-9 at 30 min. Paragraph 15.6.3.2.1 states, however, that leakage flow through the ruptured tube is assumed to be terminated within 30 min of the initiation of the event. Unless these parameters show discontinuous behavior at 30 min, it would appear that the assumption and the figures are in conflict. Please resolve this.

If leakage flow is terminated at 30 min, how is it accomplished? Any equipment used should be listed in table 15.0.9-1 and qualified.

If the leakage flow is not terminated at 30 min and since the flow through the steam generator safety valve has approached a non-zero asymptote at this time, it would appear that additional radioactive material will be released to the atmosphere. In this event, you will need to reanalyze the radiological consequences.

Response

A subgroup of the Westinghouse Owners Group (WOG) has been formed to generically address several of the steam generator tube rupture licensing issues which have been raised for NTOL plants. Georgia Power Company is represented in the subgroup by Southern Company Services. The WOG submitted WCAP 10698 in December 1984 in response to steam generator tube rupture. Two appendixes are scheduled for this WCAP. The first is a response to radiological consequences of a steam generator tube rupture (scheduled for April 1985) and the second is a response to SC overfill (scheduled for July 1985). Following completion of the generic program, ~~resolution of these concerns will be addressed.~~ GPC will implement the recomendations of the generic study.