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SOUTHERN COMPANY
Energy to Serve Your World℠

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LCV-1149-D

Docket Nos. 50-424
50-425

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

Ladies and Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
REQUEST TO REVISE TECHNICAL SPECIFICATIONS
CONTAINMENT EQUIPMENT HATCH

In accordance with the requirements of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to revise the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.9.4, Containment Penetrations. The proposed changes would allow the equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment. Appropriate Bases changes are included to reflect the proposed changes. The basis for the proposed changes is provided in Enclosure 1. Pursuant to 10 CFR 50.92, an evaluation that demonstrates that the proposed changes do not involve a significant hazard consideration is provided in Enclosure 2. The proposed changes are marked on the affected TS and Bases pages and provided in Enclosure 3. In addition, clean typed TS and Bases pages are provided in Enclosure 4.

By letter dated June 26, 1998, (LCV-1149) and as supplemented by letters dated September 18, 1998, (LCV-1149-A) and November 30, 1998, (LCV-1149-B), Southern Nuclear Operating Company (SNC) proposed to revise the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.3.6, Containment Ventilation Isolation Instrumentation, and LCO 3.9.4, Containment Penetrations. The proposed changes affected these requirements during core alterations and movement of irradiated fuel assemblies inside containment. In summary, the proposed changes:

- Deleted requirements for automatic and system level manual initiation of containment ventilation isolation;
- Allowed the emergency air lock to be open; and,
- Would have allowed the containment equipment hatch to be open.

A001

The basis for the proposed changes was that the site boundary radiological consequences of a fuel handling accident inside containment would be less than 25% of 10 CFR 100, and the control room radiological consequences would be within the acceptance criteria given in Standard Review Plan (SRP) section 15.7.4 and General Design Criteria (GDC) 19. Furthermore, SNC notes that the offsite dose consequences of a fuel handling accident in the fuel handling building envelope the offsite dose consequences of a fuel handling accident inside containment with the equipment hatch open. Refer to FSAR section 15.7.4. (Control room doses are bounded by the fuel handling accident inside containment. The limiting release path to the control room air intake is via the personnel air lock.) It should be noted that the dose analysis of record for a fuel handling accident in the fuel handling building takes no credit for filtration by the fuel handling post accident exhaust system.

By letter dated January 29, 1999, the NRC approved the proposed changes for the containment ventilation isolation capability and the emergency air lock. The proposed changes for the containment equipment hatch were denied. The NRC staff safety evaluation report notes that the inclusion of an open equipment hatch runs contrary to the principle of defense in depth. "In particular, the potential for relatively quick core uncovery (e.g., 30 minutes) due to loss of decay heat removal (DHR) relating to cooling may make the timely closure of the equipment hatch unfeasible. Concerns of this type have been noted in Generic Letter 88-17, "Loss of Decay Heat Removal," dated October 17, 1988."

SNC notes that, with respect to loss of DHR, the capability to close the equipment hatch was addressed in our responses to Generic Letter 88-17. Furthermore, Generic Letter 88-17 was concerned with operation during reduced inventory conditions, i.e., when reactor vessel water level is below the reactor vessel flange. Core alterations and movement of irradiated fuel cannot take place unless refueling cavity water level is maintained $\geq$ 23 feet above the reactor vessel flange. Therefore, the consequences of a loss of DHR under these conditions are much less severe than during a condition of reduced inventory, and the time allowed for closing the equipment hatch would be much longer. Finally, SNC notes that there is no requirement to postulate a loss of DHR in conjunction with a fuel handling accident. Hence, the environmental conditions inside containment associated with a loss of DHR (conditions such as steaming that could hinder personnel in their efforts to close the hatch) would not apply to the fuel handling accident.

As part of the proposed change to allow the equipment hatch to be open during core alterations and movement of irradiated fuel inside containment, SNC committed to be able to close the equipment hatch within one hour in the event of a fuel handling accident inside containment. The dose analysis supporting the proposed change concludes that site boundary doses remain less than 25% of 10 CFR 100 and control room doses meet SRP section 15.7.4 and GDC 19 without credit for closure of the equipment hatch. Therefore, the commitment to close the hatch within one hour maintains defense in depth.

In addition, SNC notes that by letter dated March 11, 1999, the NRC granted Amendment 102 to Facility Operating License No. NPF-58 – Perry Nuclear Power Plant, Unit 1. This amendment revised the Perry TS for handling irradiated fuel in the primary containment and the fuel handling building and selected specifications associated with performing core alterations. While the designs of the primary containments for a boiling water reactor and a pressurized water reactor are different, the net effect of the approved changes to the Perry TS is the same as that proposed by SNC for the VEGP equipment hatch. The subject Perry

amendment relaxed primary containment, secondary containment, and fuel handling building integrity requirements and requirements for those engineered safety features originally relied upon to mitigate a fuel handling accident.

In its review of the Perry submittal, the NRC staff focused on the four following areas:

- Dose calculations – Control room and offsite dose consequences must be within acceptable regulatory limits without taking credit for the integrity of primary containment (and in the case of Perry, the secondary containment, fuel handling building, and fuel handling accident mitigating systems). In the case of VEGP, control room and offsite dose consequences must be within acceptable regulatory limits without taking credit for containment integrity, including no credit for containment ventilation isolation. At VEGP, the control room dose calculations do take credit for the control room emergency filtration system (CREFS) which will continue to be required to be operable by the VEGP TS.

  In addition, Perry introduced the concept of "recently irradiated fuel," which is an extended period of decay beyond which the dose consequences of a fuel handling accident will be less than 25% of 10 CFR 100 limits. The length of this extended period is determined by a plant-specific dose calculation. At VEGP, movement of irradiated fuel is limited by the Technical Requirements Manual, section 13.9.1, which requires that the reactor be subcritical for $\geq$ 100 hours prior to movement of irradiated fuel in the reactor vessel. The dose analysis for VEGP, which supports the proposal to allow the equipment hatch to be open during movement of irradiated fuel assemblies, is based on a decay period of 100 hours. Further details regarding the dose analysis are provided in Enclosure 1.

- Administrative controls – Shutdown safety controls must address 1) procedures to assess the impact of removing systems from service during shutdown conditions, 2) the ability to implement prompt methods to close the primary containment (and in the case of Perry, the fuel handling building) in the event of a fuel handling accident, and 3) controls to avoid unmonitored releases. Due to the differences in designs between the boiling water reactors and pressurized water reactors, closing the fuel handling building is not an issue at VEGP. As stated above, the accident of concern with respect to allowing the equipment hatch to be open is a fuel handling accident inside containment. The fuel handling building is a separate building altogether at VEGP, and it has its own analysis for a fuel handling accident, which bounds the analysis for inside containment. Shutdown safety controls, the ability to promptly close containment in the event of a fuel handling accident, and controls to avoid unmonitored releases are discussed in Enclosure 1.

- Risk significance – The licensee's risk-related discussion needs to support the proposed TS changes. See Enclosure 1 for discussion.

- Shutdown operations – The licensee's proposed amendment should be consistent with the Commission's December 11, 1997, instructions to the staff. See Enclosure 1 for discussion.

The proposed changes will permit the optimization of refueling outages to achieve an overall risk reduction while also reducing outage time and cost. Many of the containment load-in / load-out activities would be performed while the reactor is defueled or the reactor vessel is fueled, open and covered by 23 feet of water (risk of a severe core damage accident is very low at this time). The containment equipment hatch would be opened and closed one time, instead of three times as required based on the existing TS. This would permit containment load-in / load-out activities to be performed in one continuous duration. This, in turn, would reduce the need to maintain the containment equipment hatch open during the fueled midloop condition when containment closure is not required but the risk of a severe core accident is increased due to the reduced inventory.

SNC requests approval of the proposed changes by June 30, 2000, so that the proposed changes can be utilized for the Unit 1 refueling outage scheduled for Fall of the year 2000.

Mr. J. B. Beasley, Jr. states that he is a Vice President of Southern Nuclear Operating Company and is authorized to execute this oath on behalf of Southern Nuclear Operating Company and that, to the best of his knowledge and belief, the facts set forth in this letter are true.

SOUTHERN NUCLEAR OPERATING COMPANY

By: _____
J. B. Beasley, Jr.

Sworn to and subscribed before me this 6th *day of* March, *2000.*

_____
Notary Public

My commission expires: 11/10/02

JBB/NJS

Enclosure 1 – Basis for Proposed Changes
Enclosure 2 – Significant Hazards Consideration Evaluation
Enclosure 3 – Marked-up TS and Bases Pages
Enclosure 4 – Clean Typed TS and Bases Pages

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Enclosure 1

Vogtle Electric Generating Plant
Request to Revise Technical Specifications
Containment Equipment Hatch

Basis for Proposed Changes

Proposed Changes

LCO 3.9.4 would be revised to allow the equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment, provided that it is capable of being closed. A new surveillance requirement would be added to verify the capability to install the equipment hatch, if the hatch is open, at a frequency of seven days. Appropriate Bases changes are included to reflect the proposed changes.

Basis

The following basis addresses the four areas of concern identified by the NRC staff in their approval of Amendment No. 102 to the Perry TS. These four areas are dose calculations, administrative controls, risk significance, and shutdown operations.

Dose Calculations - By letter dated November 30, 1995, the NRC issued Amendments 92 (Unit 1) and 70 (Unit 2) to the Vogtle Electric Generating Plant (VEGP) Technical Specifications (TS) to allow both personnel air lock doors to be open during core alterations and movement of irradiated fuel assemblies inside containment. Similarly, by letter dated January 29, 1999, the NRC issued Amendments 105 (Unit 1) and 83 (Unit 2) to the VEGP TS to allow both emergency air lock doors to be open as well as deleting requirements for automatic and system level manual initiation of containment ventilation isolation. The dose analyses supporting all of these amendments were performed in support of the original request to allow the personnel air lock doors to be open during core alterations or movement of irradiated fuel assemblies inside containment. Please see our letters dated March 17, 1995, (LCV-0527) as supplemented by letters dated July 6, 1995, (LCV-0527-B) and November 30, 1995, (LCV-0527-C).

Given a fuel handling accident inside containment, the resulting offsite dose consequences with both personnel air lock doors open were calculated to be 65.6 rem thyroid and 0.28 rem whole body. These results were less than 25% of the 10 CFR 100 limits, and they are bounded by the current fuel handling accident analysis for the spent fuel pool. A fuel handling accident in the spent fuel pool results in offsite doses of 73 rem to the thyroid and 0.29 rem to the whole body, and no credit is taken for the fuel handling building emergency filtration system charcoal filters.

The control room dose associated with a fuel handling accident inside containment with the personnel air lock doors open was found to remain below 30 rem thyroid if one of the four emergency control room filtration units is operating within seven minutes of the accident. These results are within the guidelines of General Design Criteria (GDC) 19 of Appendix A to 10 CFR 50 as defined by Standard Review Plan (SRP) Section 6.4. Automatic actuation of the control room emergency filtration system on intake radiogas will continue to be required with either or both units in Modes 1, 2, 3, or 4 and/or during movement of irradiated fuel and core alterations. In addition, LCO 3.3.6, Table 3.3.6-1 would continue to require the containment radiation monitors (gaseous, particulate, iodine, and area low range) to be operable to the extent that they would provide alarms in the control room in the event of a fuel handling accident inside containment.

The control room dose for the fuel handling accident inside containment would bound that for the accident in the fuel handling building because of the shorter release path via the personnel air lock doors.

Basis for Proposed Changes

Similarly, the control room doses via the personnel air lock would bound that from the equipment hatch because the equipment hatch is on the opposite side of containment. The offsite dose calculation is bounded by the analysis for the fuel handling building which is independent of the relative locations of the personnel air lock and the equipment hatch.

With respect to Amendments 92 and 70, the NRC staff performed an independent analysis to determine conformance with the requirements of 10 CFR 100 and GDC 19. The staff's analysis used the accident source term given in Regulatory Guide 1.4, the assumptions contained in Regulatory Guide 1.25, and the review procedures specified in SRP Sections 15.7.4 and 6.4. The staff assumed an instantaneous puff release of noble gases and radioiodines from the gap and plenum of the broken fuel rods. These gas bubbles will then pass through at least 23 feet of water covering the fuel prior to reaching the containment atmosphere. All airborne activity reaching the containment atmosphere is assumed to exhaust to the environment within two hours. The gap activity was assumed to have decayed for a period of 100 hours. The offsite doses calculated by the NRC staff were 37.0 rem to the thyroid and 0.18 rem whole body. The control room operator doses calculated by the NRC staff were 1.38 rem to the thyroid and 0.29 rem to the whole body. The NRC staff's independent analysis confirmed that the consequences of a fuel handling accident inside containment with the personnel air lock doors open are within the acceptance criteria given in SRP Section 15.7.4 and GDC 19.

Administrative Controls - Shutdown safety controls must address 1) procedures to assess the impact of removing systems from service during shutdown conditions, 2) the ability to implement prompt methods to close the primary containment (and in the case of Perry, the fuel handling building) in the event of a fuel handling accident, and 3) controls to avoid unmonitored releases. In the draft NUMARC 93-01 guideline, Section 11, under the subheading of "Containment – Primary (PWR)/Secondary (BWR), the following guidance is provided.

> "..... for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:
>
> - During fuel handling/core alterations, ventilation system and radiation monitor availability should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.
>
> - A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

Enclosure 1

Vogtle Electric Generating Plant
Request to Revise Technical Specifications
Containment Equipment Hatch

Basis for Proposed Changes

The proposed change does not affect the operability requirements for any ventilation system or radiation monitors, nor does it affect their availability. The control room emergency filtration (CREFS) will be required to be operable by the TS as well as containment radiation monitors. The only affected containment penetration that provides direct access to the outside atmosphere is the equipment hatch. Existing TS requirements on other penetrations that provide direct access are not affected.

Containment ventilation at VEGP is accomplished via a preaccess purge system and a minipurge system. These systems are not credited in any of the dose analyses, so there are no TS operability requirements associated with them. The preaccess, or hereafter referred to as normal purge, is only used for high flowrate purge during refueling and is required by the TS to be sealed closed during normal power operation. The minipurge system is used for low flowrate purge during power operation, but there are no operational or TS constraints that would prevent its use during Modes 5 and 6 as well. The normal purge system consists of a supply air handling unit and an exhaust charcoal filtration unit. The minipurge system utilizes the supply air handling unit and exhaust charcoal filtration unit of the normal purge system, but uses separate low flowrate fans. Both systems share purge supply and exhaust containment penetrations. Each penetration is equipped with two valves in parallel inside containment and two valves in parallel outside containment. The flowpath arrangement provides each penetration with both a 24-inch flowpath (normal purge) that can be used in Modes 5 and 6 in parallel with a 14-inch flowpath (minipurge) that can be used for containment purge during Modes 1 through 4 as well as Modes 5 and 6. Each valve is equipped with its own handswitch located in the control room on section 2 of the QHVC panels, and the handswitches are grouped together. FSAR Figure 18.1-1 shows the location of the QHVC panels in relation to the main control boards. The panels are easily accessible for an operator at the main control boards.

The purge supply air handling unit (used by both the normal and minipurge systems) consists of two fans in parallel. One fan is rated at 15,000 ft^3/min and is used for normal purge during refueling. The other fan is rated at 5,000 ft^3/min and is used for minipurge. The exhaust unit consists of a filtration unit with a moisture separator, heating coil, HEPA filter, charcoal adsorber, and a second HEPA filter. Two fans in parallel are located at the unit outlet, one for normal purge and the other for minipurge. Each fan has the same capacity as the corresponding supply unit fans. The containment purge high flowrate system is designed to maintain the airborne radioactivity below the level required for personnel occupancy during refueling, and the minipurge system is designed to maintain airborne radioactivity below the required level for personnel occupancy during reactor power operation. The exhaust from these systems is ducted to the plant vent that is located at the top of the containment building. The HEPA filter elements and charcoal adsorber sections are tested periodically in accordance with Regulatory Guide 1.140. The handswitches for starting and stopping the fan units (normal purge and minipurge) are located adjacent to the handswitches for the purge valves. Therefore, in the event of a fuel handling accident inside containment with the equipment hatch open, the containment purge can be easily controlled from the control room.

There is also a preaccess filter system which consists of two 100 percent capacity filtering units inside containment. The preaccess filter system is a recirculating system designed to clean the internal air without providing new air makeup. The operation of this system is initiated from the control room by starting the fan.

Enclosure 1

Vogtle Electric Generating Plant
Request to Revise Technical Specifications
Containment Equipment Hatch

Basis for Proposed Changes

Containment radiation is monitored via the purge exhaust radiation detectors (RE-2565A, B, and C) which monitor particulate, iodine and noble gases, respectively. In addition, two individual channels of containment area low range gamma monitors (RE-0002 and RE-0003) are provided. The TS (LCO 3.6.6) require at least two channels of containment radiation instrumentation to be operable. These two channels may consist of the two area low range channels or a combination of one area low range channel and the purge exhaust radiation monitor. The purge exhaust radiation monitor is considered operable if the particulate (RE-2565A) and iodine (RE-2565B) are operable, or the noble gas monitor (RE-2565C) is operable. In the event of a fuel handling accident inside containment, the control room alarm function of the required containment radiation monitors will be in service, and the radiation monitors will help to provide indication of the magnitude of the release, thereby minimizing the potential for an unmonitored release.

During core alterations, the Technical Requirements Manual (TRM), TR 13.9.2, requires that direct communications be maintained between the control room and personnel at the refueling station. Therefore, if a fuel handling accident were to occur inside containment, the control room would be immediately informed, and action would be promptly initiated to mitigate the consequences. Containment work activities that could preclude closure of the equipment hatch or the operation of the containment purge exhaust system would be administratively controlled while core alterations were in progress. During core offload and reload, the containment purge exhaust system would be maintained in service providing an inward flow of air into containment. In the event of a fuel handling accident inside containment, the containment purge exhaust system would be momentarily secured to support placement and initial bolting of the containment equipment hatch. Once initial bolting was complete, the containment purge exhaust and supply systems and containment preaccess filtration system would be operated as appropriate to minimize radiological releases while providing containment habitability to respond to the fuel handling accident.

In addition, VEGP is already committed to having two designated individuals for closing the personnel and emergency air locks if open during core alterations or movement of irradiated fuel assemblies inside containment. The existing VEGP TS specify requirements for maintaining the air locks in an isolable condition (LCO 3.9.4), and the Bases provide additional information. If open, the equipment hatch will also be maintained in an isolable condition, and the TS and Bases will contain requirements similar to those for the air locks. Specifically, the equipment hatch will be considered to be isolable when 1) the necessary equipment required to close the hatch is available, 2) at least 23 feet of water is maintained over the top of the reactor vessel flange, and 3) a designated trained hatch closure crew is available. The equipment hatch will be capable of being cleared of obstructions so that closure can be achieved as soon as possible, and the necessary hardware, tools and equipment will be available for moving the hatch from its storage location and installing it in the opening. Presently, VEGP is committed to closing any open air lock(s) and isolating the containment purge penetrations within 15 minutes of a fuel handling accident inside containment. This was to further minimize releases from containment, but was not required to maintain the offsite dose to less than 25% of 10 CFR 100 limits. Under the proposed change, VEGP will maintain the commitment to close the air lock doors within 15 minutes while pursuing timely closure of the equipment hatch. Once the hatch is closed, the hatch closure crew will be able to exit containment by opening one air lock door at a time, further minimizing the extent of an unmonitored, untreated release. Best estimate thyroid doses to the containment closure crew were calculated to be 22.3 rem based on their being inside containment for an hour while installing the hatch. The "best estimate" thyroid doses to the

containment closure crew were evaluated using more realistic assumptions in the same fashion as the thyroid dose calculation that was performed for personnel designated for closing the air lock in support of Amendments 92 and 70 that allowed the personnel air lock to be open during fuel movement.

Risk Significance – Based on the results of conservative dose calculations provided in this submittal, the risk to the health and safety of the public as a result of a fuel handling accident inside containment with the equipment hatch open is minimal. Actual fuel handling accidents which have occurred in the past have resulted in minimal or no releases, which shows that the assumptions and methodology utilized in the radiological dose calculations are very conservative. Radioactive decay is a natural phenomenon. It has a reliability of 100 percent in reducing the radiological release from fuel bundles. In addition, the water level that covers the fuel bundles is another natural method that provides an adequate barrier to a significant radiological release. The requirement for at least 100 hours of decay prior to fuel movement will be maintained in the TRM, and the requirement for water level will be maintained in the TS. In addition, the requirements for isolable air locks, an isolable equipment hatch, and containment radiation monitors will be maintained in the TS. The containment purge system will be available in accordance with the aforementioned NUMARC 93-01 guidelines to further reduce a radiological release. Therefore, the risk to the health and safety of the public as a result of allowing the equipment hatch to be open during fuel movement is minimal.

Shutdown Operations – In the interim period until the revision to NUMARC 93-01 is endorsed as a formal industry position, the above referenced NUMARC 93-01 words will be utilized at VEGP with respect to core alterations or movement of irradiated fuel assemblies inside containment with the equipment hatch open. Operators will be provided with the necessary procedural guidance and training to implement the aforementioned administrative controls.

Conclusions

Fuel handling accidents are not sufficiently risk-significant to warrant the restrictive containment closure requirements that presently exist in the TS.

Adequate defense in depth is maintained by the requirements for water level and radioactive decay.

Very conservative dose calculations show that site boundary doses remain less than 25% of 10 CFR 100, and control room radiological consequences are within the acceptance criteria given in SRP section 15.7.4 and GDC 19 without requirements for containment closure.

Administrative controls over shutdown safety will be in effect to control monitoring and filtration in order to minimize the potential for unmonitored, untreated releases resulting from a fuel handling accident.

Enclosure 2

Vogtle Electric Generating Plant
Request to Revise Technical Specifications
Containment Equipment Hatch

Significant Hazards Consideration Evaluation

The proposed changes have been evaluated against the criteria of 10 CFR 50.92 as follows:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The proposed changes will allow the equipment hatch to be open during core alterations and movement of irradiated fuel assemblies inside containment. The existing VEGP TS allow the air lock doors to be open during core alterations and movement of irradiated fuel assemblies inside containment, and the dose analyses for a fuel handling accident inside containment remain bounding for the case of the open air lock doors. The proposed changes will not alter the manner in which fuel is handled or core alterations are performed. Therefore the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?

   No. The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   No. The previously determined radiological dose consequences for a fuel handling accident inside containment with the air lock doors open remain bounding for the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR 100 and they meet the acceptance criteria of SRP section 15.7.4 and GDC 19. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Enclosure 3

Vogtle Electric Generating Plant
Request to Revise Technical Specifications
Containment Equipment Hatch

Marked-Up TS and Bases Pages

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	----------NOTE---------- Only required for unisolated penetrations ---------- Verify at least two containment ventilation valves in each open containment ventilation penetration providing direct access from the containment atmosphere to the outside atmosphere are capable of being closed from thecontrol room.	18 months
SR 3.9.4.3	----------NOTE---------- Only required for an open equipment hatch. ---------- Verify the capability to install the equipment hatch.	7 days

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

If closed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. Alternatively, the equipment hatch can be open provided it can be installed with a minimum of four bolts holding it in place.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is

(continued)

BASES

BACKGROUND (continued)

In MODE 6, the 24 inch main or shutdown purge and exhaust valves are used to exchange large volumes of containment air to support refueling operations or other maintenance activities. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment any open 24 inch valves are capable of being closed (LCO 3.3.6). The 14 inch mini-purge and exhaust valves, though typically not opened during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, if opened are also capable of being closed (LCO 3.3.6).

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment (Ref. 1).

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly onto another irradiated fuel assembly.

To support the plant configuration of both air lock doors open (personnel and/or emergency air locks), it was assumed in FSAR calculations for dose analysis that the designated individual for closure of the air lock would have the air lock closed within 15 minutes of the fuel handling accident. The 15 minute duration was chosen as the limit for the response capability for the person who is designated for closing the air lock door. The NRC

and to further minimize an unmonitored, untreated release,

will

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptance of this specification was based on doses for a 2 hour release as well as a licensee commitment for a person designated to close the door quickly.

The

~~Also, the~~ requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The radiological consequences of a fuel handling accident in containment have been evaluated assuming that the containment is open to the outside atmosphere. All airborne activity reaching the containment atmosphere is assumed to be exhausted to the environment within 2 hours of the accident. The calculated offsite and control room operator doses are within the acceptance criteria of Standard Review Plan 15.7.4 and GDC 19. Therefore, although the containment penetrations do not satisfy any of the NRC Policy Statement criteria, LCO 3.9.4 provides containment closure capability to minimize potential offsite doses.

LCO

The LCO requires the equipment hatch, the air locks, and any penetration providing direct access to the outside atmosphere to be closed or capable of being closed.

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires the equipment hatch and any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed. Personnel air lock closure capability is provided by the availability of at least one door and a designated individual to close it. Emergency air lock closure capability is provided by the availability of at least one door and a designated individual to close it. For the OPERABLE containment ventilation penetrations, this LCO ensures that each penetration is isolable by the Containment Ventilation Isolation valves. The OPERABILITY requirements for LCO 3.3.6, Containment Ventilation Isolation Instrumentation ensure that radiation monitor inputs to the control room alarm exist so that operators can take timely

Equipment hatch closure capability is provided by a designated trained hatch closure crew and the necessary equipment.

(continued)

BASES

LCO
(continued)

action to close containment penetrations to minimize potential offsite doses. The LCO requirements for penetration closure may also be met by the automatic isolation capability of the CVI system.

Item b of this LCO includes requirements for both the emergency air lock and the personnel air lock. The personnel and emergency air locks are required by Item b of this LCO to be isolable by at least one air lock door in each air lock. Both containment personnel and emergency air lock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided at least one air lock door is isolable in each air lock. An air lock is isolable when the following criteria are satisfied:

1. one air lock door is OPERABLE,

2. at least 23 feet of water shall be maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated individual is available to close the door.

OPERABILITY of a containment air lock door requires that the door seal protectors are easily removed, that no cables or hoses are being run through the air lock, and that the air lock door is capable of being quickly closed.

The equipment hatch is considered isolable when the following criteria are satisfied:

1. the necessary equipment required to close the hatch is available.

2. at least 23 feet of water is maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated trained hatch closure crew is available.

Similar to the air locks, the equipment hatch opening must be capable of being cleared of any obstruction so that closure can be achieved as soon as possible.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve in each open containment ventilation penetration actuates to its isolation position. The 18 month Frequency maintains consistency with other similar testing requirements. Also, SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances Performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
2. FSAR, Subsection 15.7.4.
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

SR 3.9.4.3

The equipment hatch is provided with a set of hardware, tools, and equipment for moving the hatch from its storage location and installing it in the opening. The required set of hardware, tools, and equipment shall be inspected to ensure that they can perform the required functions.

The 7 day frequency is adequate considering that the hardware, tools, and equipment are dedicated to the equipment hatch and not used for any other functions.

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in its opening, the availability of the means to install the hatch is not required.

Enclosure 4

Vogtle Electric Generating Plant
Request to Revise Technical Specifications
Containment Equipment Hatch

Clean Typed TS and Bases Pages

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	----------NOTE---------- Only required for unisolated penetrations. ---------- Verify at least two containment ventilation valves in each open containment ventilation penetration providing direct access from the containment atmosphere to the outside atmosphere are capable of being closed from the control room.	18 months
SR 3.9.4.3	----------NOTE---------- Only required for an open equipment hatch. ---------- Verify the capability to install the equipment hatch.	7 days

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. Alternatively, the equipment hatch can be open provided it can be installed with a minimum of four bolts holding it in place.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is

(continued)

BASES

BACKGROUND (continued)

In MODE 6, the 24 inch main or shutdown purge and exhaust valves are used to exchange large volumes of containment air to support refueling operations or other maintenance activities. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment any open 24 inch valves are capable of being closed (LCO 3.3.6). The 14 inch mini-purge and exhaust valves, though typically not opened during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, if opened are also capable of being closed (LCO 3.3.6).

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment (Ref. 1).

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly onto another irradiated fuel assembly.

To support the plant configuration of both air lock doors open (personnel and/or emergency air locks), and to further minimize an unmonitored, untreated release, the designated individual for closure of the air lock will have the air lock closed within 15 minutes of the fuel handling accident. The 15 minute duration was chosen as the limit for the response capability for the person who is designated for closing the air lock door. The NRC

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptance of this specification was based on doses for a 2 hour release as well as a licensee commitment for a person designated to close the door quickly.

The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The radiological consequences of a fuel handling accident in containment have been evaluated assuming that the containment is open to the outside atmosphere. All airborne activity reaching the containment atmosphere is assumed to be exhausted to the environment within 2 hours of the accident. The calculated offsite and control room operator doses are within the acceptance criteria of Standard Review Plan 15.7.4 and GDC 19. Therefore, although the containment penetrations do not satisfy any of the NRC Policy Statement criteria, LCO 3.9.4 provides containment closure capability to minimize potential offsite doses.

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(continued)

BASES

LCO (continued)

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

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SR 3.9.4.2

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SR 3.9.4.3

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