Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877 7122

C. K. McCoy
Vice President, Nuclear
Vogtle Project

Georgia Power
the southern electric system

September 13, 1995

LCV-0636-B

Docket Nos. 50-424
50-425

Tac Nos. M74485
M74486

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

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INDIVIDUAL PLANT EXAMINATION

The enclosure of your letter dated June 16, 1995, contained requests for additional information that were developed during the NRC's review of the Individual Plant Evaluation for the Vogtle Electric Generating Plant. Attached to this letter are the responses to those requests.

Sincerely,

C.K.McC

C. K. McCoy

CKM/KWK/HWM/gmb

Enclosure

cc: See next page

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Georgia Power

cc: Georgia Power Company
Mr. J. B. Beasley, Jr.
Mr. M. Sheibani
NORMS

U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. L. L. Wheeler, Licensing Project Manager, NRR
Mr. C. R. Ogle, Senior Resident Inspector, Vogtle

LCV-0636-B

VOGTLE - UNITS 1 AND 2
INDIVIDUAL PLANT EXAMINATION SUBMITTAL

FRONT-END QUESTIONS

Question 1

Credit was taken in the analysis for three procedure enhancements, as described in Subsections 1.4.1 and 6.1 of the individual plant examination (IPE) submittal. However, the status of these enhancements is not clear. Although in some places in the submittal it is stated or implied that these enhancements have been implemented, the beginning of Subsection 6.1 states that the enhancements have been scheduled for implementation.

a. Please provide the status and schedule for completion of these procedure enhancements.

b. The IPE submittal indicates that these enhancements collectively reduce the core damage frequency (CDF) from 8.2E-05/yr to 4.9E-05/yr. If available, please provide an estimate of the CDF reduction resulting from each modification.

Response 1

a. The procedures identified to implement the plant improvements noted in Subsection 6.1 were all implemented in August 1992, prior to the IPE submittal. The specific procedures for each of the three improvement areas identified are as follows:

- Opening of power room doors upon loss of ESF Electrical HVAC

  13302-1&2; Control Building ESF Ventilation Systems
  17050-1&2; Annunciator Response Procedure for ALB 50 on QHVC Panel
  17053-1&2; Annunciator Response Procedure for ALB 53 on QHVC Panel

- Manual control of AFW turbine driven pump during a loss of all AC power and DC power

  19100-C, Emergency Operating Procedure, ECA-0.0 Loss of All AC Power

Response 1a (continued)

- Establishment of one NSCW pump operation on loss of NSCW initiating event

18021-C, Abnormal Operating Procedure, Loss of Nuclear Service Cooling Water System

b. The IPE submittal indicates that three procedural enhancements collectively reduce the core damage frequency. The benefit resulting from each procedural modification can be estimated using the most dominant sequences from an updated version of the Vogtle IPE model. This updated Vogtle IPE model has not resulted in significant changes to the major contributors to CDF and is therefore considered to be representative of the base model submitted in response to GL-88-20.

To determine the benefit of each procedural enhancement, a sensitivity case was performed to show the increase in CDF when the recovery action developed for each procedural change is not credited in the IPE model. This was done by setting the failure probability for the action that models the procedural change to 1.0 and then calculating new failure probabilities for impacted plant response tree top events. Table 1 contains the results of the sensitivity case for each procedural enhancement.

TABLE 1
CHANGE IN CDF WHEN PROCEDURAL ENHANCEMENTS ARE REMOVED FROM IPE MODEL

Procedure Enhancement	Percent Increase in CDF without Enhancement
Manual control of the turbine driven auxiliary feedwater pump during a loss of all AC power and loss of DC power (Station Blackout)	31%
The establishment of one NSCW pump operation on a loss of NSCW initiating event	9%
Opening of the inverter room and switchgear room doors on a loss of Control Building ESF Electrical Room HVAC	49%

Response 1b (continued)

The manual control of the turbine driven AFW pump is applicable to the loss of all AC power event (Station Blackout). Station Blackout, as an event, is a dominant contributor to CDF, therefore this procedural enhancement significantly reduces core damage frequency. The establishment of one NSCW pump operation is applicable to the loss of NSCW initiating event, however, this event has a very low frequency of occurrence and the procedural enhancement has a marginal benefit in reducing core damage frequency. Credit for high temperature Reactor Coolant Pump seals provides additional benefit for this event. The opening of the inverter room doors on a loss of CB ESF HVAC is applicable to all events where one or both trains of Control Building ESF Electrical Room HVAC fail due to component or support system failures. With the loss of Control Building ESF Electrical Room HVAC system, several rooms with important ESF electrical equipment, such as DC buses and panels and 480 V motor control centers reach temperatures that cause electrical equipment failures. This procedural enhancement significantly reduces core damage because the impacted electrical equipment is critical for actuating, controlling, and powering other equipment necessary to mitigate the consequences of all initiating events.

Question 2

According to the IPE submittal, the freeze date of the analysis was January 1, 1991, "with some exceptions." Subsection 2.1 of the submittal states that these exceptions are explicitly cited throughout the report; however, no explicit discussion of these exceptions was found in the submittal. It appears that one of these exceptions is related to the pending installation of new reactor coolant pump (RCP) O-rings in Unit 1 as of the IPE date, as credit for new RCP O-rings was taken in the analysis for both units. The only other possible exception to the analysis freeze date appears to be the procedure enhancements described above in question 1.

a. Please identify and describe all exceptions to the analysis freeze date.

b. If available, describe the impact of the "exceptions" on the CDF, both individually and collectively.

Response 2

a. The freeze date, January 1, 1991, referred to in Subsection 2.1 was the date established for the initial modeling and quantification of the PSA. It was established to provide a baseline date for design and equipment reliability data. After the initial quantification, the recovery process commenced, from which model changes were expected and subsequently implemented. As noted above credit was taken for the RCP high temperature O-rings installed on Unit 2 and scheduled for installation on Unit 1 (they have since been installed). Procedure enhancements were also identified and implemented (see response to 1a above) as a result of the recovery process. Two additional post freeze date items were: 1) the diesel generator reliability data (see IPE report Subsection 1.4.1 and 3.3.2) and 2) essential chilled water reliability data (see Subsection 3.2.2). Both of these systems had benefited from reliability program enhancements. The Independent Review Group recommended that the results of these program enhancements be included in the PSA in order to more accurately reflect the plant as built, operated, and maintained status.

b. The response to question 2a identifies the following exceptions to the freeze date credited in the analysis;

- RCP high temperature O-rings,
- procedural enhancements (see response to 1a),
- diesel generator, and
- essential chilled water reliability data.

Analysis that determines the impact on CDF of old reactor coolant pump (RCP) O-rings is not available. No analysis using old RCP O-rings is available because the decision to credit high temperature RCP O-rings was made early in the IPE development process. This decision was based upon the fact that, at the time of the IPE analysis, high temperature O-rings were already installed on Unit 2 and scheduled for installation on Unit 1. The high temperature O-rings have since been installed on Unit 1.

Procedural enhancements credited in the analysis and the impact of each enhancement on CDF is detailed in the responses to questions 1a and 1b.

To assess the impact of using post freeze date diesel generator and essential chilled water reliability data, sensitivity analyses were performed. Cases were run to assess the impact on CDF of using failure data up to the freeze date for components within the diesel generator and essential chilled water systems. Cases to assure each system individually and a case to collectively assure the combined impact of the systems were run. Table 1 contains the results (impact on CDF) of the sensitivity case for the diesel generator and essential chilled water systems.

TABLE 1 CHANGE IN CDF USING "FREEZE DATE" DG and ECW RELIABILITY DATA		
System	Dominant Contributor	Percent Decrease in CDF
Diesel Generator System	Failure to start DG	19%
Essential Chilled Water System	Failure to start ECW chiller	62%
DG and ECW System	Failure to start ECW system	67%

Question 3

The IPE includes loss of 120 volt AC instrument panels and dc buses as initiating events. However, no mention is made in the submittal of possible consideration of failures of 4,160-Vac and 480-Vac buses as initiating events. Please provide the basis for omitting, as initiating events, equipment failures related to 4,160-Vac and 480-Vac buses.

Response 3

Special initiating events (plant-specific initiating events) are those systems or component failures which result in a reactor trip or LOCA and simultaneously disable or degrade the performance of accident mitigation systems required to respond to the event. These events generally involve the loss of support systems, such as loss of nuclear service cooling water, loss of two 120V vital AC buses or one 125V DC bus, or loss of instrument air. For special initiating events, the loss of a system or component directly results in a reactor trip and the need for decay heat removal.

Because these events are plant specific in nature, a review of plant information such as the plant's design, abnormal operating procedures and operating history was conducted to identify these initiators.

In order to determine whether the loss of a plant system or component should be treated as a special initiating event, several factors were considered:

1. If the event frequency was below the frequency of approximately 1E-08/year, and the expected level of degradation to other plant systems was not significant, then the event was eliminated from further consideration.

2. If the event had the same effect on plant systems as a previously defined LOCA or transient event and the estimated frequency was less than that LOCA or transient event frequency, then the special initiating event was subsumed into the LOCA or transient event.

The first source examined for special initiating events was the Vogtle abnormal operating procedures. These procedures were reviewed to determine if 1) the loss of the event causes a reactor trip and 2) to determine what systems would be impacted by failure of that system. In addition, during the systems analysis, a review was conducted for an individual system's impact on other systems and for the potential to cause an initiating event. The Vogtle Units 1 and 2 reactor trip operating histories were also reviewed to determine if any special initiators had occurred at the plant.

The loss of a single 4160V or 480V emergency safeguards features (ESF) bus does not result in an immediate reactor trip. This event could require a manual shutdown due to Technical Specification requirements, however, orderly shutdowns such as this were not considered initiating events for the IPE.

The loss of both 4160V ESF buses is similar to the station blackout scenario, with the exception that the non-ESF buses could be available. Therefore, the station blackout condition is more limiting. The failure of both 4160V ESF busses is considered in the determination of a station blackout condition given a loss of offsite power, however, the failure probability of both emergency busses is an insignificant contributor. The initiating event frequency for a loss of both busses is also small compared to the loss of offsite power initiating event frequency. Therefore, the loss of both buses was not modeled as a separate initiating event.

The loss of a non-ESF 4160V or a non-ESF 480V electrical bus was not included as a special initiating event. The loss of a non-1E bus does not have an effect on the systems required for safe shutdown. It does affect the availability of some equipment modeled in the Vogtle IPE, for example, instrument air and main feedwater which were modeled as initiating events.

Question 4

The common-cause beta factors used in the IPE for residual heat removal pumps (RHR) and motor-operated valves (MOV) are substantially lower (more than an order or magnitude) than corresponding data in NUREG/CR-4550.

a. Please identify the source(s) of the common-cause data for these two component groups.

b. For the above components, explain how it was determined that the common-cause data from the sources used in the IPE are applicable to the Vogtle plant.

c. In your search for vulnerabilities, did you determine if the results of the analysis are sensitive to the use of these beta factors for this equipment? Please discuss the impact on the CDF if higher common-cause data had been used for these components.

Response 4a. and 4b.

Based on NUREG/CR-4550, Volume 1, Revision 1, the common cause factors were estimated based on a combination of the beta factor model from EPRI NP-3967 and the Binomial Failure Rate (BFR) model from Atwood's work in the early 1980's (NUREG/CR-2098, ?770 and 2099). The beta factors from EPRI NP-3967 were used for common cause events of two out of two components. For higher order common cause events, the beta factors derived for the two out of two components were multiplied by the ratio of common cause parameters from Atwood's BFR analysis. In addition, for NUREG/CR-4550, a review and classification of the generic and plant specific data was not performed. However, the data from EPRI NP-3967 was reviewed for applicability to the NUREG/CR-4550 plants and EPRI NP-3967 was the data source used to quantify most of the common cause basic events. Thus, comparison of these factors to the Vogtle-specific factors is not valid.

Three references were used as the basis in the formulation of Vogtle IPE common cause factors: NUREG/CR-4780 "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," EPRI NP-3967 " Classification and Analysis of Reactor Operating Experience Involving Dependent Events," and Plant Vogtle specific operating history.

When the common cause calculations were performed (1991), the EPRI common cause event data base was in an interim state (draft 1990 document). The data base was documented in a report EPRI NP-3967 issued in June 1985 and was later updated and issued in April 1992.

The Vogtle-specific Multiple Greek Letter (MGL) factors resulted from conducting a site specific review of the EPRI events data base (from 1990 draft EPRI document) for applicability to Vogtle and also the investigation into failure events at Vogtle for possible inclusion. Inclusion or exclusion of events was based on a review by cognizant personnel from Southern Nuclear, Plant Vogtle and Westinghouse. The rationale for eliminating or modifying events from the database was documented. In many cases, the common cause event could not happen at Plant Vogtle since the Vogtle-specific equipment configuration did not match the system/component configuration as depicted in the event's description.

For motor-operated valves, of the 41 MOV events reviewed by the experts, it was determined that 20 events were applicable to Plant Vogtle. For RHR pumps, of the 7 RHR pump events reviewed, 2 events were determined to be applicable to Plant Vogtle. The events were screened based on the following criteria (as described in NUREG/CR-4780):

- events that did not occur in the same time frame, such as second failures occurring after the restoration of the first, were discarded from the database
- events in which the same cause was not readily apparent were discarded from the database
- off-tolerance conditions, such as packing leaks and setpoint drifts, were discarded from the data base because they did not constitute a failure
- failures that were very easily recoverable were discarded from the database
- for those events where a defense mechanism exists, the events were discarded from the database. Defense mechanisms are a set of operation, maintenance, and design measures taken to diminish the frequency or consequences of common cause events.

These events were then mapped, as recommended in NUREG/CR-4780 with associated probabilities to calculate the MGL factors.

The Vogtle Independent Review Group (which included an independent PRA consultant from PLG) reviewed the process and documentation involved in the common cause analysis (see Table 5.3-1 of the Vogtle IPE submittal report) and their comments were included in the common cause analysis documentation.

Response 4c.

To assess the impact of using generic beta factors for the common cause failures of motor operated valves and residual heat removal pumps, sensitivity analyses were performed. Cases were run in which the common cause failure probability of basic events for MOVs and RHR pumps were recalculated using the generic beta factors supplied in NUREG/CR-4550. The new CCF values where then used to assess the increase in CDF. Table 1 shows the percent increase in core damage frequency when generic beta factors are used for motor operated valves and residual heat removal pumps. The use of generic beta factors for MOVs and RHR pumps does not significantly impact core damage frequency and would not change the insights from the IPE results.

TABLE 1 CHANGE IN CDF WHEN GENERIC BETA FACTORS ARE USED FOR CCF			
Component Type	IPE Beta Factors	NUREG/CR-4550 Beta Factors	Percent Increase in CDF Using Generic Beta Factors
Motor Operated Valves	6.9E-3	8.8E-2	5%
Residual Heat Removal Pumps	2.5E-3	1.5E-1	2%

Question 5

The plant response tree (PRT) for an interfacing-systems loss-of-coolant (ISLOCA) accident assumes that use of the high-pressure system in the injection mode would be sufficient to provide primary system makeup during the 24-hour post-accident mission time. Please explain how it was determined in the submittal that high-pressure injection could be continued throughout the post-accident mission time, given:

a. inventory depletion considerations and

b. possible adverse environmental effects of coolant discharged outside containment on equipment needed to sustain mitigating system operation.

Response 5a. and 5b.

The process of determining the interfacing systems LOCA (ISLOCA) initiating event frequency identified the RHR hot leg suction ISLOCA as a dominant contributor. This ISLOCA is generally considered the most severe ISLOCA due to its effect on the long term heat removal capability of the plant. In this event both RHR pumps are assumed to fail due to the ISLOCA, failing low pressure injection and ECCS recirculation from the containment sump. Therefore, as shown in the plant response tree on Figure A.9-4 of the Vogtle IPE submittal, this case was chosen for the ISLOCA model. For this event, two hot leg isolation valves in series fail, exposing the RHR system to the higher pressure of the RCS. The pressure increase fails the RHR pump seal for both RHR pumps and also opens the RHR relief valves. The pump seal failures are assumed to fail the RHR pump motors due to water spray. The flow from the RHR pump seals collects in the RHR pump rooms. The flow from the relief valves is expected to eventually fill the pressurizer relief tank and burst its rupture disc, spilling into containment. The result is a concurrent release of coolant inside and outside containment. Water leakage in the RHR pump rooms, or from the pressurizer relief tank, will not affect the operation of the high pressure ECCS because the active components (e.g., pumps and valves) are not located in these areas.

On the ISLOCA plant response tree, the only path which does not lead to core damage requires successful operation of the high pressure ECCS pumps (top event HPI), the containment cooling units (CCU), and operator action (OSR). The containment cooling units are required to prevent containment spray from being actuated due to the RCS flow into containment through the pressurizer relief tank. By preventing the containment sprays from actuating, the inventory in the RWST is available for RCS injection via the high pressure ECCS pumps. The operator action is required to reduce the ECCS flow which extends the water supply in the RWST.

Calculations were performed, using minimum required flow data for decay heat removal from Vogtle emergency procedure "Loss of Emergency Coolant Recirculation," to determine how many gallons of water are required for decay heat removal for the 24 hour IPE mission time. Based on this calculation and the flow rates of the high pressure pumps, the time available for the operator to reduce the ECCS flow was calculated. This operator action time and the procedural steps the operator would follow to reduce ECCS flow were used in the determination of the human error probability for top event OSR.

Question 6

The IPE submittal provides system success criteria for individual PRT headings. However, the submittal does not list the minimum success criteria for front-line systems that mitigate each initiating event or group of events, as requested in NUREG-1335. Without information compiled in this manner, it is difficult to determine success criteria pertinent to PRT success paths. Please provide a table that lists the complete set of minimum success criteria for the front-line systems required to prevent core damage for each of the initiating events.

Response 6

Tables FE6-1 through FE6-9-4 describe the success criteria for front-line systems in the top events for each plant response tree. Note that separate success criteria for the loss of offsite power initiating event are not included in the tables. The loss of offsite power initiating event follows either the general transients, small LOCA, secondary side break, ATWT, or station blackout plant response tree. For loss of offsite power and station blackout, restart of the operating centrifugal charging pump (CCP), component cooling water (CCW) pump, and containment cooling units (CCU) are included in the system unavailability calculations.

Table FE6-1 Large LOCA Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
Low-Pressure Injection (LPI)	1 out of 2 pumps injecting to 2 out of 3 intact cold legs.	30 minutes
High-Pressure Injection (HPI)	Not required to prevent core damage. If LPI fails, 2 out of 4 high pressure pumps (CCPs or SIPs) injecting to 2 out of 3 intact cold legs to provide water to containment for long term cooling.	1 hour
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours
Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	30 minutes
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
Low Pressure Recirculation (LPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains to 1 out of 3 intact cold legs.	10.5 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23.5 hours

Hot Leg Recirculation (HLR)	Not required to prevent core damage or containment failure. Alignment of 1 out of 2 RHR pumps, 1 out of 2 cross-connect valves opens, 1 out of 1 hot leg valve opens, 2 out of 2 cold leg valves close and flow to 2 out of 2 hot legs, or alignment returned to cold leg injection to prevent interruption of recirculation flow.	13 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-2 Medium LOCA Plant Response Tree System Success Criteria		
Event Tree Top Event	**Success Criteria**	**Mission Time**
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
High-Pressure Injection (HPI)	2 out of 4 CCPs/SIPs injecting to 2 out of 3 intact cold legs.	1 hour
Auxiliary Feedwater (AFW)	1 out of 3 pumps to 2 out of 4 steam generators (SG) with steam relief from 1 out of 5 safety valves/SG.	5 hours
Secondary-Side Depressurization (SGP)	2 out of 4 SG atmospheric relief valves (1/SG fed by AFW).	Not applicable
Primary-Side Depressurization (PZR)	2 out of 2 pressurizer PORVs open (and block valves if necessary).	Not applicable
Accumulators (ACC)	3 out of 3 tanks injecting to 3 out of 3 intact cold legs.	Not applicable
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours
Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	30 minutes
Low-Pressure Injection (LPI)	1 out of 2 pumps injecting to 2 out of 3 intact cold legs.	30 minutes
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours

High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 sump valve on 1 out of 2 trains, 2 out of 4 high pressure pumps (CCPs or SIPs) and valves to 2 out of 3 intact loops.	10 hours
Low Pressure Recirculation (LPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains to 1 out of 3 intact loops.	10.5 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23.5 hours
Hot Leg Recirculation (HLR)	Not required to prevent core damage or containment failure. <u>Low Pressure</u>: Alignment of 1 out of 2 RHR pumps, 1 out of 2 RHR cross-connect valves opens, 1 out of 1 hot leg valve opens, 2 out of 2 cold leg valves close and flow to 2 out of 2 hot legs, or alignment returned to cold leg injection to prevent interruption of recirculation flow. <u>High Pressure</u>: Alignment of 1 out of 2 RHR pumps to 1 out of 2 SI pumps, 1 out of 2 SI cross-connect valves close, 1 out of 1 hot leg valve opens and cold leg valve closes, flow to 2 out of 2 hot legs, or alignment returned to cold leg injection to prevent interruption of recirculation flow (1 out of 2 CCPs).	13 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-3 Small LOCA Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Reactor Trip (RT)	Reactor trip breakers open.	Not applicable
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
Auxiliary Feedwater (AFW)	1 out of 3 pumps to 2 out of 4 SGs with steam relief from 1 out of 5 safety valves/SG.	5 hours
High-Pressure Injection via CCP (CCP)	1 out of 2 pumps injecting to 3 out of 4 cold legs.	3 hours
High-Pressure Injection via SIP (SIP)	1 out of 2 pumps injecting to 3 out of 4 cold legs.	6 hours
High-Pressure Injection via CCP (HPI)	1 out of 2 pumps injecting to 3 out of 4 cold legs.	3 hours
Secondary-Side Depressurization (SGP)	2 out of 4 SG atmospheric relief valves (1/SG fed by AFW) or 3 out of 3 steam dumps open.	Not applicable
Primary-Side Depressurization (PRP)	1 out of 2 pressurizer PORVs open (and block valves if necessary).	Not applicable
Primary-Side Depressurization (PZR)	1 out of 2 pressurizer PORVs open (and block valves if necessary).	Not applicable
Accumulators (ACC)	3 out of 4 tanks injecting to cold legs.	Not applicable
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours

Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour
Low-Pressure Injection (LPI)	1 out of 2 pumps injecting to 3 out of 4 cold legs.	1 hour
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
Normal Charging (NCH)	1 out of 2 CCPs and the opening of valves in the normal charging flow path	21 hours
Normal RHR (RHR)	1 out of 2 RHR pumps, 2 out of 2 hot leg suction valves open on 1 out of 2 trains and discharge to 2 out of 4 cold legs.	21 hours
High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains, and 1 out of 2 CCPs or 1 out of 2 SIPs to 3 out of 4 cold legs.	21 hours for CCPs 18 hours for SIPs
Low Pressure Recirculation (LPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains to 2 out of 4 cold legs.	23 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-4 Steam Generator Tube Rupture Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Reactor Trip (RT)	Reactor trip breakers open.	Not applicable
Auxiliary Feedwater (AFW)	1 TDAFW pump or 2 MDAFW to 3 out of 3 intact SGs with steam relief from 1 out of 5 safety valves/SG.	5 hours
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
High-Pressure Injection (HPI)	1 out of 4 pumps (CCPs/SIPs) injecting to 3 out of 4 cold legs with AFW. 1 out of 2 CCPs injecting to 3 out of 4 cold legs without AFW.	3 hours
Terminate AFW to Ruptured SG (AFR)	2 out of 2 AFW valves to ruptured SG close.	Not applicable
MSIV Closure (MSR, MSI)	1 out of 2 MSIVs on ruptured SG close (MSR); 1 out of 2 MSIVs on 3 out of 3 intact SGs close (MSI).	Not applicable
Secondary-Side Depressurization (SGP)	3 out of 3 intact SG ARVs open or 3 out of 3 steam dump valves open (MSR successful) or 3 out of 3 intact SG ARVs open (MSR fails).	Not applicable
Primary Pressure Relief (PRP)	1 out of 2 pressurizer PORVs open (and block valves if necessary)	Not applicable
Primary-Side Depressurization (PZR)	1 out of 2 pressurizer PORVs open (and block valves if necessary)	Not applicable
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours

Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
Normal RHR (RHR)	1 out of 2 RHR pumps, 2 out of 2 hot leg suction valves open on 1 out of 2 trains and discharge to 2 out of 4 cold legs.	21 hours
High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains, and 1 out of 2 CCPs or 1 out of 2 SIPs to 3 out of 4 cold legs.	21 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-5
Secondary Side Break Plant Response Tree System Success Criteria

Top Event	Success Criteria	Mission Time
Reactor Trip (RT)	Reactor trip breakers open.	Not applicable
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
Steam Generator Isolation (SGI)	Isolatable Break: 1 out of 2 MSIVs and a MFIV or MFRV (and associated bypass valves) close on 3 out of 4 SGs. Non-Isolatable Break: AFW to faulted SG isolated and; 1 out of 2 MSIVs and MFIVs (and associated bypass valves) close on faulted SG, or 1 out of 2 MSIVs and a MFIV or MFRV (and associated bypass valves) close on 3 out of 4 intact SGs.	Not applicable
Auxiliary Feedwater (AFW)	1 out of 3 pumps to 2 out of 3 intact SGs with steam relief from 1 out of 5 safety valves/SG.	5 hours
High-Pressure Injection (HPI)	1 out of 4 pumps (CCPs/SIPs) injecting to 3 out of 4 cold legs with AFW. 1 out of 2 CCPs injecting to 3 out of 4 cold legs without AFW.	3 hours for break outside containment (no CS); 1 hour for break inside containment (with CS)
RCS Bleed (PZR)	1 out of 2 pressurizer PORVs open (and block valves if necessary)	Not applicable
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours

Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains, and 1 out of 2 CCPs or 1 out of 2 SIPs to 3 out 4 cold legs.	21 hours without CS 23 hours with CS
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-6
General Transients Plant Response Tree System Success Criteria

Top Event	Success Criteria	Mission Time
Reactor Trip (RT)	Reactor trip breakers open.	Not applicable
Turbine Trip (TT)	Turbine tripped (stop valves close).	Not applicable
Auxiliary Feedwater (AFW)	1 out of 3 pumps to 2 out of 4 SGs with steam relief from 1 out of 5 safety valves/SG.	5 hours
Condensate Feed (CON)	1 out of 3 condensate pumps, 1 out of 4 bypass feedwater reg. valves open, 1 out of 4 bypass feedwater isolation valves open, 1 out of 4 main feedwater isolation valves open, SG feed pump discharge valves open, with flow to 1 out of 4 SGs.	5 hours
Main Feedwater (MFW)	1 out of 2 SG feed pumps with associated equipment providing flow to 1 out of 4 SGs.	5 hours
Secondary-Side Depressurization (SGP)	1 out of 1 SG ARV or 3 out of 3 steam dumps open.	Not applicable
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
High-Pressure Injection (HPI)	1 out of 2 CCPs injecting to 3 out of 4 cold legs.	3 hours
RCS Bleed (PZR)	1 out of 2 pressurizer PORVs open (and block valves if necessary)	Not applicable
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours

Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains, and 1 out of 2 CCPs or 1 out of 2 SIPs to 3 out 4 cold legs.	21 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-7
ATWT Plant Response Tree System Success Criteria

Top Event	Success Criteria	Mission Time
CRDM MG Sets (MG)	2 out of 2 MG sets deenergized.	Not applicable
Control Rod System (CR)	Control rods inserted at least 48 steps for at least 1 minute.	1 minute
AMSAC (AM)	Turbine tripped and AFW actuated.	Not applicable
Auxiliary Feedwater (AFW)	For power level above 40% and MG fails, 4 out of 5 safeties open on 4 out 4 SGs, and either 3 out of 3 AFW pumps to 4 out of 4 SGs, or 2 out of 2 MD AFW pumps or 1 out of 1 turbine-driven AFW pump to 4 out of 4 SGs. For power level less than 40%, or if MG successful, 1 out of 3 pumps to 2 out of 4 SGs with 1 out of 5 safeties open/SG.	5 hours
Primary Pressure Relief (PPR)	3 out of 3 pressurizer safety valves and either 2 out of 2, 1 out of 2, or no PORVs (and block valves if necessary).	Not applicable
Pressurizer PORVs Close (PVC)	3 out of 3 pressurizer safety valves close and either 2 out of 2 PORVs or block valves close, 1 out of 2 PORVs or block valves close, or no PORVs or block valves close.	Not applicable
Secondary-Side Valves Close (SSC)	All secondary ARVs and safeties reclose after SG pressure decreases.	Not applicable
Main Steam Isolation (MSV)	1 out of 2 MSIVs on 3 out of 4 SGs close.	Not applicable
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours

Emergency Boration (EBR)	1 out of 2 boric acid transfer pumps and boric acid storage tank to 1 out of 2 CCPs, and CCP suction valve opens.	6 hours
High-Pressure Injection (HPI)	1 out of 4 pumps (CCPs/SIPs) injecting to 3 out of 4 cold legs with AFW. 1 out of 2 CCPs injecting to 3 out of 4 cold legs without AFW.	3 hours
RCS Bleed (PZR)	1 out of 2 pressurizer PORVs open (and block valves if necessary)	Not applicable
Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours
Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains, and 1 out of 2 CCPs or 1 out of 2 SIPs to 3 out 4 cold legs.	21 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-8 Station Blackout Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Pressurizer PORVs Close (PVC)	3 out of 3 pressurizer safety valves and 2 out of 2 PORVs stay closed or reclose after opening.	Not applicable
AFW Turbine-Driven Pump (TDP)	Flow to 3 out of 4 SGs with steam relief from 1 out of 5 safeties/SG.	4 hours
Secondary-Side Depressurization (SSD)	3 out of 4 ARVs open and 4 out of 4 accumulators inject.	Not applicable
AFW Turbine-Driven Pump Continues (TDC)	Flow to 3 out of 4 SGs with steam relief from 1 out of 5 safeties/SG (manual flow control).	4 hours with cooldown 20 hours without cooldown
AC and DC Power Restored (RPW)	One train of AC and DC power (Note: restoration of two trains was modeled in the IPE).	24 hours
NSCW Restored (RWS)	2 out of 3 NSCW pumps provide flow (Note: restoration of two trains was modeled in the IPE).	24 hours
Auxiliary Feedwater (AFW)	1 out of 3 pumps to 2 out of 4 SGs with steam relief from 1 out of 5 safety valves/SG.	5 hours
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
High-Pressure Injection (HPI)	1 out of 4 pumps (CCPs/SIPs) injecting to 3 out of 4 cold legs with AFW. 2 out of 2 CCPs injecting to 3 out of 4 cold legs without AFW.	3 hours
Pressurizer PORVs (PZR)	1 out of 2 pressurizer PORVs open (and block valves if necessary)	Not applicable

Containment Cooling Units (CCU)	4 out of 8 CCUs to prevent core damage and for containment cooling if 1 RHR HX not available; 2 out of 8 CCUs to prevent containment failure if 1 RHR HX not available.	24 hours
Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour
Component Cooling Water (CCW)	1 out of 2 trains of CCW with 2 out of 3 CCW pumps operating supplying the operating RHR train.	24 hours
High Pressure Recirculation (HPR)	1 out of 2 RHR pumps, 1 out of 1 sump valve on 1 out of 2 trains, and 1 out of 2 CCPs or 1 out of 2 SIPs to 3 out 4 cold legs.	21 hours
Containment Spray Recirculation (CSR)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps, 2 out of 2 sump valves in series open on 1 out of 2 trains for fission product scrubbing.	23 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-9-1 Loss of Instrument Air Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Unit 2 Air Compressors (U2C)	4 out of 7 air compressors, and isolation valves between the units open.	24 hours

Table FE6-9-2 Loss of Nuclear Service Cooling Water Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
NSCW One Pump Operation (RSW)	1 out of 2 NSCW standby pumps and associated valves.	24 hours
Establish Seal Injection (SINJ)	1 out of 2 charging pumps and associated valves providing RCP seal injection.	24 hours
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-9-3 Reactor Vessel Rupture Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Containment Cooling Units (CCU)	2 out of 8 CCUs to prevent containment failure.	24 hours
Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	30 minutes
Containment Isolation (CI)	Not required to prevent core damage. Identified penetrations >2 inches must be closed to reduce offsite doses.	Not applicable

Table FE6-9-4 Interfacing System LOCA Plant Response Tree System Success Criteria		
Top Event	**Success Criteria**	**Mission Time**
Refueling Water Storage Tank (TK)	≥631,478 gal. ≥2400 ppm boron, tank intact.	24 hours
High-Pressure Injection (HPI)	2 out of 4 pumps (CCPs/SIPs) injecting to 2 out of 4 cold legs.	24 hours
Containment Cooling Units (CCU)	2 out of 8 CCUs to prevent containment failure.	24 hours
Containment Sprays (CS)	Not required to prevent core damage or containment failure. 1 out of 2 CS pumps delivering flow to 1 spray header needed for fission product scrubbing.	1 hour

Question 7

It is not clear from the submittal whether plant changes as a result of the Station Blackout rule were credited in the analysis. Please provide the following:

a. Identify whether plant changes (e.g., procedures for load shedding, alternate ac power) made in response to the blackout rule were credited in the IPE and identify the specific plant changes that were credited.

b. If available, identify the total impact of these plant changes on the total plant CDF and on the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF)

c. If available, identify the impact of each individual plant change on the total plant CDF and on the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF)

d. Identify any other changes to the plant that have been implemented or are planned to be implemented that are separate from those in response to the station blackout rule that reduce the station blackout CDF

e. Identify whether the changes in d. above are implemented or planned

f. Identify whether credit was taken for the changes identified in d. above, in the IPE

g. If available, identify the impact of the changes identified in d. above, on the station blackout CDF.

Response 7

a. The only plant improvement specifically identified, resulting from the Station Blackout rule that was taken credit for in the IPE, was the opening of the Control Building electrical equipment room doors. As indicated in Section 6.3 of the IPE report no credit was taken for load shedding or possible sources of alternate AC power.

b. The total specific impact of the plant change due to implementation of the procedure noted in 7.a above is not available. This change was one of numerous items noted in the recovery process. The changes which resulted from a recovery meeting were taken collectively and incorporated into the model and a new CDF was calculated. The revised model was then used as the basis for the next recovery meeting. The impact, however, is included as part of the 49% CDF change noted in the response to question 1.b.

c. This information is not available (see 7.b. above).

d. An additional offsite power source, the standby auxiliary transformer (SAT), has been added to the low voltage switchyard. The SAT will serve as a swing offsite power source capable of connecting to any one of the 4160 Volt, Class 1E safety busses on either unit. The high side of the SAT is connected, by underground cable, to the Plant Wilson combustion turbine switchyard of GPC's Plant Wilson. The underground cable for the SAT can be fed either from the offsite grid or from Plant Wilson. Plant Wilson is a combustion turbine plant (6 CT's) adjacent to the Plant Vogtle boundary and is under the direct authority of Plant Vogtle management. Proposed Technical Specification (TS) changes associated with this design change will extend the allowed outage time (AOT) for the diesel generators from 3 to 7 days (provided the SAT is available) and provide for a one time per refueling cycle AOT of 21 days to allow on line diesel maintenance (again provided the SAT is available).

e. The Unit 1 design (noted in d above) was installed and functionally tested during 1994 Fall refueling outage. The Unit 2 design was installed and tested during the Spring 1995 outage. The TS changes have been submitted to the NRC for approval.

f. No, the SAT was a post IPE submittal improvement.

g. The Plant Wilson (SAT) design decreases the IPE CDF by 33.3 percent based on a conservative analysis performed on the top 85 IPE accident sequences which contribute a total of 71% of the CDF. The station blackout contribution to the CDF changes from 61% to 47%.

Question 8

The IPE submittal does not provide the basis for the RCP seal LOCA model.

a. Provide the basis for the IPE RCP seal LOCA model.

b. As stated on page 6-12 of the submittal, current procedures related to loss of nuclear service cooling water (NSCW) instruct the operator to trip the RCPs. However, the operator's failure to trip the RCPs is not included in the PRT for loss of NSCW. Explain why the failure of the operator to trip the RCPs in not reflected in the PRT for loss of NSCW.

c. Provide a discussion of the time-to-seal failure and the resultant leakage for the situation in which all cooling to an operating RCP is lost and the RCP is not tripped by the operators.

Response 8

The following responses provide information regarding the RCP seal LOCA model.

a. The probabilistic RCP seal LOCA model outlined in Reference 1, with the added conservatism outlined in Reference 2, is used for the calculations of seal failure leading to core uncovery as a function of time. Three postulated failure mechanisms for the reactor coolant pump (RCP) seal system binding and/or seal popping open are addressed for the number 1 and number 2 seals. Seal binding and seal popping open of either the number 1 seal or the number 2 seal are modeled as immediate failures (within the first 10 to 15 minutes of the event). The third seal is assumed to fail if both the number 1 seal and the number 2 seal bind or pop open. Binding or popping open of the second seal is also modeled as a possible immediate failure if the number 1 seal does not bind or pop open or if the number 1 O-ring extrudes. Failure of the third seal (binding or popping open) is postulated only if the third seal is subjected to exposure, i.e., the number 2 seal fails.

   O-ring extrusion is modeled as a time dependent failure with the failure rate changing (increasing) as a function of time. Two O-rings in each seal are conservatively considered to be critical to sealing in each seal section. If either critical O-ring fails, then the seal leakage is postulated to increase. If the number 1 critical O-rings do not fail in the first hour, there is an increased probability that one will fail in the second hour. O-ring extrusions of the second and third seals are only postulated if the seal is exposed to high temperature, or if the preceding seal fails. The model was simplified for use in the Vogtle IPE because the Vogtle RCP seal O-rings have been replaced with O-rings qualified for high temperature conditions. Therefore, following a loss of all cooling, only the failure mechanism identified as catastrophic binding of the pump shaft or seal popping are addressed. These postulated seal failures would occur within the first hour and would result in a seal LOCA of 480 gpm/pump. There is no other time dependence associated with the model when considering the O-rings installed at Vogtle.

   If there are no seal failures, the expected seal leakage is 21 gpm. If the number 1 seal and the number 2 seal either bind or pop open, the maximum seal leakage is postulated to be 480 gpm.

An event tree was developed to model the seal failures at each hour. However, only the event tree for the first hour applies to the Vogtle IPE because of the high temperature O-rings, as described in the second paragraph of this response. In this event tree, only the immediate failures are addressed (seal binding and seal popping open). If no failures occur, then the seal leakage rate of 21 gpm is shown on the first path. The combinations of binding and popping failures define the potential leakage rates. If the number 1 seal does not bind or pop, and the number 2 seal does, then the leakage rates depend on whether the number 3 seal remains intact. If the number 3 seal does not fail the leakage rate is 57 gpm; if the number 3 seal fails, the leakage rate is 183 gpm. If the number 1 seal either binds or pops and the number 2 seal remains intact, the leakage rate increases to 76 gpm. If the number 1 and number 2 seal both fail from either binding or popping, then the leakage rate increases to the maximum of 480 gpm. The leakage rates shown on Figure 1 are postulated to begin in the first hour and continue at these rates at succeeding hours.

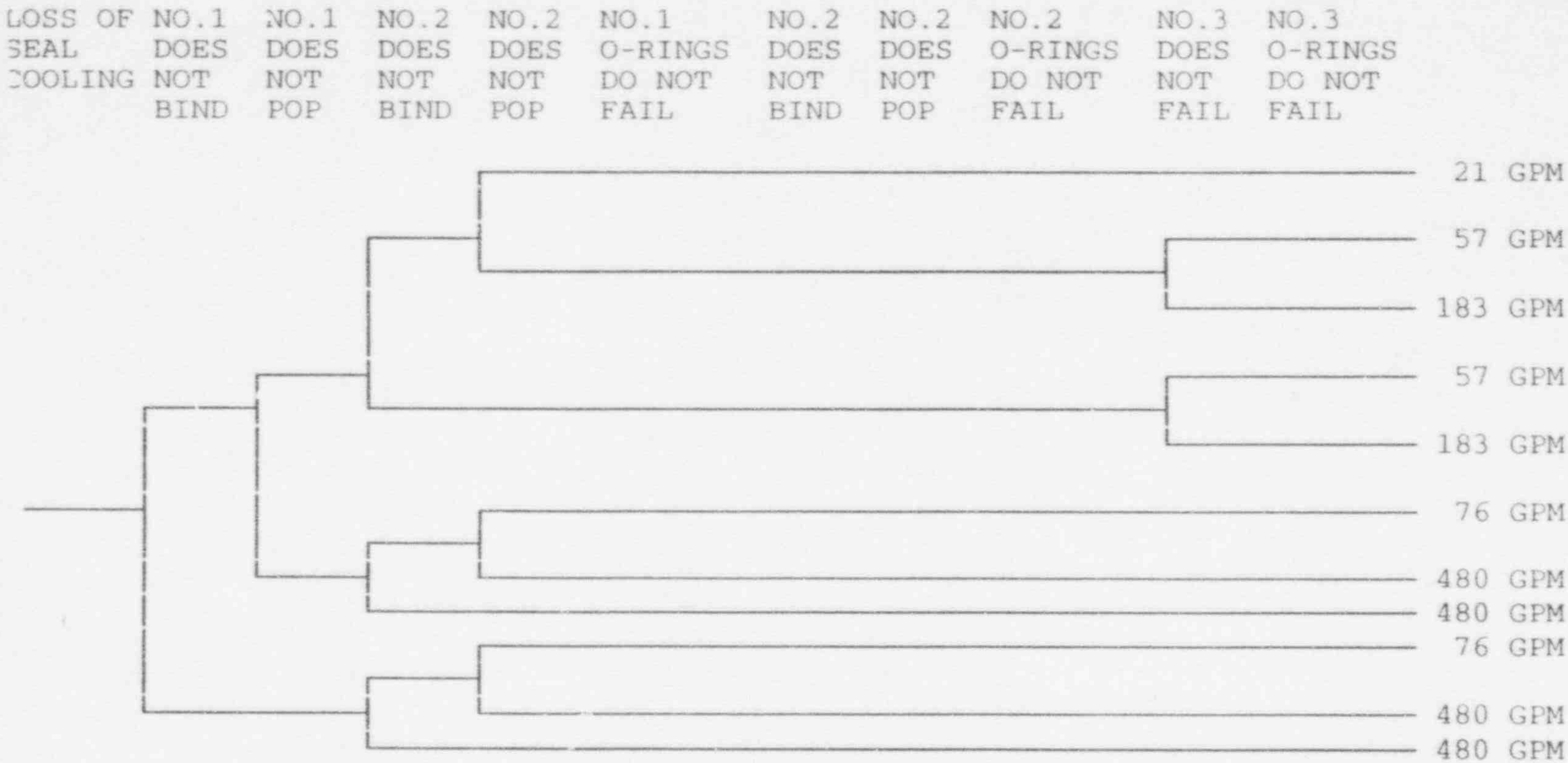


Figure 1
Seal Leakage During the First Hour

b. For the loss of nuclear service cooling water initiating event, plant response tree top event OSW is the operator action to establish one NSCW pump operation by continued operation of the auxiliary component cooling water system while reducing the heat loads to keep the RCP seals cooled, then establishing the single NSCW pump by isolating NSCW loads and starting the standby pumps. This top event is shown on Vogtle IPE Submittal Figure A.9-2. The description of the one NSCW pump operation is in submittal Section 6, Plant Improvements and Unique Safety Features (page 6-2). Operator action OSW models a subset of the actions listed on submittal page 6-3 for this procedural improvement. The actions modeled for OSW maintain RCP seal cooling. The actions modeled include reducing the ACCW and NSCW loads and starting the standby NSCW pumps. The actions do not explicitly include tripping the RCPs, however, it is judged that the actions modeled would still dominate the OSW failure probability if it included the failure to trip the RCPs. Based on the IPE core damage frequency results, OSW is not an important contributor and small changes in the OSW failure probability are not significant.

c. The probability of a complete loss of RCP seal cooling is low due to the systems which provide cooling. The RCP thermal barrier heat exchanger is cooled by auxiliary component cooling water (ACCW), which is cooled by nuclear service cooling water. As mentioned in the part (b) response, ACCW does not immediately fail upon loss of NSCW cooling allowing time for operator recovery action. The charging pumps provide seal injection, which will cool the seals if ACCW fails. The charging pumps are also cooled by NSCW. As shown on Vogtle submittal Table 3.1-1, the yearly loss of NSCW is calculated to be 1.4E-04 which makes the loss of RCP seal cooling a low probability event. If RCP seal cooling is lost and the pumps are not tripped, then the maximum seal leakage of 480 gpm is assumed because the seal failure would be an immediate failure as described in part (a).

References

1. Westinghouse Electric Corporation, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power", WCAP-10541, Revision 2, November 1986.

2. Westinghouse Electric Corporation, "RCP Seal Integrity, Generic Issue B-23 Slides Presented to the NRC", WCAP-11550, July 1987.

Question 9

The PRT for station blackout contains two recovery actions that are not clearly explained in the IPE submittal. These events are ORS, "operator action to restore systems following loss of offsite power/station blackout," and RPW, "restore power systems." Please identify:

a. The specific recovery activities modeled,

b. The time available for these activities, and

c. How sequence-specific considerations were addressed.

Response 9 a, b, & c.

The station blackout plant response tree top event ORS is the operator action to restore systems following loss of offsite power/station blackout. The primary and immediate objective of this action is to restore power to and operation of essential systems after at least one AC emergency bus has been energized. Following plant procedures, the operator would perform the following actions after AC power is restored:

Restore DC loads
Energize 480V AC switchgear
Energize battery chargers, instrumentation and control, emergency lighting, communications, and battery room fans
Reset containment isolation phase A (if actuated)
Verify instrument air available
Start ACCW and CCW pumps
Align reactor makeup system
Start CCP or CCP and SIP
Align for either normal charging or ECCS injection
Start containment fan coolers
Start RHR pump (if SI required)
Establish RCP seal cooling

These actions were included during the determination of the human error probability for ORS. The time available to complete these actions is 30 minutes, which represents the time the operator has from restoration of power to establishing core cooling prior to core damage. Sequence-specific actions have been addressed by the incorporation of the maximum number of actions which would be required at the point in the event tree where ORS is addressed. For example, the operator action includes starting the RHR pump, although it may not always be required based on the success of prior top events in the plant response tree.

The station blackout plant response tree top event RPW (restore power systems) models the AC and DC systems restored by the operator action ORS. Following restoration of AC power, the ESF AC buses and DC buses must energize. The battery chargers are loaded onto the ESF AC buses to provide DC power to the DC buses. To obtain the failure probability for RPW, the equipment failure probabilities for all trains of AC and DC power were combined. RPW models only equipment failures for the electric systems restored. It does not include operator failures. Success for RPW is defined as energizing both trains of AC and DC power.

Question 10

The PRT for station blackout includes events related to the recovery of offsite power, however, an explanation of the model for offsite power recovery could not be found in the IPE submittal. Please explain the offsite recovery model used in the analysis, specifically:

a. The basis for the recovery model,

b. How plant-specific considerations were accounted for in the recovery model, and

c. The offsite power recovery curve or data used in the analysis.

Response 10 a, b, & c.

Recovery of offsite power is quantified as a probability distribution. The distribution is based on the power recovery curve for cluster group 1 in NUREG 1032 (Reference 1).

The probability of recovering power is categorized in NUREG-1032 by factors which are determined for the site: the plant's switchyard configuration (I), losses of offsite power from grid-related events (GR), the probability of losing offsite power from severe weather conditions (SR), and the probability of losing offsite power from extremely severe weather events (SS). Recovery factors are also applied. These definitions are also consistent with the definitions and methodology outlined in Regulatory Guide 1.155 (Reference 2).

Plants are categorized as belonging to one of five offsite power cluster groups, based on the factors I, GR, SR, SS and the recovery factors. Figure A.15 of NUREG-1032 shows the distributions of the estimated frequency of occurrence of losses of offsite power exceeding specified durations for 5 offsite power cluster groups. Analysis of the factors for Vogtle Units 1 and 2 showed that these plants could be placed in power recovery cluster group 1. Therefore, the power recovery distribution for cluster group 1 was used to model the recovery of offsite power.

The frequency distributions were converted to probabilities (normalized to 1.0 at time 0) and are used to determine the probability of recovering power for each of the power cluster groups. The table below shows the data used based on cluster group 1.

Probability of Non-Recovery of Offsite Power,
and Conditional Probabilities, at Each Hour

Hour	Probability Power Not Recovered	Conditional Probability
0	1.00E+00	
1	2.48E-01	0.248
2	6.15E-02	0.248
3	4.27E-02	0.694
4	2.96E-02	0.693

5	2.46E-02	0.831
6	2.05E-02	0.833
7	1.71E-02	0.834
8	1.42E-02	0.832
9	1.27E-02	0.894
10	1.14E-02	0.898
11	1.02E-02	0.895
12	9.17E-03	0.898
13	8.22E-03	0.897
14	7.36E-03	0.895
15	6.60E-03	0.897
16	5.92E-03	0.897
17	5.30E-03	0.895
18	4.75E-03	0.896
19	4.26E-03	0.897
20	3.82E-03	0.897
21	3.42E-03	0.895
22	3.07E-03	0.898
23	2.75E-03	0.896
24	2.47E-03	0.898

The conditional probability is the probability of failing to recover power in a specific hour given the failure to recover power in the previous hour. For example, the probability of not recovering power at hour 3 is 4.27E-02. The conditional probability of not recovering power at hour 3 is .694 given that the probability of not recovering power at hour 2 is 6.15E-02 (i.e., .694 = 4.27E-02/6.15E-02).

The power recovery distribution is used for the station blackout event tree to evaluate three top events: 1HR, XHR and YHR.

<u>1HR - (AC Power Restored in 1 Hour)</u> - This top event is successfully restoring AC power within 1 hour. AC power must be restored within 1 hour if either a pressurizer safety relief valve fails open or the turbine driven AFW pump fails to start. The failure of this top event is quantified as the probability that power is not restored at one hour as determined from the power recovery distribution.

<u>XHR - (AC Power Restored Before Core Damage Occurs in X Hours)</u> - Conservatively assuming the turbine-driven AFW pump loses DC control power at four hours, and the pump cannot be manually controlled, the secondary side begins to boil and eventually decay heat removal would be lost and the core would start to uncover. Following the loss of flow from the turbine-driven AFW pump, AC power must be restored within about the next 2 hours to prevent core damage if the RCS cooldown was not successful (XHR = 6 hours). This success criterion is coupled with high pressure SI recovery. If the cooldown was successful, AC power must be restored within the next 4 hours (XHR = 8 hours).

If RCS cooldown is successful, the condensate storage tank (CST) inventory would last for approximately 8 hours and AC power must be restored within about the next 5 hours (XHR

= 13 hours). If the RCS cooldown is not successful, the CST will contain sufficient inventory for over 24 hours. AC power must be restored within 16 hours to address core uncovery from the minimum RCP seal LOCA (XHR = 16 hours). The conditional failure probability is determined from the power recovery distribution.

YHR - (AC Power Restored Before Containment Failure Occurs in Y hours) - If AC power is not restored in time to prevent core damage, then this top event addresses restoring AC power to prevent containment failure. If core damage occurs, then AC power must be restored within some time period following vessel failure to activate containment systems. To conservatively bound the situation in which a pressurizer relief valve sticks open or RCP seal leakage becomes high early in the transient, a single time of 20 hours following the start of station blackout is selected for YHR. Based on XHR times of 1, 6, 8, 13, and 16 hours, power must be restored within 19, 14, 12, 7, and 4 hours respectively, following these core uncovery times to prevent containment failure. The failure probability of YHR is determined as a conditional probability from the AC power recovery distribution at each of the designated hours.

References

1. U. S. Nuclear Regulatory Commission, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," NUREG-1032, June 1988.
2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.155, Station Blackout, June 1988.

Question 11

The IPE submittal does not appear to identify all the types of failures considered in the modeling of an interfacing system LOCA (ISLOCA). Quantification of an ISLOCA initiating event is provided in Table 3.1-1 of the submittal with no accompanying discussion of the types of ISLOCAs that are represented. Pages 4-64 and C-4 of the submittal indicate that the base-case ISLOCA was modeled as a 0.1 square foot break in the RHR hot-leg outside containment and that this area was based on an upper bound of the pump seals. However, it is not clear whether other ISLOCAs beyond this base case were considered. NUREG-1335 requests that the rationale for grouping of initiating events be provided.

a. Describe the method used to identify ISLOCAs.

b. Describe all the types of ISLOCA that are accounted for in the IPE analysis.

c. Provide the basis used to exclude any category of ISLOCA from the analysis.

Response 11

a. ISLOCAs can be divided into two categories according to the location of reactor coolant system loss relative to containment: those for which the coolant remains within containment and those for which the coolant escapes the containment.

In general, the potential consequences of an ISLOCA outside containment are more severe than those of an ISLOCA inside containment. The limiting factor in an ISLOCA inside containment is the loss of the function performed by the breached interfacing system. The limiting factors in an ISLOCA outside containment are the loss of the function performed by the breached interfacing system, the failure of containment isolation, and the loss of emergency core coolant inventory for long term core cooling. If an ISLOCA outside containment cannot be isolated before a significant fraction of the reactor coolant inventory escapes the RCS, a significant release of fission products to the environment may result. Because they are generally more severe, the Vogtle Electric Generating Plant (VEGP) IPE assumed ISLOCAs outside containment bounded those inside containment.

The ISLOCAs were identified from the systems which interface with the Reactor Coolant System (RCS) and may be subjected to normal RCS operating pressure. Specifically, the emergency core cooling system (low pressure injection, accumulators, high pressure injection), chemical and volume control system, auxiliary component cooling water to the reactor coolant pumps, and the sampling system were examined. VEGP piping and instrumentation drawings were examined to identify all significant ISLOCA flow paths. Significant [illegible]aths are those with a diameter greater than 3/8 inch and through which low p[illegible] piping outside containment could be exposed to RCS pressure. Further, low pressure piping is any piping system whose pressure boundary would be expected to fail in whole or in part when exposed to normal RCS operating pressure. Two examples of pressure boundary failure are pipe rupture and pump seal failure. Significant ISLOCA pathways were also examined to identify instrumentation and valves which may be useful in diagnosing and isolating an ISLOCA. The normal alignment, accident alignment, and setpoint or actuation signal were then identified for use in the ISLOCA frequency calculation.

b. In general, all significant pathways from the RCS interfaces to piping outside containment rated for pressures below that of the RCS were examined during the ISLOCA analysis. The total ISLOCA initiating event frequency is the sum of the following individual event frequencies:

Reactor Coolant Pump Seal Water Return Line,
Reactor Coolant Pump Thermal Barrier Heat Exchanger,
Lines to Charging Pump Discharge Header,
Safety Injection Pump Discharge Lines,

RHR Discharge Lines and,
RHR Suction Lines.

The process of determining the interfacing systems LOCA (ISLOCA) initiating event frequency identified the RHR hot leg suction ISLOCA as a dominant contributor. This ISLOCA is generally considered the most severe ISLOCA due to its effect on the long term heat removal capability of the plant. In this event both RHR pumps are assumed to fail due to the ISLOCA, failing low pressure injection and ECCS recirculation from the containment sump. Therefore, this ISLOCA was chosen for the case to model as shown in the plant responce tree on Figure A.9-4 of the Vogtle IPE submittal.

c. Pathways through pipes with an inside diameter less than or equal to 3/8 inches were not considered significant since the maximum RCS flow rate through these pipes is less than the capacity of the normal charging system. Any ISLOCA through pipes this small would not be expected to directly generate a reactor trip or SI signal, giving the operators sufficient time to identify and isolate the leak or take the plant to cold shutdown per the plant Technical Specifications (Reference 1).

More specifically, pathways between the accumulators, pressurizer relief tank (PRT), and reactor coolant drain tank (RCDT) and piping outside containment are not considered significant pathways for ISLOCA outside containment because the pressure ratings of the accumulators, PRT, and RCDT are significantly below normal RCS pressure. Therefore, the pressure relief valves and/or rupture disks installed on these components would result in primary coolant loss inside containment rather than outside containment.

Additionally, pathways through the normal letdown line are not considered significant ISLOCA pathways because of letdown orifices installed in the lines. The three parallel letdown orifices installed in the normal letdown line are sized to allow a combined maximum RCS letdown flow which, in addition to normal RCP seal injection flow, is below the capacity of normal charging. There is also no credible failure mechanism whereby the effective inside diameter of the letdown orifice is dramatically increased without rupturing the orifice itself. Any catastrophic rupture of a letdown orifice would effectively be a pipe break inside containment at the orifice.

Finally, leakage of reactor coolant through the steam generator tubes is excluded from consideration because tube rupture events are addressed as a separate initiating event.

References

1. Vogtle Electric Generating Plant Technical Specifications, Limiting Condition for Operation 3.4.6.2.

HUMAN RELIABILITY QUESTIONS

Question 1

To identify the more likely accident sequences that could occur and to determine the potential vulnerabilities as a consequence of these severe accidents, an understanding of the potential of the human contribution to an accident is required. Identification of the human events that can disable a system, such as failure to properly restore after test or maintenance or miscalibration of critical instrumentation, are essential to the human reliability analysis. The submittal includes consideration of restoration of a limited number of manual valves after testing. However, no mention is made in the submittal of consideration given to the disabling of a system as a result of miscalibration of critical instrumentation. Please provide a discussion of the rationale and justification for not considering miscalibration of critical instrumentation in the IPE analysis.

Response 1

The method used for including pre-initiator human errors in the fault trees used the following guideline.

> The failure of test and maintenance personnel to return valves, pumps, and other safety system components to their normal position after test and/or maintenance is considered as a credible fault in development of fault tree models if: 1) proper valve positioning cannot be detected using specified pump flow tests; and/or 2) valve or other component misposition is not immediately detectable by status lights and/or alarms at the main control board, and the valve is not automatically realigned by an ESFAS signal.
>
> Miscalibration errors were not modeled in the IPE, since such errors would be as likely to produce an early actuation as a delayed or prevented actuation. Moreover, there are normally multiple input signals or actuation devices. Miscalibration errors have seldom been shown to be important in past probabilistic risk assessments.
>
> There are numerous procedures that take equipment out of service for test and maintenance, and require operator action to restore that equipment to service. For most powered equipment, the failure of these steps would result in either an annunciator or a status light indication in the control room, alerting the operators to the misposition. The instrument status lights provide clear indication, and checks during each shift and at shift changes would reveal burned out indicators. Further, most safeguards equipment receives confirmatory signals to assume the correct position if an ESFAS signal is generated. Therefore, misposition failures are not included for such equipment, per item 2) in the above guideline.

For non-powered equipment, such as manual isolation valves, failure to restore to the correct position following a test or maintenance would not be readily detectable by control room personnel. Therefore, misposition errors were included for any manual valve that would disable the functioning of systems modeled in the IPE.

Question 2

The submittal does not clearly describe the method used to identify and select response and recovery-type actions for analysis. The method used should confirm that the plant procedures, design and operational practices and policies were examined and understood in order to identify potential severe accident sequences. Please provide the following:

a. A description of the process that was used for identifying and selecting the response-type actions evaluated.

b. A description of the process that was used for identifying and selecting the recovery-type actions evaluated.

Response 2

a. The response-type operator actions were identified and selected through an iterative process with the use of event sequence diagrams (ESDs). The ESDs, prepared by the event tree or system analysts, show the accident progression, in a block logic format, for the major categories of initiating events: large LOCA, medium LOCA, small LOCA, steam generator tube rupture, general transients (e.g., reactor trip, loss of offsite power, steam line break), and anticipated transient without scram.

The ESDs display success and failure paths of critical safety functions and operator actions, and take into account (1) Engineered Safety Features (ESF) equipment which actuates automatically following reactor trip or SI signals, (2) the consequences of failure of ESF equipment, and (3) any key decision points or operator actions called for in the emergency operating procedures (EOPs) which significantly alter the progression of the accident.

As stated previously, development of the ESDs was an iterative process. The event tree/system analyst prepared the draft ESD in collaboration with the human reliability analyst. Talk-throughs were then held at the site with the analysts and plant operations personnel to review the ESDs and form agreement on the response-type operator actions to be selected.

The ESD developed for the large LOCA initiating event is provided as an example; Table 1 identifies the symbols and notes, Table 2 lists the acronyms, and Figure 1 shows the large LOCA ESD.

b. The recovery analysis for the Vogtle Electric Generating Plant (VEGP) IPE was conducted, after the majority of the plant response tree and fault tree modeling had been completed, through an iterative process generally consisting of the following steps:

   - reviewing quantification results to determine where contribution to core damage frequency of dominant contributors could be reduced through credit for appropriate and reasonable actions or equipment not already in the IPE models;

   - modeling these actions or equipment;

   - requantifying the results;

   - and repeating the process to address new dominant contributors.

   In general, operator actions called for in the VEGP emergency or abnormal procedures were considered to be expected rather than recovery actions, as long as there was a clear path through the procedures for each event being considered. Actions that could be taken from the control room (e.g., manually starting a pump that failed to start automatically, operating valves that failed to actuate automatically, and so forth), and for which the operators would receive indication as to the need for the action, were treated as anticipated responses rather than recovery actions. Such actions are generally included in the fault tree quantification for the appropriate top event. Actions for which all or most of the diagnosis and response required outside-control-room activity were generally considered as recoveries.

For the recovery analysis process, the specific steps included:

- identifying possible recoveries in the fault tree models, plant response tree (PRT) models, support system models, or combinations of these;

- identifying necessary additional modeling;

- discussions and meetings among IPE analysts and SNC personnel familiar with VEGP and the IPE to clarify and verify the actions to be taken or equipment needed;

- modifications to the various models and quantification input values (including additional human reliability and success criteria analysis, where needed) to accommodate the recoveries;

- requantification of results; and

- modify/improve plant procedures, if necessary, to credit recovery actions assumed in the IPE.

The modeled recoveries generally take one of several forms:

- credit for existing systems or procedures that were not included in the initial models;
- credit for procedural enhancements that were proposed by the IPE analysts or SNC personnel, and, after review, deemed acceptable by VEGP personnel and committed to be implemented at the plant;
- credit for equipment modifications that were proposed by the IPE analysts or SNC personnel, and, after review, deemed acceptable by VEGP personnel and committed to be implemented at the plant; or
- combinations of the above.

For most of the items selected, a summary description was prepared. These summary descriptions briefly describe the situation for which a recovery is needed, provide information on the operator actions and time available and any plant equipment required, reference to the HRA quantification of the modeled actions, information on available procedural guidance, and an indication as to how the recovery is to be factored into the event tree and/or fault tree models. Each of the summaries was reviewed and commented on by SNC and VEGP personnel, corrections were made, and the necessary modeling changes were made.

Question 3

The submittal does not indicate whether a screening process was used to help differentiate the more important post-initiator human events. If a screening process was used, please provide the screening value(s) used and the basis for the value(s); that is, provide the rationale for how the selected screening value did not eliminate (or truncate) important human events. Also, as requested in NUREG-1335, provide the list of errors that were screened.

Response 3

The Vogtle IPE did not use a screening approach as part of the human reliability analysis. As discussed in the response to HRA Question 2, the post-initiator human errors are identified as important to the overall plant response during the process of defining the event sequences. This process included reviews of the VEGP Emergency Operating Procedures, Abnormal Operating Procedures, and System Operating Procedures, and discussions with VEGP and SNC personnel. This resulted in the development of event sequence diagrams and the event tree models, which were also reviewed with VEGP and SNC personnel with operations and training background.

Question 4

Although the SLIM method provides a way of interpolating between "anchor point" failure probabilities provided outside the SLIM method, inappropriate selections for anchor points will produce inappropriate results from the SLIM process. Limitations in the method(s) used to calculate the anchor points may "flow through" to the SLIM - derived probabilities. Please provide a detailed rational for, and description of, the selection of the human error events identified as anchor points in the submittal.

Response 4

The selection of the success likelihood index method (SLIM) "anchor points" operator actions was based upon accepted industry methodology. The first step in the process was to relate the Vogtle specific success likelihood index (SLI) values to known or accepted values of error probabilities (P_e) for two specific Operator Actions (OA) contained within a specific group of operator actions. This required the use of "reference" OAs that were very similar if not identical to the Vogtle IPE operator actions being evaluated. The methodology, which was followed, was to select specific operator actions that had been evaluated through means other than SLIM, (e.g., empirical simulation data; THERP HRA methodology) to arrive at their estimated values for P_e.

Following the SLIM technique, the selection of the anchor point operator actions must address two criteria. First, it must reference the psychological model that underlies the SLIM methodology. This holds that if expert ratings reflect the actual state of the defined performance shaping factors (PSFs) at the plant, then positively-rated PSFs (leading to high SLIs) indicate a plant environment that is conducive to avoiding error for their OAs. Likewise, negatively-rated PSFs (low SLIs) indicate an environment that is error-prone for their OAs.

Second, the estimated relationship between P_e and SLI for each group of operator actions must extend over the range of P_e to approximate the range achieved by other methodologies. For a function relating $\mathrm{Log}_{10}(1\text{-}P_e)$ to SLI, this suggests the optimal slope is a positive one between those two OAs selected as the anchor points. Ideally, at one extreme of the function, the OA from a correlation group with the highest P_e would match an OA that has the lowest SLI. At the other end, the OA from the same correlation group with lowest P_e would match an OA with the highest SLI.

Thus, the selection of which specific operator actions to use as anchor points was based on:

- Operator actions that were very similar if not identical to the Vogtle IPE operator actions being evaluated, such as the need to open/close valves, start/stop equipment, etc.

- Operator actions that were modeled within the same context for which the Vogtle IPE operator actions were being evaluated, such as the same initiating event, event sequence, and timing.

- Operator actions that would have the similar Performance Shaping Factors (PSFs) as the Vogtle IPE operator actions being evaluated.

- Operator actions that were commonly modeled in industry PRA studies such that sufficient data exist to select a typical operator error probability.

- Operator actions with the highest or lowest SLI values within a group. If no industry data could be found or conflicting data was found, the appropriate operator actions with the next highest or lowest SLI values were used.

The selection of operator action anchor points was done for four (4) specific groups of operator actions that were grouped together based on their similar PSFs profiles. Thus, four times as many reference points were used in the calculation of the Vogtle IPE SLIM operator action values as would have been the case if using one group consisting of all operator actions.

When the operator actions were selected as anchor points, care was taken to ensure that the corresponding SLI values bounded the range of SLI values within the group of operator actions being considered. By following this practice a more accurate mathematical relationship can be developed that will avoid calculating erroneous operator action error probabilities.

Based on the above considerations, the following operator actions were selected as anchor points for each group:

Group 1	OALa	Realign ECCS low pressure system for hot leg recirculation
	OARb	Establish/realign ECCS high pressure system for cold leg recirculation
Group 2	SGI	Isolate faulted steam generator
	OABb	Establish feed and bleed cooling of RCS (actuate SI)
Group 3	OCIb	Isolate containment - with power - auto align
	OSR	Minimize ECCS flow following Interfacing Systems LOCA
Group 4	ORT	Initiate manual reactor tr p
	OAPa	Depressurize primary side (MLOCA, SLOCA)

Question 5

Use of PRA-generated anchor points as SLIM anchors requires assessment of the performance shaping factors (PSFs) used in the SLIM assessment for the anchor events. It is not clear in the submittal that some Vogtle PSFs, such as complexity, would be described in the sources of the anchor points selected from other PRAs. Please provide a description of the assessment of PSFs to these "anchor point" events to show that the anchoring process was performed correctly.

Response 5

As discussed in the response to Vogtle IPE HRA question 4, one aspect of the selection of the operator action anchor points was to consider operator actions modeled in other Probabilistic Risk Assessment (PRA) studies that would have similar Performance Shaping Factors (PSFs) as the Vogtle IPE operator actions being evaluated.

Before the operator action anchor points were selected, a general assessment was performed of the Vogtle IPE PSFs and how they would compare to the known or expected PSF profiles for the reference PRA operator actions. This general assessment or comparison can be illustrated by reviewing the Vogtle IPE PSFs and the manner in which they were considered during the selection of the reference plant operator actions as anchor points.

1. Complexity of the Operator Action (CPX) - The reference operator actions were reviewed to determine if the number of tasks, type of tasks, task sequencing, and relationships among the tasks were similar to those considered in the Vogtle IPE. For example, would the operator be required to perform similar tasks to realign the emergency core cooling low pressure injection system for hot leg recirculation (OALa) or establish/realign ECCS high pressure system for cold leg recirculation (OARb).

2. Time Factors (TIM) - The reference operator actions were reviewed to determine if the time factors were similar to those considered in the Vogtle IPE. For example, the Vogtle operator action to initiate a manual reactor trip (ORT) was defined as having a six minute time window for the operator to complete the required actions. The reference operator actions were then reviewed to make sure that the time windows for the similar operator action was comparable to the Vogtle time window.

3. Crew's Level of Knowledge, Training, and Experience (TRN) - The Vogtle IPE assumed that the Vogtle operating crew's level of knowledge, training, and experience was at least similar to if not better than that of the operating crews of the reference PRA plants. There were no indications of any negative operator performance issues that would impact or change this assumption.

4. Adequacy of Guidance Materials (PRC) - The Vogtle IPE assumed that the guidance material available to the operating staff, such as procedures, databases, job performance aids, and technical specifications were at least similar to if not better than those available to the operating crews of the reference PRA plants. This is based on the fact that the reference PRA plants are all Westinghouse NSSS plants and as such all utilize the Westinghouse Owners Group (WOG) Emergency Response Guidelines as the basis for developing their plant specific Emergency Operating Procedures. In addition, all the anchor point operator actions that were selected for the Vogtle IPE are specifically contained in the WOG Emergency Response Guidelines. Based on this information and a review of the emergency operating procedures for Vogtle and the reference PRA plants, this assumption was considered to be valid.

5. Characteristics of the Interface relevant to this task (MMI) - The reference plant operator actions were reviewed to ensure that the "interface" of the anchor point operator action, such as where the actions are to be performed (remote or local), the need to dispatch an operator to perform a specific action (i.e., trip a breaker, install a jumper, or operate a valve), was similar to the corresponding "interface" of the operator action being evaluated for Vogtle.

6. Previous, Subsequent, and Concurrent Actions (ACT) - The reference plant operator actions were reviewed to determine if the effects of other actions that are not part of the operator action being evaluated were or were not considered. For example, in order to assess the appropriateness of the operator action to isolate the faulted steam generator (SGI), it was necessary to review portions of the reference plant event sequences in which this action was modeled. Modeling of events, such as a steam generator tube rupture, generally includes several operator actions, some of which are dependent on the success or failures of preceding operator actions.

7. Stress (STR) - The Vogtle IPE assumed that the level of stress experienced by the operating crew was the same for the operating crews of the reference PRA plants. This PSF considers that some aspects and levels of stress may actually be beneficial from the standpoint of PRA analysis, by bringing the crew to a higher state of alertness, decreasing their likelihood of committing an error. Other aspects and degrees of stress can create a more error-likely situation. Other elements of stress may include the operators' perceptions of threat to themselves or their jobs; physical stressors such as noise, vibration, radiation, humidity, temperature, and light levels; belief by the crew that their knowledge and understanding of the current plant status is inadequate (degree of

surprise) and that they may not have adequate resources (including time) to deal satisfactorily with the event; and the nature of expected surveillance in the event by plant management, regulatory personnel, or others. It was judged that the stress brought-on from the need to perform a particular operator action in response to an accident sequence that is similar from one plant to the next, in terms of the other PSFs (timing, training, complexity, etc.), would be the same.

Question 6

It is not clear from the submittal what the bases were for calculating response-action human-error probabilities (HEPs) through the application of the seven plant-specific PSFs in the SLIM process. Please provide a discussion and examples of the process used to determine the appropriateness of applying these PSFs to post-initiator response-action human events. Please illustrate this discussion with Operator Action Summaries and PSF assessment survey sheets for each of the following human errors:

- OMG	- OABa and OABb
- OCIa	- ORS
- OARaLP	- OAS

Response 6

For each Operator Action (OA), SLIM was used to generate a Success Likelihood Index (SLI) that provides a basis for deriving an error probability (P_e). The SLI was generated through the use of expert sessions (plant operators) to evaluate the degree to which the seven performance shaping factors (PSFs) influence the probability of human error in carrying out that OA. For each OA, expert ratings of both the importance and the effects of all seven PSFs were combined mathematically to compute the SLI, which can then be transformed into a P_e.

Prior to conducting the SLIM expert sessions, the IPE analysts prepared written summaries of each operator action to be evaluated. These summaries described the operator action and the sequence in which the operator action is being modeled, identified success criteria and the available time window for performing the action, and summarized the applicable portions of the Vogtle emergency procedures that the operators would be following to accomplish the action.

These written summaries were reviewed by SNC personnel with experience in Vogtle Operations. The objectives of these reviews were: to ensure that the descriptions in the IPE summaries utilized the same terminology used by plant operators and the emergency procedures; to enhance the ability of the operators to quickly recognize the scenario being described by the IPE personnel; and to make any necessary corrections to the summaries so that they included the proper emergency procedure steps and references. The summaries were revised as necessary before the expert sessions.

The requested Operator Action Summaries and operator evaluation sheets for the requested operators actions are provided at the conclusion of this response.

The next step was to collect data from experts about how these PSFs influence the OAs. This was done by using two Vogtle operating crews to assess the specific influences of each PSF on their own performance of each OA. Each crew consisted of a Supervisor (licensed Senior Reactor Operator (SRO)), and two licensed Operators. The data collection session was conducted by an IPE team, consisting of an HRA expert who was knowledgeable about SLIM, a PRA analyst, the IPE project leader, and an IPE liaison from the Vogtle project. The latter three were available to resolve questions concerning the way IPE actions were modeled. Since the crew members were not necessarily familiar with the ongoing IPE, they were provided with an introduction of the initiating events, the list of operator actions, and an explanation of the SLIM process.

The PSF descriptions, and the manner in which the operating crews evaluated them, are as follows:

PSF 1: Complexity of the Operator Action
This PSF looks at the number, kind, sequence, and relationships among elements within the task. This includes requirements for synthesizing multiple sources of information: keeping track of the crew's progress through a long series of steps which may require coordination and communication among members of the crew, or with other plant staff outside the control room; assessing whether the resources required for this task are immediately available, or require a significant degree of preparation (e.g., assembly, alignment) before they can be brought to bear.

During the sessions, for a given action, the operators tended to evaluate the number and complexity of the selected Vogtle emergency procedure steps for the current action against Vogtle procedure steps for other actions with which they also had experience.

PSF 2: Adequacy of Time Factors
This PSF takes into account the adequacy of time available to accomplish the action under consideration. Depending on the action and its context, adequacy of time can apply in different ways. One application of the timing PSF is: what is the window for diagnosing and starting the action, and how is this window defined? For example, the time window for some actions may be defined in the IPE in terms of minutes from the start of the event until the time an action is required; but if the operators may not be immediately aware of event initiation, their available time for taking action may be less. For other actions, the time window might be defined in terms of "compelling signals" (e.g., alarms, equipment status, and so forth), so that the timing definition may be clearer.

During the sessions, the operators were provided, for each action, information from the Vogtle IPE success criteria analyses regarding the available time window for initiation and/or completion of the action. Specific timing simulations were not performed for the IPE. However, review of this information with the operators tended to result in a discussion of the IPE-defined time window boundary conditions (e.g., the amount of time between the occurrence of a steam generator tube rupture and isolation of the ruptured steam generator by the operators) relative to the operators' experience with time needed to perform the identified emergency procedure steps on the Vogtle simulator.

PSF 3: Crew's Knowledge, Skills, Training, and Experience

This PSF assesses what the crew brings to the event, in the form of ready-to-use knowledge and skills that are assumed to result from the combined training and experience of all members of a crew. This includes assessments of skills in communication, team training, and leadership, as well as knowledge of procedures, the reasons for them, and familiarity with the plant, its components, and the relevant controls and displays. It is also expected to assess the degree of confidence the crew has in dealing with the event in question, which in turn may affect assessments of other factors. For example, a crew that has recently performed well in rigorous training on the event, and specifically on the operator action in question, is expected to be less likely to be negatively influenced by factors such as "stress" or "adequacy of time".

The expert sessions were conducted with Vogtle operating crews, with similar and plant-specific training, and a similar mix of operating experience among each crew. Each crew's assessment of this PSF reflects its experience in working together in operating the plant.

PSF 4: Adequacy of Guidance Materials

This PSF references the quality of all guidance materials expected to be available to the crew as they perform a particular action. The guidance includes any sources that might be used in accomplishing the action being modeled, such as procedures, databases, job performance aids, technical specifications, and the like. Assessments considered both form and content of the guidance items. Are they complete, well-written, clearly formatted, with a minimum of ambiguity? Are they written at a level appropriate to the expected level of knowledge of the users? Are they consistent, complete, and up to date?

The operators based their evaluation of this PSF on their familiarity with and use of the Vogtle guidance materials (e.g., normal, abnormal, and emergency procedures).

PSF 5: Characteristics of the Interface

This PSF assesses the quality of human engineering of both hardware and interactive software in the control room and in other locations, as well as the design of the actions themselves. Does the "interface" for the action include manual, local actions that require physically dispatching an operator to trip a breaker, install a jumper, or operate a valve? Is the task organized so that the same individual who detects a signal is also responsible for acting on it, or must this requirement be communicated? This PSF assesses the extent to which the sum of interface conditions help or hinder error-free accomplishment of the action.

The operators based their evaluation of this PSF on their familiarity with Vogtle in general, and with the Vogtle control room layout and interfaces in particular. Where appropriate, the operating crews factored into their evaluations the need for communications with operators in the auxiliary or turbine buildings, and other outside-control room interfaces particular to Vogtle.

PSF 6: Previous, Subsequent, and Concurrent Actions

This factor considered the effects of other actions that are not part of the operator action being evaluated, but may occur in close temporal, spatial, or logical proximity. Some of these actions may actually improve likelihood of success in the modeled action by leading the operator in a helpful direction. On the other hand, some "neighboring" tasks can lead operators in counterproductive directions. Of particular interest are those operator actions that may have input to the operator action being evaluated.

During the sessions, the operators relied upon their experience in performing a given action on the Vogtle simulator, as well as their knowledge of Vogtle procedures, for their assessment of the impact of previous, current, and subsequent additional actions.

PSF 7: Stress

This performance shaping factor is multidimensional. Some aspects and levels of stress may actually be beneficial from the standpoint of this analysis, by bringing the crew to a higher state of alertness, decreasing their likelihood of committing an error. Other aspects and degrees of stress can create a more error-likely situation. Elements of stress which may be taken into account include operators' perceptions of threat to themselves or their jobs; physical stressors such as noise, vibration, radiation, humidity, temperature, and light levels; belief by the crew that they do not understand what is going on (degree of surprise) and that they may not have adequate resources (including time) to deal satisfactorily with the event; and the nature of expected surveillance during the event by plant management, regulatory personnel, or others.

This was perhaps the most subjective of the PSFs, but again the operators based their assessments on their experiences at and familiarity with Vogtle.

The expert sessions were conducted by providing a copy of each operator action summary to each crew member and providing them time to read it through. Following this, questions, comments, improvements or corrections to the OA Summary were solicited. In a few cases, the OA Summary was modified on the spot to more accurately reflect reality for that OA at the plant prior to evaluating the OA.

Once everyone had agreed on the summary for the OA, the crew was then asked to provide two evaluations for each PSF on each OA. First, as a group, they were asked to rank, on a numeric scale from 1 to 10, as shown below, the importance of each PSF relative to the others for this OA.

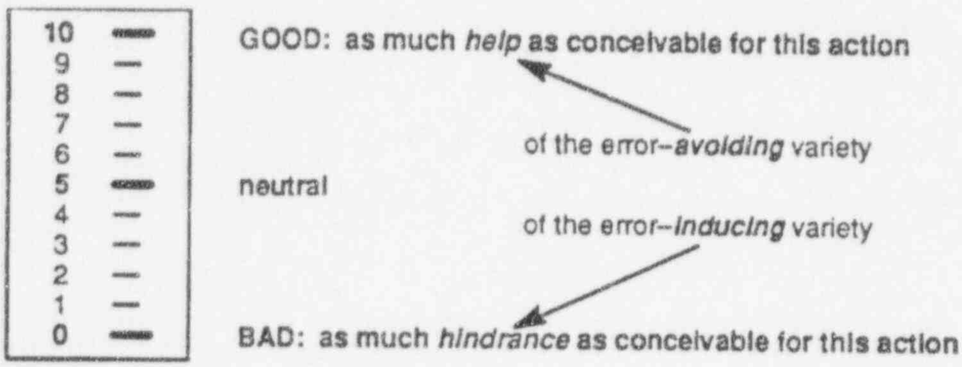
10	MAXIMUM CONCEIVABLE IMPORTANCE OF THE MOST IMPORTANT FACTOR(S) ON THIS OPERATOR ACTION
9	
8	
7	
6	
5	SOME IMPORTANCE
4	
3	
2	
1	
0	ABSOLUTELY *NO* IMPORTANCE

Importance was defined as the weight that should be given to the effect of this PSF on successful performance of the kind of task represented by the OA, without regard for how that OA happened to be implemented at this particular plant. A cited example was the importance of PSF 4, guidance and procedures, on a hypothetical OA that, as modeled for the IPE, requires immediate response (e.g., on the order of seconds). For successful performance of such an action, the importance of PSF 4 would be low: since there is no time to use procedures for such an action, their quality, whether helpful or harmful, would have little impact, or weight for this PSF on this OA. The same kind of OA could be used as a counter example for high importance on another PSF (e.g., PSF 3 - knowledge, training and experience) which would in general be expected to be important (although not necessarily most important).

Discussion of PSF importance was facilitated by the SLIM analyst who also "kept score" for the group, writing the importance scores on the blackboard, and changing them as necessary, as the group arrived at its successive decisions for all seven PSFs. The final scores were recorded on a data collection form. To facilitate the initial discussion, the group was asked to first determine the one or more PSFs that were most important, and give that (those) PSF(s) a score of 10. The importance of the remaining PSFs could then be estimated relative to this conceptual "anchor". For example, another PSF viewed as half as important as the "heaviest hitter" would receive a score of 5.

This consensus-building phase of the data collection served to get all members of the group thinking sensitively about the ways in which PSFs could relate to the OA in question. This had a "consciousness-raising" effect that was a helpful lead-in to the second phase of data collection., i.e., rating the effects of PSFs on the OA under discussion. In some cases, the weightings discussion led to additional refinement of OAs, i.e., the definition of variant OAs that were judged to be significantly different enough (from those that had been pre-defined by the IPE analysts) to require a separate evaluation in terms of both weights and ratings. In a very few cases, a prior distinction between two OAs (or variants of one) that had also been made by the IPE analysts, was judged to be insignificant, and the two categories would be collapsed into one for scoring purposes.

Following arrival at the group consensus on the seven PSF importance weights, the experts were then asked to individually provide a numerical rating of the plant-specific effects, of the same seven PSFs, on the same OA whose PSF importance had just been determined. They performed this task as individuals, with no discussion, using a second 10-point numeric scale as shown below to define the numeric scores that they wrote onto the scoring sheet.



Scorers were given as much time as needed to complete their answers. Following the completion of this scoring procedure, the group then moved on to the next OA, and iterated the procedure described in this section until all the OAs were evaluated.

As requested, data collected through this process is provided at the conclusion of this response. An explanation of the tables which contain the data is provided below. Copies of the evaluator's score sheets for operator action OCI are also provided as an example.

The calculation of the SLI is demonstrated in Figure 1. First, the calculated group average importance weights are entered as shown by the dotted-line boxes (1). These are the averages for the group of similar operator actions. Second, the group average importance weights from row (1) are normalized to 1.0, yielding the row of results in (2). Then the average effect ratings are calculated for each of the seven PSFs (3). The average effect ratings were divided by 10 to produce a value less than 1.0. (This is not a requirement of the methodology, and has no effect on the subsequent calculation of P_e). For clarity, the results of these last two sets of calculations are repeated as shown in the dashed-line box (4). Each pair of weights and ratings is then multiplied to produce the set of PSF scores (5). This is the multiplication step of the formula:

$$\mathrm{SLI} = \sum_{i=1}^{7} w_i r_i$$

Finally, the PSF scores are summed per the formula to produce a single SLI for this OA (6).

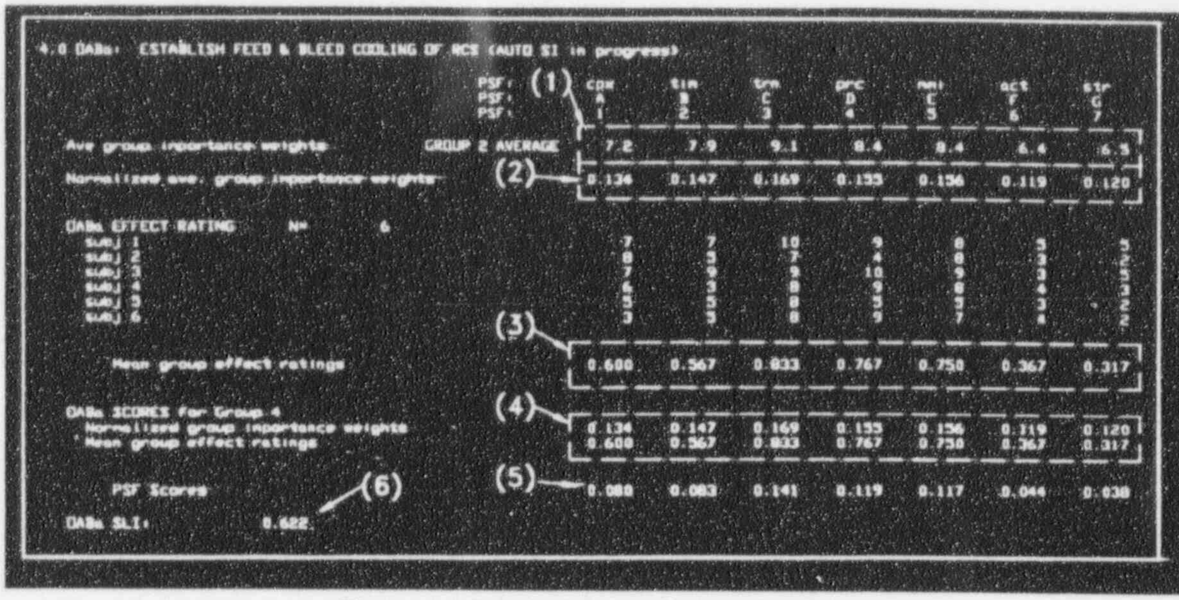


Figure 1 Tabulation for Calculating SLIs

Note that lines subj 1, subj 2, etc. are the ratings provided by the individual evaluators.

VEGP IPE HRA

OPERATOR ACTION SUMMARY

for

Action Nr: 14a PRT Variable: OMG

Action: Trip the Control Rod Drive Mechanism (CRDM) motor-generator (MG) sets

Applicable Event Tree(s): Anticipated Transient Without Trip (ATWT)

SUMMARY DESCRIPTION OF REQUIRED ACTION:

The primary goal behind this action is to attempt to limit peak RCS pressure to below the ASME Code Level C service limit in any transient where reactor trip is a normally expected plant response, but has not occurred. This action seeks to accomplish this by reducing the heat being generated from fission in the reactor, by causing the control rods to drop into the core.

The immediate objective of this action is to deenergize the supply of power from the Control Rod Drive Mechanisms' Motor/Generator (MG) sets to the CRDM rod-gripping magnets to allow the control rods to drop into the core. This action is accomplished by manually tripping the supply breakers to the buses which supply the MG sets. This action [OMG] is carried out only if the operator action to manually trip the reactor [ORT] has failed to insert control rods into the core.

CONTEXT - ACTIONS & EVENTS

Preceding:	Transient has occurred for which automatic reactor trip is expected but did not occur (see ATWT initiating events in E-0). Limiting ATWT initiating event is loss of MFW/loss of condenser vacuum with no reactor trip. Unsuccessful OA to trip reactor [ORT]
Concurrent:	Verify turbine trip and AFW actuation
Subsequent:	If this action is not successful: - OA to manually insert control rods [OCR] - OA to depressurize RCS (if necessary) [OAP], and to establish emergency boration [OBR])

SUCCESS/FAIL CRITERIA: Supply breakers tripped for Buse which supply CRDM MG sets

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VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 2 of 3)
for Action Nr: 142 PRT Variable: OMG

Action: Trip Control Rod Drive Mechanism (CRDM) Motor-Generator (MG) sets.

for event tree(s): ATWT

19000 - Done in this procedure now

APPLICABLE PROCEDURE(S): 19~~211~~-C, FR.S-1, "Response to Nuclear Power Generation/ATWT", Rev 4.

TIME WINDOW AVAILABLE TO INITIATE THIS ACTION: 2	FROM boundary condition	TO boundary condition
- ATWTa TBD h / ~~TBD~~ m	power < 40% and no Rx trip (ATWT PRT Tbl 4; ATWT SCNB p. TBD)	SG dryout
- ATWTb TBD h / ~~TBD~~ 6 m	power > 40%, MFW available for 5 min, and no Rx trip (ATWT PRT Tbl 4; ATWT SCNB p. TBD)	SG dryout
- ATWTc TBD h / ~~TBD~~ 1 m	power > 40%, no MFW, and no Rx trip (ATWT PRT Tbl 4; ATWT SCNB p. TBD) (This is the case being modeled in event tree.)	SG dryout

TIME WINDOW AVAILABLE TO COMPLETE THIS ACTION:	FROM boundary condition	TO boundary condition
1 to ~~10~~ 6 min per conditions cited above	Event initiation	SG dryout

MIN TIME WINDOW REQUIRED TO COMPLETE THIS ACTION:	___ min.

11/5/91

VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 3 of 3)
for Action Nr: 14a PRT Variable: OMG

Action: Trip Control Rod Drive Mechanism (CRDM) Motor-Generator (MG) sets.

for event tree(s): ATWT

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Procedure
Diagnose event	Rod bottom lights not lit -or- Reactor trip & bypass breakers closed -or- Neutron flux not decr asing	Rod bottom lights Rx trip & Bypass breaker indication Power Range nuclear instruments	ctrl rm	19000 19~~211~~-C ~~FR-S.1~~ s1
Trip CRDM MG set bus supply breaker		Supply feed breakers to 1NB08 and 1NB09: CB 1NB08-01 CB 1NB09-01	ctrl rm	19000 19~~211~~-C ~~FR-S.1~~ s1.1

New revis rno

11/5/91

27.0 OMG: TRIP CRDM MG SETS (OPEN NB08 & NB09 FEEDERS FORM MAIN CR)

	PSF:	cpx	tim	trn	prc	mmi	act	str	
	PSF:	A	B	C	D	E	F	G	
	PSF:	1	2	3	4	5	6	7	
Ave. group importance weights	GROUP 4 AVERAGE	3.4	8.2	8.4	6.0	8.4	5.4	8.3	
Normalized ave. group importance weights		0.071	0.170	0.175	0.124	0.175	0.113	0.172 =	1
OMG EFFECT RATINGS N= 6									
subj 1		8	7	10	10	9	7	6	
subj 2		7	4	6	7	7	5	3	
subj 3		7	4	7	8	8	5	4	
subj 4		7	5	9	9	9	4	4	
subj 5		9	1	9	7	7	8	1	
subj 6		5	9	10	10	7	5	6	
Mean group effect ratings		0.717	0.500	0.850	0.850	0.783	0.567	0.400	
OMG SCORES									
Normalized group importance weights		0.071	0.170	0.175	0.124	0.175	0.113	0.172	
Mean group effect ratings		0.717	0.500	0.850	0.850	0.783	0.567	0.400	
PSF Scores		0.051	0.085	0.149	0.106	0.137	0.064	0.069	

OMG SLI: 0.660

VEGP IPE HRA

OPERATOR ACTION SUMMARY

for

Action Nr: 3 PRT Variable: OAB

Action: Establish bleed and feed cooling of RCS

Applicable Event Tree(s):		
	SMALL LOCA (SLOCA)	SECONDARY BREAK (SSB)
	SG Tube Rupture (SGTR)	GENERAL TRANSIENTS
	AT WITHOUT TRIP (ATWT)	STATION BLACKOUT (SBO)

Should separate into automatic vs. manual SI initiation required.
(SI signal generated) (flip switch to start)

SUMMARY DESCRIPTION OF REQUIRED ACTION:

The primary goal of this action is to ensure adequate removal of core decay heat through continued covering of the core with cooled inventory, plus removal of heat that is transferred to the RCS. The intermediate goal is to establish an alternate path for removing core decay heat following loss of secondary heat sink (i.e., neither MFW nor AFW to the SGs).

The immediate objective of this top event is to establish bleed and feed cooling of the RCS following loss of secondary heat sink (i.e., loss of Main Feedwater (MFW) AND Auxiliary Feedwater (AFW) AND condenser/condensate to the SGs). This is accomplished by opening both Pressurizer (PZR) Power Operated Relief Valves (PORVs) to provide the bleed from the RCS, in coordination with controlled High Pressure SI (i.e., either CCPs or SIPs) to provide feed to the RCS. For all above listed event trees except General Transients, SI is assumed to have occurred. For General Transients, this action OAB also includes the alignments and setups required for starting SI. This action [all of OAB] must be accomplished before the SGs dry out in order to prevent core damage due to over temperature / pressure in the RCS.

If a loss of secondary heat sink occurs and (1) wide range level in any 3 SGs is less than 25% [40% for adverse containment] with no feedwater established or (2) pressurizer pressure is $\geq$ 2335 psig due to a loss of secondary heat sink, then the RCPs are tripped and bleed and feed is immediately initiated. Otherwise actions to establish main feedwater are performed.
CONTEXT - ACTIONS & EVENTS

Preceding:	loss of AFW loss of condenser/condensate loss of MFW
Concurrent:	bleed & feed path alignment with or without prior SI signal monitor containment pressure
Subsequent:	PZR fills, PORV relieves to the PZR relief tank, which ruptures to containment RWST depletes, requiring shift to cold leg recirc

VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 2 of 3)
for Action Nr: 3 PRT Variable: OAB

Action: Establish bleed and feed cooling to RCS

for event tree(s): SLOCA SGTR ATWT SSB SBO TRANSIENTS

SUCCESS/FAIL CRITERIA: alternate decay heat removal via PZR PORV(s) prior to SG dryout, balanced by inventory maintenance via HPSI

APPLICABLE PROCEDURE: 19231-C(FR-H.1), Response to Loss of Secondary Heat Sink, Rev. 11.
19200-C(F-0.3), Heat Sink Status Tree, Rev. 6.

TIME WINDOW AVAILABLE TO INITIATE THIS ACTION:	FROM boundary condition	TO boundary condition
- SLOCA ___ h / 20 m	event initiation (SLOCA PRTNB Tbl 4)	SG dryout
- SGTR ___ h / 20 m	event initiation (SGTR PRTNB Tbl 4)	SG dryout
- ATWT ___ h / 10 m	power < 40% AND AFW not available (ATWT PRTNB Tbl 4)	SG dryout
ATWT ___ h / ___ m	power > 40% AND MGs tripped and MFW available and AFW not available (ATWT PRTNB Tbl 4)	SG dryout
- SSB ___ h / 20 m	event initiation (SSB PRTNB Tbl 5)	SG dryout
- SBO ___ h / ___ m	power recovery ~~event initiation~~ (SBO PRTNB Tbl 4 - (tbd))	SG dryout
- TRNSIENT ___ h / 7 m	MFW pump trip (GT PRTNB Tbl 5)	SG dryout

TIME WINDOW AVAILABLE TO COMPLETE THIS ACTION:	FROM boundary condition	TO boundary condition
10 min	Initiation of recovery action	SG dryout

MIN TIME WINDOW REQUIRED TO COMPLETE THIS ACTION:	
30 min	for all except TRANSIENT, model assumes HPSI is on and operator only verifies its function. This may take 1 min. for TRANSIENT, model assumes HPSI requires manual start by operator. This adds __ m to time required

VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 3 of 3)
for Action Nr: 3 PRT Variable: OAB

Action: Establish bleed and feed cooling to RCS

for event tree(s): SLOCA SGTR ATWT SSB SBO TRANSIENTS

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Procdure
Recognition of loss of heat sink symptoms	NR level in all SGs < 5%(27%) AND total feedwater flow to SGs < 570 gpm			19200-C, F-0.3
Recognition of bleed and feed initiation symptoms	WR level in any 3 SGs < 25% (40%) with no MFW established OR PZR pressure > = 2335 psig			19231-C s 1.0, Second Caution
	trip all RCPs			19231-C s 1.0, Second Caution
Verify RCS feed path	Actuate SI	Operators quantified both automatic SI and manual SI actuation.		19231-C s10.0
	Verify or initiate HPSI			s11.0
	-at least 1 CCP or 1 SIP running			s11.a
	-ECCS valve alignment			s11.b
Establish RCS bleed path	-verify power to PZR PORV block valves			s12.a
	-verify PZR PORV block valves open			s12.b & s13.0
	-Open PZR PORVs			s12.c
	-Verify PZR PORVs open			s13.0
	-Monitor RWST level			s18.0 Caution

4.0 OABa: ESTABLISH FEED & BLEED COOLING OF RCS (AUTO SI in progress)

		cpx	tim	trn	prc	mmi	act	str	
	PSF:	cpx	tim	trn	prc	mmi	act	str	
	PSF:	A	B	C	D	E	F	G	
	PSF:	1	2	3	4	5	6	7	
Ave group importance weights	GROUP 2 AVERAGE	7.2	7.9	9.1	8.4	8.4	6.4	6.5	
Normalized ave. group importance weights		0.134	0.147	0.169	0.155	0.156	0.119	0.120	1
OABa EFFECT RATING N= 6									
subj 1		7	7	10	9	8	5	5	
subj 2		8	5	7	4	8	3	2	
subj 3		7	9	9	10	9	3	5	
subj 4		6	3	8	9	8	4	3	
subj 5		5	5	8	5	5	3	2	
subj 6		3	5	8	9	7	4	2	
Mean group effect ratings		0.600	0.567	0.833	0.767	0.750	0.367	0.317	
OABa SCORES									
Normalized group importance weights		0.134	0.147	0.169	0.155	0.156	0.119	0.120	
Mean group effect ratings		0.600	0.567	0.833	0.767	0.750	0.367	0.317	
PSF Scores		0.080	0.083	0.141	0.119	0.117	0.044	0.038	

OABa SLI: 0.622

5.0 OABb: ESTABLISH FEED & BLEED COOLING OF RCS (manual SI)

		PSF:	cpx	tim	trn	prc	mmi	act	str	
		PSF:	A	B	C	D	E	F	G	
		PSF:	1	2	3	4	5	6	7	
Ave. group importance weights		GROUP 2 AVERAGE	7.2	7.9	9.1	8.4	8.4	6.4	6.5	
Normalized ave. group importance weights			0.134	0.147	0.169	0.155	0.156	0.119	0.120 =	1
OABb EFFECT RATING	N= 6									
subj 1			6	6	10	10	9	7	3	
subj 2			4	5	6	3	5	3	2	
subj 3			4	6	10	10	8	3	5	
subj 4			5	3	8	9	8	5	3	
subj 5			1	2	9	8	9	3	1	
subj 6			3	5	8	7	3	2	2	
Mean group effect ratings			0.383	0.450	0.850	0.783	0.700	0.383	0.267	
OABb SCORES										
Normalized group importance weights			0.134	0.147	0.169	0.155	0.156	0.119	0.120	
Mean group effect ratings			0.383	0.450	0.850	0.783	0.700	0.383	0.267	
PSF Scores			0.051	0.066	0.144	0.122	0.109	0.046	0.032	

OABb SLI: 0.570

VEGP IPE HRA

OPERATOR ACTION SUMMARY

for

Action Nr: 17 PRT Variable: OCI

Action: Manually Isolate Containment

Applicable Event Tree(s): All Events, Station Blackout (SBO)

Separate SBO event since power is not available

SUMMARY DESCRIPTION OF REQUIRED ACTION:

The primary goal of this action is to limit any offsite dose should core damage occur as a result of the event.

The immediate objective of this action is to verify containment Phase A isolation, and if isolation has NOT occurred, to manually actuate containment Phase A isolation. This is accomplished by first checking the status of CI-A MLB (indicators correct for SI), and if not correct for SI, then actuating either of two switches on the main control board.

CONTEXT - ACTIONS & EVENTS

Preceding:	The conditions for safety injection/containment isolation signal have been met, but containment isolation has not occurred.
Concurrent:	
Subsequent:	Verify that either the containment isolation valves close or that the appropriate indicators on CI-A MLB are lit.

SUCCESS/FAIL CRITERIA:	Operator sucessfully identifies failure of containment isolation, and then successful manually actuates containment Phase A isolation
APPLICABLE PROCEDURE:	19000- C (E-0), Reactor Trip or Safety Injection, Rev. 9.

VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 2 of 2)
for Action Nr: 17 PRT Variable: OCI

Action: Manually Isolate Containment

Applicable Event Tree(s): All Events

TIME WINDOW AVAILABLE TO INITIATE THIS ACTION:	FROM boundary condition	TO boundary condition
h / m	Event initiation	prior to core exit T/Cs @ 1200 degF
TIME WINDOW AVAILABLE TO COMPLETE THIS ACTION:	**FROM boundary condition**	**TO boundary condition**
* 1.0 hour	Core damage	Vessel failure
MIN TIME WINDOW REQUIRED TO COMPLETE THIS ACTION:	Only the action of checking the CI-A MLB lights, then actuating one of the two containment isolation phase A switches is modeled. Total time for these actions may take 1 min.	

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Procedure
Isolate cntmt	Cntmt not isolated - manually actuate Phase A and CTMT ventilation isolation	CI-A MLB indicators not correct for SI; actuate CTMT ISO PHASE A	Ctrl rm	19000-C s7.0
SBO (w/o power) Isolate Containment			Local	19100-C attachment B, C & D

* Note: The appropriate time window is from event initiation to potential core damage, not post-core damage as indicated above. The time window considered during the operator evaluation was from event initiation to potential core damage.

32.0 OCIa: MANUALLY INITIATE CONTAINMENT ISOLATION (SBO)

		cpx	tim	trn	prc	mmi	act	str		
	PSF:	cpx	tim	trn	prc	mmi	act	str		
	PSF:	A	B	C	D	E	F	G		
	PSF:	1	2	3	4	5	6	7		
Ave. group importance weights	GROUP 3 AVERAGE	8.0	6.9	9.5	8.6	8.0	7.5	7.5		
Normalized ave. group importance weights		0.143	0.123	0.170	0.154	0.143	0.134	0.134	=	1
OCIa EFFECT RATING N= 6										
subj 1		4	7	10	9	9	5	5		
subj 2		3	5	4	6	5	3	4		
subj 3		4	5	8	8	8	5	5		
subj 4		3	5	7	8	7	3	3		
subj 5		2	4	8	8	7	6	3		
subj 6		5	5	8	9	6	3	4		
Mean group effect ratings		0.350	0.517	0.750	0.800	0.700	0.417	0.400		
OCIa SCORES										
Normalized group importance weights		0.143	0.123	0.170	0.154	0.143	0.134	0.134		
Mean group effect ratings		0.350	0.517	0.750	0.800	0.700	0.417	0.400		
PSF Scores		0.050	0.063	0.127	0.123	0.100	0.056	0.054		

OCIa SLI: 0.573

0C1

VEGP IPE Operator Action Evaluation Session 1 Evaluator Operator 1 Date 12/5/91

For Action Manually Isolate Cont (SBO)

OA IDENT		PERFORMANCE SHAPING FACTOR (PSF)						
OPERATOR ACTION IPE NAME	FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
		PSF IMPORTANCE WEIGHT: 10 — MAXIMUM conceivable importance for this operator action; 5 — some importance; 0 — ABSOLUTELY NO IMPORTANCE						
	SBO	8	5	10	9	8	4	7
	Power Available	4	~~8~~ 4	10	9	8	7	7
		PSF EFFECT RATING: 10 — GOOD: as much *help* (of the error-avoiding variety) as conceivable for this action; 5 — neutral; 0 — BAD: as much *hindrance* (of the error-inducing variety) as conceivable for this action						
	SBO	7	7	10	9	9	5	5
	Power Available	8	9	10	9	9	7	7

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VEGP IPE Operator Action Evaluation Session I Evaluator Operator 2 Date 12/5/91

For Action OCI - ISOLATE CNMT

OA IDENT		PERFORMANCE SHAPING FACTOR (PSF)						
OPERATOR ACTION IPE NAME	FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
		PSF IMPORTANCE WEIGHT: 10 MAXIMUM concelvable importance for this operator action; 5 some importance; 0 ABSOLUTELY NO IMPORTANCE						
	STATION BLACKOUT	8	5	10	9	8	4	7
	MANUAL WITH POWER	4	4	10	9	8	7	7
		PSF EFFECT RATING: 10 GOOD: as much help (of the error-avoiding variety) as conceivable for this action; 5 neutral; 0 BAD: as much hindrance (of the error-inducing variety) as concelvable for this action						
	STA BLACKOUT	3	5	4	6	5	3	4
	MANUAL WITH POWER	5	5	8	4	5	2	4

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OCI

VEGP IPE Operator Action Evaluation Session 1 Evaluator Operator 3 Date 12/5

For Action ~~SBO~~ Manually isolate CONT OCI

OA IDENT		PERFORMANCE SHAPING FACTOR (PSF)						
OPERATOR ACTION IPE NAME	FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
		PSF IMPORTANCE WEIGHT: 10 MAXIMUM conceivable importance for this operator action; 5 some importance; 0 ABSOLUTELY NO IMPORTANCE						
Manually No Power		8	5	~~8~~ 10	9	8	4	7
With Power		4	4	10	9	8	7	7
		PSF EFFECT RATING: 10 GOOD: as much *help* (of the error-avoiding variety) as conceivable for this action; 5 neutral; 0 BAD: as much *hindrance* (of the error-inducing variety) as conceivable for this action						
No Power		4	5	8	8	8	5	5
With Power		8	5	8	8	8	6	5

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OCI

VEGP IPE Operator Action Evaluation Session 1 Evaluator Operator 4 Date 12/5/91

For Action 17 OCI Manually Isolate Cnmt

OA IDENT		PERFORMANCE SHAPING FACTOR (PSF)						
OPERATOR ACTION IPE NAME	FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
		PSF IMPORTANCE WEIGHT: 10 MAXIMUM conceivable importance for this operator action; 5 some importance; 0 ABSOLUTELY NO IMPORTANCE						
	No power	8	5	10	9	8	4	7
	With Power	4	4	10	9	8	7	7
		PSF EFFECT RATING: 10 GOOD: as much help as conceivable for this action (of the error-avoiding variety); 5 neutral; 0 BAD: as much hindrance as conceivable for this action (of the error-inducing variety)						
	No Power	3	5	7	8	7	3	3
	With Power	6	6	8	8	8	4	4

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OCl

VEGP IPE Operator Action Evaluation Session 1 Evaluator Operator 5 Date 12-5-91

For Action OCl

PSF IMPORTANCE WEIGHT:
- 10 — MAXIMUM conceivable importance for this operator action
- 5 — some importance
- 0 — ABSOLUTELY NO IMPORTANCE

OA IDENT: OPERATOR ACTION IPE NAME	OA IDENT: FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
	LOSA	8	5	10	9	8	4	7
	Power	4	~~8~~ 4	10	9	8	7	7

PSF EFFECT RATING:
- 10 — GOOD: as much *help* as conceivable for this action (of the error-avoiding variety)
- 5 — neutral
- 0 — BAD: as much *hindrance* as conceivable for this action (of the error-inducing variety)

OPERATOR ACTION IPE NAME	FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
		2	4	8	8	7	6	3
		5	5	8	8	7	8	3

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OCI

VEGP IPE Operator Action Evaluation Session__________ Evaluator_ Operator 6 Date__________

For Action ______ O C I ______

OA IDENT		PERFORMANCE SHAPING FACTOR (PSF)						
OPERATOR ACTION IPE NAME	FOR INITATING EVENT	1 TASK COMPLEXITY	2 TIME FACTORS	3 KNOWLEDGE, TRAINING, & EXPERIENCE	4 GUIDANCE & PROCEDURES	5 PLANT INTERFACE: CONTROLS & INDICATIONS	6 PRECEDING, CONCURRENT, & SUBSEQUENT ACTIONS	7 STRESS
		PSF IMPORTANCE WEIGHT: 10 MAXIMUM conceivable importance for this operator action; 5 some importance; 0 ABSOLUTELY NO IMPORTANCE						
	Blackout	8	5	10	9	8	4	7
	w/power	4	4 ~~5~~ [illegible]	10	9	8	7	7
		PSF EFFECT RATING: 10 GOOD: as much *help* [of the error-avoiding variety] as conceivable for this action; 5 neutral; 0 BAD: as much *hindrance* [of the error-inducing variety] as conceivable for this action						
		5	5	8	9	6	3	4
		5	5	8	9	6	3	4

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VEGP IPE HRA

OPERATOR ACTION SUMMARY

for

Action Nr: 16 PRT Variable: ORS

Action: Restore systems following loss of offsite power/station blackout

Applicable Event Tree(s): Station Blackout (SBO)

SUMMARY DESCRIPTION OF REQUIRED ACTION:

The primary goal and immediate objective of this action is to restore power to and operation of essential systems after at least 1 AC emergency bus has been energized. The operator would perform the following actions after AC power is restored:

Restore DC loads
Energize 480V AC switchgear
Energize battery charges, instrumentation and control, emergency lighting, communications, and battery room fans
Verify NSCW operation
Reset Phase A (if actuated)
Verify instrument air available
Start ACCW and CCW pumps
Align reactor makeup system
Start CCP or CCP and SIP
Align for either normal charging or ECCS injection*
Start containment fan coolers
Start RHR pump (with SI required)
Establish RCP seal cooling

* Note, leakage from the RCP seals during the station blackout event could eventually cause the pressurizer to empty and the RCS to reach saturation. As a result, SI may also be required after power is restored.

CONTEXT - ACTIONS & EVENTS

Preceding:	Loss of all AC power Defeated autostart of safeguards equipment TD-AFW controlled to maintain SG levels DC loads minimized Begin SG depressurization SI reset AC power restored RCP seals isolated
Concurrent:	
Subsequent:	Control SG narrow range level and pressure Control pressurizer level and pressure Verify natural circulation, adequate shutdown margin

for Action Nr: 16 PRT Variable: ORS

Action: Restore systems following loss of offsite power/station blackout

for event tree(s): SBO

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Procdure
Restore DC loads previously shed. Align deenergized inverters, per 13431, prior to closing DC feeder breakers		See EOP 19100-C, Attachment A		19100-C s23.0
Verify equip. loaded on energized AC emergency bus		-480 V AC switchgear: Unit 1 TRAIN A / TRAIN B 1AB04 / 1BB06 1AB05 / 1BB07 1AB15 / 1BB16 1NB01 / 1NB10 Unit 2 TRAIN A / TRAIN B 2AB04 / 2BB06 2AB05 / 2BB07 2AB15 / 2BB16 2NB01 / 2NB10 Essential 480V AC loads: o Batter chargers o Instrumentation and control o Emergency lighting o Communications o Battery room fans.		s25.0
Verify NSCW operation		- Verify valves open: TRAIN A / TRAIN B HV-1806 / HV-1807 HV-1808 / HV-1809 HV-1822 / HV-1823 HV-1830 / HV-1831		s26.a
		- 2 NSCW pumps running on each of 2 trains		s26.b & 19101-C s3.0

for Action Nr: 16 PRT Variable: ORS

Action: Restore systems following loss of offsite power/station blackout

for event tree(s): SBO

SUCCESS/FAIL CRITERIA: The operator must establish essential systems within __ minutes after AC power is restored.

APPLICABLE PROCEDURES: 19100-C (ECA-0.0), Loss of All AC Power, Rev. 5

19101-C (ECA-0.1), Loss of All AC Power Recovery Without SI Required, Rev. 8

19102-C (ECA-0.2), Loss of All AC Power Recovery With SI Required, Rev. 4

TIME WINDOW AVAILABLE TO INITIATE THIS ACTION:	FROM boundary condition	TO boundary condition
- SBO ____ h / ____ m	AC Emergency Bus Energized (SBO PRT NB Tbl 4)	Potential core damage
TIME WINDOW AVAILABLE TO COMPLETE THIS ACTION:	**FROM boundary condition**	**TO boundary condition**
30 minutes	AC Emergency Bus Energized	Potential core damage
MIN TIME WINDOW REQUIRED TO COMPLETE THIS ACTION:	__ min.	

Action: Restore systems following loss of offsite power/station blackout

for event tree(s): SBO

TASK ELEMENTS - continued

Subtask	Step	Equip't, MMI d/c	Location	Procdure
Restore Phase A (Inst. Air)		- If Phase A actuated and inst. air pressure normal, then check HV-9378 open - If Phase A actuated and inst. air pressure not normal, then start air compressor per 13710, and (when inst. air pressure normal) check HV-9378 open		19101-C s2.0
Start an ACCW pump	(without SI)			19101-C s3.b
Start one ACCW & 2 CCW pumps	(with SI req.)			19102-C s7.a& s7.c
Align makeup source	(without SI)	- CCP Suction valves from VCT: LV-0112B open LV-0112C open - VCT makeup control system set for greater than RCS boron concentration and automatic control - Charging line isolation valves: HV-8105 shut HV-8106 shut - CCP normal miniflow isolation valves-open		19101-C s3.C.1

for Action Nr: 16 PRT Variable: ORS

Action: Restore systems following loss of offsite power/station blackout

for event tree(s): SBO

TASK ELEMENTS - continued

Subtask	Step	Equip't, MMI d/c	Location	Procdure
Align makeup source (continued)	(with SI req.)	- RWST level > 39%		19102-C s1.0
		- CCP suction from RWST valve-open		s3.a
		- CCP suction from VCT valve-open		s3.b
		- RCP seal injection isolation valves-shut		s4.b
Start CCP	(without SI)			19101-C s3.C.2
Start CCP & SIP	(with SI req.)			19102-C s2.0 & s4.C
Align CCP flow through BIT	(with SI req.)	- CCP alternate miniflow isolation valves - open		s5.a
		- CCP normal miniflow isolation valves - shut		s5.b
		- BIT isolation valves-open		s5.c
		- Charging line isolation valves: HV-8105 shut HV-8106 shut		s5.d
Start safe-guards equip.	(without SI)	- Start CTMT fan coolers		19101-C s3.d
	(with SI req.)	- Start RHR pump - Start CTMT fan coolers		19102-C s7.b s7.d

110791/KGD

Action: Restore systems following loss of offsite power/station blackout

for event tree(s): SBO

TASK ELEMENTS - continued

Subtask	Step	Equip't, MMI d/c	Location	Procdure
Establish charging flow	(without SI)	- HC-0182 - set to maximum seal flow HV-0182 shut		19101-C s4.a
		- Charging line isolation valves: HV-8105 open HV-8106 open		s4.b
		- Establish charging flow using control valves: FV-0121 HV-0182		s4.c

31.0 ORS: RESTORE SYSTEMS FOLLOWING LOSP (WITHOUT SI REQUIRED, SI REQUIRED)

		cpx	tim	trn	prc	mmi	act	str		
	PSF:	cpx	tim	trn	prc	mmi	act	str		
	PSF:	A	B	C	D	E	F	G		
	PSF:	1	2	3	4	5	6	7		
Ave. group importance weights	GROUP 3 AVERAGE	8.0	6.9	9.5	8.6	8.0	7.5	7.5		
Normalized ave. group importance weights		0.143	0.123	0.170	0.154	0.143	0.134	0.134	=	1
ORS EFFECT RATINGS N= 6										
subj 1		4	7	8	10	9	6	6		
subj 2		3	5	6	5	6	4	4		
subj 3		3	4	7	9	9	4	5		
subj 4		2	3	7	9	8	3	3		
subj 5		2	2	8	7	6	6	4		
subj 6		6	6	7	9	9	5	4		
Mean group effect ratings		0.333	0.450	0.717	0.817	0.783	0.467	0.433		
ORS SCORES										
Normalized group importance weights		0.143	0.123	0.170	0.154	0.143	0.134	0.134		
Mean group effect ratings		0.333	0.450	0.717	0.817	0.783	0.467	0.433		
PSF Scores		0.048	0.055	0.122	0.126	0.112	0.063	0.058		

ORS SLI: 0.583

VEGP IPE HRA

OPERATOR ACTION SUMMARY

for

Action Nr: 5a PRT Variable: OARa

Action: Realign ECCS low pressure system for cold leg recirculation

Applicable Event Tree(s): SMALL LOCA (SLOCA)
MEDIUM LOCA (MLOCA)
LARGE LOCA (LLOCA)
SG Tube Rupture (SGTR)

SUMMARY DESCRIPTION OF REQUIRED ACTION:

For all of the above events, the primary goal is to provide a source of cooled water to the RCS to ensure the core remains covered and that adequate removal of decay heat continues after SI depletes the RWST. In support of that goal, the immediate objective of this action is to establish cold leg recirculation by realigning the RHR pumps to take suction from the containment sump, cool the fluid using the RHR HX to transfer energy to the CCW loop in the RHR HX, and discharge the cooled fluid from the RHR HX into the RCS cold legs.

Note that RHR HXs discharge is also opened to provide suction to the CCPs and SIPs, although for this action, success on the high-pressure part of the procedure is not modeled. The rationale for discharging RHR HXs to both low and high pressure paths to the RCS is that as pressure in the RCS drops below the low pressure recirculation shutoff value, this flow path with its larger pumping capacity will automatically begin to add inventory to the RCS. It is assumed that RHR pumps are already on, drawing suction from the RWST, and discharging either to miniflow or to the RCS cold legs.

For this operator action, the containment sump level is verified to be sufficiently full, RHR pumps are verified to be running and suction is aligned from the RWST to the sump. CCW to the RHR HXs is verified (at least 2 CCW pumps running per train), and the CCW pump discharge pressures and flows and NSCW cooling to the CCW heat exchangers are verified.

CONTEXT - ACTIONS & EVENTS

Preceding:	SI has occurred: ECCS Injection occurring RHR pumps are on, injecting or on miniflow RWST level < 39 percent 1 to 8 Containment Cooling Units removing heat 1 or 2 spray pumps drawing on RWST
Concurrent:	align ECCS high pressure for cold leg recirc [OARb] requires success in this action as a subset
Subsequent:	Align for hot leg recirculation at 11 hours for medium and large LOCAs

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VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 2 of 4)
for Action Nr: 5a PRT Variable: OARa

for event tree(s): SLOCA MLOCA LLOCA SGTR

Action: Realign ECCS low pressure system for cold leg recirculation

SUCCESS/FAIL CRITERIA: ECCS low pressure cold leg recirculation

APPLICABLE PROCEDURES: 19010-C, E-1, "Loss of Reactor or Secondary Coolant," Revision 10.
19013-C, ES-1.3, "Transfer to Cold Leg Recirculation," Revision 7.

TIME WINDOW AVAILABLE to INITIATE THIS ACTION:		FROM boundary condition	TO boundary condition
- LLOCA ~~&~~ ~~MLOCA~~	h / 24 m	RWST low-low level alarm with containment spray operation	complete realignment
- MLOCA	h / 130 m	RWST low-low level alarm, no containment spray operation	complete realignment
- SLOCA & SGTR	h / 130 m	RWST low-low level alarm, no containment spray operation	complete realignment

TIME WINDOW AVAILABLE TO COMPLETE THIS ACTION:	FROM boundary condition	TO boundary condition
(as above)		

MIN TIME WINDOW REQUIRED TO COMPLETE THIS ACTION:	

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VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 3 of 4)
for Action Nr: 52 PRT Variable: OARa

for event tree(s): SLOCA MLOCA LLOCA SGTR

Action: Realign ECCS low pressure system for cold leg recirculation

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Prcdre
verify cold leg recirc capability	power to: RHRP suction RHR pump RHRP discharge RHR HX operable	 HV 8811A/B RHR pump A/B HV 8809A/B	ctrl rm	19010-C E-1 step 12
RWST < 39 %				E-1 s14 & foldout page
Reset SI				19013-C ES-1.3 s1
verify CCW for RHR HX	two CCW pumps per train			ES-1.3 s2.a
	discharge press & flow			s2.b
	NSCW cooling to CCW HX	2 NSCW pumps/train 4 NSCT fans/train		s2.c
align ECCS for CL recirc	RHR pumps running			ES-1.3 s3.a
	RHR HL suc shut	HV-8701A/B HV-8702A/B		s3.b
	RHRP sump suc valves open	HV-8811A/B		s3.c.2
	RHR disch valves shut	HV-8716A/B		s3.c.5
	close RHR RWST valve	HV-8812A/B		s3.f s3.h

VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 4 of 4)
for Action Nr: 52 PRT Variable: OARa

for event tree(s): SLOCA MLOCA LLOCA SGTR

Action: Realign ECCS low pressure system for cold leg recirculation

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Prcdre
verify flow paths for RHR pumps			ctrl rm	19013-C ES-1.3
	CNMT sump level > 5in [23 in]	LI-764 LI-765		s4.a s5.a
	RHR pumps running			s4.b s5.b
	RHR isolation valves open	HV 8809A/B		s4.c s5.c
	RHR HX flow > 500 gpm	FI-618A FI-619A		s4.d s5.d

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8.0 OARa: ALIGN ECCS LOW PRESSURE SYSTEM FOR COLD LEG RECIRCULATION

	PSF:	cpx	tim	trn	prc	mmi	act	str	
	PSF:	A	B	C	D	E	F	G	
	PSF:	1	2	3	4	5	6	7	
Ave. group importance weights	GROUP 1 AVERAGE	4.6	6.1	9.7	9.1	7.1	4.3	5.1	
Normalized ave. group importance weights		0.099	0.133	0.211	0.198	0.155	0.093	0.111 =	1
OARa EFFECT RATING N= 6									
subj 1		8	7	10	10	7	7	7	
subj 2		5	4	8	8	7	5	5	
subj 3		5	5	7	9	7	5	5	
subj 4		7	7	8	9	8	5	5	
subj 5		6	6	7	8	7	5	5	
subj 6		6	5	8	7	4	5	4	
Mean group effect ratings		0.617	0.567	0.800	0.850	0.667	0.533	0.517	
OARa SCORES									
Normalized group importance weights		0.099	0.133	0.211	0.198	0.155	0.093	0.111	
Mean group effect ratings		0.617	0.567	0.800	0.850	0.667	0.533	0.517	
PSF Scores		0.061	0.075	0.168	0.168	0.103	0.050	0.058	

OARa SLI: 0.684

VEGP IPE HRA

OPERATOR ACTION SUMMARY

for

Action Nr: 12 PRT Variable: OAS

Action: Establish containment spray recirculation

Applicable Event Tree(s): Large LOCA (LLOCA) Medium LOCA (MLOCA)
Small LOCA (SLOCA) SG Tube Rupture (SGTR)
AT without Trip (ATWT) Secondary Side Break (SSB)
General Transients

May not have to establish spray recirc. for second break.

SUMMARY DESCRIPTION OF REQUIRED ACTION:

The primary goal behind this action is to maintain containment integrity to prevent/mitigate escape of fission products to the environment. Intermediate supporting goals are to avoid breaching containment integrity through overpressurization due to hydrogen combustion or steam concentration. Other intermediate goals of this action are to provide fission product scrubbing within containment, and to provide cooling for the containment atmosphere and control sump pH.

The immediate objective of this action is to realign the spray suction from the RWST to the containment sump in time to prevent cavitation of the spray pumps.

CONTEXT - ACTIONS & EVENTS

Preceding: Containment spray setpoint (Hi-3 cont. pressure (21.5 psig)) reached
Spray pumps running and drawing from the RWST
HHSI &/or LHSI aligned for cold leg recirculation
RWST Empty (9%) level alarm setpoint reached

Concurrent:

Subsequent: Trip spray pumps when containment pressure < 15 psig

SUCCESS/FAIL CRITERIA: At least one containment spray pump must be running and aligned to take suction from the containment sump.

APPLICABLE PROCEDURE(S): 19013-C, (ES-1.3) "Transfer to Cold Leg Recirculation", Rev. 7

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VEGP IPE HRA - OPERATOR ACTION SUMMARY (p. 2 of 3)
for Action Nr: 12 PRT Variable: OAS

Action: Establish containment spray recirculation

for event tree(s): LLOCA MLOCA SLOCA SGTR ATWT SSB TRANSIENTS

TIME WINDOW AVAILABLE TO INITIATE THIS ACTION:	FROM boundary condition	TO boundary condition
- LLOCA 0.5 h / ~~180~~ m	Event initiation	RWST Empty level
- MLOCA 0.5 h / ~~180~~ m	Event initiation (without fan coolers)	RWST Empty level
- SLOCA 1.0 h / ~~180~~ m	Event initiation (without AFW and fan coolers)	RWST Empty level
- SGTR 1.5 h / ~~180~~ m	Event initiation (without AFW and fan coolers)	RWST Empty level
- ATWT 1.0 h / ~~180~~ m	Event initiation (with a consequential LOCA)	RWST Empty level
- SSB 1.5 h / ~~180~~ m	Event initiation (with a break inside containment)	RWST Empty level
- TRANS 1.5 h / ~~180~~ m	Event initiation (with a consequential LOCA)	RWST Empty level
TIME WINDOW AVAILABLE TO COMPLETE THIS ACTION:	**FROM boundary condition**	**TO boundary condition**
- ABOVE LIST ____ h / ~~180~~ 10 m	RWST Empty level alarm	RWST empty
MIN TIME WINDOW REQUIRED TO COMPLETE THIS ACTION:	__ min. (check with valve close/open times)	

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for Action Nr: 12 PRT Variable: OAS

Action: Establish containment spray recirculation

for event tree(s): LLOCA MLOCA SLOCA SGTR ATWT SSB TRANSIENTS

TASK ELEMENTS

Subtask	Step	Equip't, MMI d/c	Location	Procedure
Verify RWST Empty level (9%)			ctrl rm	19013-C (ES-1.3) s8
Reset Cont. Spray				s10.a
Establish* communication				
Shut spray add tank iso valve closed		HV-8894A closed HV-8894B closed	ctrl rm	s10.b
Open CTMT spray pump sump isolat. valves		HV-9002A open HV-9003A open HV-9002B open HV-9003B open	ctrl rm	s10.c s10.f
Close CTMT spray pump RWST isolat. valves		HV-9017A closed HV-9017B closed		s10.d s10.g
Verify continued satisfactory operation*		PI-0972 > 7 psig PI-0974 > 190 psig PI-0973 > 7 psig PI-0975 > 190 psig		s10.e s10.f

*Note: Satisfactory containment spray pump operation after isolating the spray additive tank is verified by local observation of the containment spray pump suction and discharge pressure gauges. Communication with an operator stationed locally must be established prior to realigning containment spray for recirculation.

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24.0 OAS: ESTABLISH CONTAINMENT SPRAY RECIRCULATION

		cpx	tim	trn	prc	mmi	act	str		
	PSF:	cpx	tim	trn	prc	mmi	act	str		
	PSF:	A	B	C	D	E	F	G		
	PSF:	1	2	3	4	5	6	7		
Ave. group importance weights	GROUP 1 AVERAGE	4.6	6.1	9.7	9.1	7.1	4.3	5.1	**	
Normalized ave. group importance weights		0.099	0.133	0.211	0.198	0.155	0.093	0.111	=	1
OAS EFFECT RATINGS N= 6										
subj 1		10	7	10	10	10	8	6		
subj 2		8	5	8	5	7	5	4		
subj 3		8	5	8	8	9	6	5		
subj 4		10	5	9	8	9	6	5		
subj 5		8	3	9	8	7	7	8		
subj 6		9	5	8	9	10	5	4		
Mean group effect ratings		0.883	0.500	0.867	0.800	0.867	0.617	0.533		
OAS SCORES										
Normalized group importance weights		0.099	0.133	0.211	0.198	0.155	0.093	0.111		
Mean group effect ratings		0.883	0.500	0.867	0.800	0.867	0.617	0.533		
PSF Scores		0.088	0.067	0.182	0.159	0.134	0.057	0.059		

OAS SLI: 0.746

Question 7

The technique for human error rate prediction (THERP) is identified in the submittal as the method used to model recovery actions and the limited number of pre-initiator human errors. However, no details are provided in the submittal as to the specific human error types, data, and PSFs that were selected to quantify these errors. For instance, no examples of THERP event trees are provided in the submittal to indicate the level of detail of modeling using THERP. Please provide a discussion and example of the process used to quantify pre-initiator and post-initiator recovery-action human error events. Please illustrate this discussion with THERP event trees and data for each of the following human errors:

- 206-EX1
- OLP-MLB
- RIAHXB
- CBHV-BOPD
- OFC1

Response 7

Application of THERP in the Vogtle IPE included the modeling of operator actions according to the associated hardware success criteria and incorporation of recovery factors where sufficient amounts of slack time exist for the tasks. These considerations provided a refined plant-specific HRA model of Vogtle plant. THERP event trees were not used in modeling the Vogtle human errors because incorporating the slack time recovery and the different combinations for the hardware success criteria becomes extremely cumbersome. By modeling the operator actions with the LOTUS spreadsheet program, the essence of the THERP event tree is captured along with the applications of slack time recovery and hardware success criteria.

The details in modeling and quantifying the operator actions 206-EX1, CBHV-BOPD, OLP-MLB and RIAHXB are provided in the following excerpts from the Vogtle HRA notebook. Operator action OFC1 is a dependent event and is included in the details on dependency evaluation provided in the response to Question 9.

3.6.2 OPERATOR ACTION: 1CWXV206-EX1

RESTORE VALVE U4-206 AFTER TEST

During performance of ESF Chiller Pump and Discharge Check Valves Inservice Test, the operator is required to restore valve U4-206 to full open position.

The test procedure is considered to be a long list and valve U4-206 does not have control room indication for immediate detection of its misposition.

The restoration of valve U4-207 is completely independent of restoration of valve U4-206. That is, the test on valve U4-206 is completed and this valve is restored, then testing and restoration of valve U4-207 is conducted.

Reference Procedure [Step]

VEGP 14809-1, Rev. 6, ESF Chiller Pump and Discharge Check Valves Inservice Test [5.1.11.1]

The applicable procedure steps are provided as a markup in Appendix D

Subtask(s)

1. Open ESF chiller pump discharge valve 1-1592-U4-206 to full open position.

Note(s):

1. It is assumed that an independent verification is conducted on the re-alignment of components in this test. This verification is modeled as recovery.

2. For this equipment restoration activity, commission error is not credible; therefore, only error of omission is modeled.

TABLE 3.6.2a
FAILURE MODEL
RESTORE VALVE U4-206 AFTER IN-SERVICE TEST
1CWXV206-EX1

N	Description	QNd	QNo	QNc	QNm	QNr	Notes
I.	ACTION (with use of procedure)						
1.	Failure to restore valve U-206 to open position; (14809-1; step 5.1.11.1)		Q1o		C1m	Q1r	(see notes 2, 4, 5)
II.	ACTION (without use of procedure)						
2.	Failure to restore valve U-206 to open position; (14809-1; step 5.1.11.1)		Q2o		Q1m	Q1r	(see notes 2, 4, 5)

Calculated Parameter	Formula
C1m	(1-Q1m)

Notes

1. QNd - Initial operator response during diagnosis
2. QNo - Errors of Omission; where CNo indicates formula for multiple components
3. QNc - Errors of Commission; where CNc indicates formula for multiple components
4. QNm - Failure to use procedure; where CNm = 1-QNm
5. QNr - Unproceduralized checking; where CNr indicates formula for unproceduralized checking multiplied by proceduralized recovery checking or verification

where:

"N" (in general) corresponds to the row number of the described step; except in cases where an HEP is repeated.

TABLE 3.6.2b
DATA FOR QUANTIFICATION
RESTORE VALVE U4-206 AFTER IN-SERVICE TEST
1CWXV206-EX1

Failure	Description	Nominal Probabilities Mean	Nominal Probabilities Variance	Mult Factor	* HEP	** Data Source	Comment
Q1m	Use written procedures during normal operating condition	1.30E-02	8.80E-05	1.000	1.30E-02	8	3
Q1o	Omission (use procedure with checkoff) long List >10 items	3.80E-03	7.90E-06	1.000	3.80E-03	15	3
Q1r	Checking routine tasks; checker using written materials	1.60E-01	4.20E-02	1.000	1.60E-01	31	3
Q2o	Omission when available written procedures are not used	8.10E-02	1.00E-02	1.000	8.10E-02	18	3
C1m	(1-Q1m)				9.87E-01		

* -- HEP is equal to Nominal Mean times Multiplicative Factor
** -- Data is taken from Appendix C, table C-2. Item numbers from table C-2 are quoted as source of data.

Comments

1. Due to the assumed operating crew experience, it is believed that failure to diagnose the event by not responding to the appropriate alarm(s) is less than nominal; thus, a multiplicative factor of 0.1 is applied.

2. Commission errors are believed to be less than nominal due to operator experience and proper labeling of equipment and controls; thus, a multiplicative factor of 0.1 is applied.

3. Low stress level is assumed; a multiplier of 1.0 is applied, (Reference 6, Table 20-16).

4. Moderate stress level is assumed; a multiplier of 2.0 is applied, (Reference 6, Table 20-16).

5. High stress level is assumed; a multiplier of 5.0 is applied, (Reference 6, Table 20-16).

6. Unproceduralized checking by 2 people; a multiplier of 0.1 is applied to item 33 of DATA SOURCE.

7. Step is not proceduralized; a multiplier of 2.0 is applied.

8. Slack time recovery is applied: medium dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + 6N) / 7, where N = 8.1E-02. Thus, a multiplier of 0.21 is applied.

9. Slack time recovery is applied; high dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + N) / 2, where N = 8.1E-02. Thus, a multiplier of 0.54 is applied.

TABLE 3.6.2c
QUANTIFICATION OF
RESTORE VALVE U4-206 AFTER IN-SERVICE TEST
1CWXV206-EX1

Index	Formula	Calculation	Resu
I. ACTION (with use of procedure)			
1.	Q1r*C1m*(Q1o)	1.60E-01 * 9.87E-01 * (3.80E-03	
6.00E-04			
II. ACTION (without use of procedure)			
2.	Q1r*Q1m*(Q2o)	1.60E-01 * 1.30E-02 * (8.10E-02	
1.68E-04			
	Total		7.69E-04

3.6.15 OPERATOR ACTION: CBHVAC-SBO

OPEN INVERTER ROOM DOORS ON LOSS OF ALL AC

During station blackout, the control building HVAC system is unavailable to cool the DC bus and inverter rooms. The operator is required to open the doors to the inverter rooms so that DC power will be available to run the turbine driven AFW pump.

Success is defined as opening 4 of 4 inverter room doors within 60 minutes; the actual time to complete the actions is estimated to be about 10 minutes.

Reference Procedure [Step]

VEGP 19000-C, Rev. 9; E-0 Reactor Trip or Safety Injection [3]
VEGP 19100-C, Rev. 9; ECA-0.0 Loss of ALL AC Power [14c]

The applicable procedure steps are provided as a markup in Appendix D

Subtask(s)

1. Recognize no power to AC emergency busses

2. Open inverter room doors

Note(s):

1. It is assumed that the operators are trained in performing this task.

2. High stress level PSF is applied because of the station blackout accident scenario.

3. The entire operating crew (including shift supervisor and shift technical advisor) is assumed to be present and slack time is believed to exist. Therefore, unproceduralized checking recovery is applied to all steps.

TABLE 3.6.15a
FAILURE MODEL
OPEN INVERTER ROOM DOORS ON LOSS OF ALL AC
CBHVAC-SBO

N	Description	QNd	QNo	QNc	QNm	QNr	Notes
I.	DIAGNOSIS						
1.	Failure to recognize no power to AC emergency busses; (19000-C, step 3)		Q1o			Q1r	(see notes 2, 5)
II.	ACTION						
2.	Failure to open 1 of 4 inverter room doors; (19100-C, step 14c)		C2o			Q2r	(see note 2)

Calculated Parameter	Formula
C2o	Q2o*4

Notes

1. QNd - Initial operator response during diagnosis
2. QNo - Errors of Omission; where CNo indicates formula for multiple components
3. QNc - Errors of Commission; where CNc indicates formula for multiple components
4. QNm - Failure to use procedure; where CNm = 1-QNm
5. QNr - Unproceduralized checking; where CNr indicates formula for unproceduralized checking multiplied by proceduralized recovery checking or verification

where:

"N" (in general) corresponds to the row number of the described step; except in cases where an HEP is repeated.

TABLE 3.6.15b
DATA FOR QUANTIFICATION
OPEN INVERTER ROOM DOORS ON LOSS OF ALL AC
CBHVAC-SBO

Failure	Description	Nominal Probabilities Mean	Nominal Probabilities Variance	Mult Factor	* HEP	** Data Source	Comment
Q1o	Omission (use procedure with checkoff) Short List <=10 items	1.30E-03	8.80E-07	5.000	6.50E-03	14	5
Q1r	Special short-term, one-of-a-kind checking with alert factors	8.10E-02	1.00E-02	0.500	4.05E-02	33	5,6
Q2o	Omission (use procedure with checkoff) Short List <=10 items	1.30E-03	8.80E-07	5.000	6.50E-03	14	5
Q2r	Checking routine tasks; checker using written materials	1.60E-01	4.20E-02	5.000	8.00E-01	31	5
C2o	Q2o*4				2.60E-02		

* -- HEP is equal to Nominal Mean times Multiplicative Factor
** -- Data is taken from Appendix C, table C-2. Item numbers from table C-2 are quoted as source of data.

Comments

1. Due to the assumed operating crew experience, it is believed that failure to diagnose the event by not responding to the appropriate alarm(s) is less than nominal; thus, a multiplicative factor of 0.1 is applied.

2. Commission errors are believed to be less than nominal due to operator experience and proper labelling of equipment and controls; thus, a multiplicative factor of 0.1 is applied.

3. Low stress level is assumed; a multiplier of 1.0 is applied. (Reference 6, Table 20-16).

4. Moderate stress level is assumed; a multiplier of 2.0 is applied. (Reference 6, Table 20-16).

5. High stress level is assumed; a multiplier of 5.0 is applied. (Reference 6, Table 20-16).

6. Unproceduralized checking by 2 people; a multiplier of 0.1 is applied to item 33 of DATA SOURCE.

7. Step is not proceduralized; a multiplier of 2.0 is applied.

8. Slack time recovery is applied; medium dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + 6N) / 7, where N = 8.1E-02. Thus, a multiplier of 0.21 is applied.

9. Slack time recovery is applied; high dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + N) / 2, where N = 8.1E-02. Thus, a multiplier of 0.54 is applied.

TABLE 3.6.15c
QUANTIFICATION OF
OPEN INVERTER ROOM DOORS ON LOSS OF ALL AC
CBHVAC-SBO

Index	Formula	Calculation	Result
I. DIAGNOSIS			
1.	Q1r*(Q1o)	4.05E-02 * (6.50E-03)	2.63E-04
II. ACTION			
2.	Q2r*(C2o)	8.00E-01 * (2.60E-02)	2.08E-02
	Total		2.11E-02

3.6.7.1 OPERATOR ACTION: OLP

STOP RHR PUMPS (WITHIN 30 MINUTES) DURING SLOCA OR MLOCA

The operator is required to prevent operating RHR pumps on miniflow for longer than 30 minutes, during an accident, if the RCS pressure exceeds 300 psig. Stopping the RHR pumps is necessary to protect them, in case there is inadequate component cooling water to the pumps or heat exchangers. The operator is also required to monitor the RCS pressure, and restart the RHR pumps if the RCS pressure falls below 300 psig.

Success is defined as stopping both RHR pumps within 30 minutes from the accident initiation. Restarting at least 1 of 2 RHR pumps, when required, is sufficient to supply water to the RCS. The actual time to complete this task is estimated at 5 minutes.

Reference Procedure [Step]

VEGP 19010-C, Rev. 7, E-1 Loss of Reactor or Secondary Coolant [9a,b,c; CAUTION]

The applicable procedure steps are provided as a markup in Appendix D

Subtask(s)

1. Recognize RCS pressure > 300 psig
2. Reset SI
3. Stop 2 RHR pumps
4. Start RHR pumps when RCS pressure < 300 psig

Note(s):

1. It is assumed that the operators are trained in performing this task.
2. High stress level PSF is applied to all steps in this event since this task is performed in the early part of the LOCA accident; stress level is believed to be moderate or low much later in the LOCAs sequences.
3. The entire operating crew (including shift supervisor and shift technical advisor) is assumed to be present and slack time is believed to exist. Therefore, unproceduralized checking recovery is applied to all steps.

TABLE 3.6.7.1a
FAILURE MODEL
STOP RHR PUMPS (WITHIN 30 MINUTES) DURING SLOCA OR MLOCA
OLP

N	Description	QNd	QNo	QNc	QNm	QNr	Notes
I.	DIAGNOSIS						
1.	Failure to realize RCS pressure greater than 300 psig; (VEGP 19010-C; [9a.1])		Q1o	Q1c		Q1r	(see notes 2, 3, 5)
II.	ACTION						
2.	Failure to reset SI; (VEGP 19010; [9b])		Q2o	Q2c		Q1r	(see notes 2, 3, 5)
3.	Failure to stop 1 of 2 RHR pumps; (VEGP 19010; [9c])		C3o	C3c		Q1r	(see notes 2, 3, 5)
4.	Failure to realize RCS pressure lowers to < 300 psig; (VEGP 19010-C; [CAUTION])		Q4o	Q4c		Q1r	(see notes 2, 3, 5)
5.	Failure to restart 2 of 2 RHR pumps; (VEGP 19010; [CAUTION])		C5o	C5c		Q1r	(see notes 2, 3, 5)

Calculated Parameter	Formula
C3o	Q3o*2
C3c	Q3c*2
C5o	Q5o*0.15
C5c	Q5c*0.15

Notes

1. QNd - Initial operator response during diagnosis

2. QNo - Errors of Omission; where CNo indicates formula for multiple components

3. QNc - Errors of Commission; where CNc indicates formula for multiple components

4. QNm - Failure to use procedure; where CNm = 1-QNm

5. QNr - Unproceduralized checking; where CNr indicates formula for unproceduralized checking multiplied by proceduralized recovery checking or verification

where:

"N" (in general) corresponds to the row number of the described step; except in cases where an HEP is repeated.

TABLE 3.6.7.1b
DATA FOR QUANTIFICATION
STOP RHR PUMPS (WITHIN 30 MINUTES) DURING SLOCA OR MLOCA
OLP

Failure	Description	Nominal Probabilities Mean	Nominal Probabilities Variance	Mult Factor	* HEP	** Data Source	Comment
Q1c	Misread display on Digital Readout (<= 4 digits)	1.20E-03	8.80E-07	0.500	6.00E-04	52	2,5
Q1o	Omission (use procedure with checkoff) long List >10 items	3.80E-03	7.90E-06	5.000	1.90E-02	15	5
Q1r	Special short-term, one-of-a-kind checking with alert factors	8.10E-02	1.00E-02	0.500	4.05E-02	33	5,6
Q2c	Select wrong control from panel with clearly drawn mimic lines	1.30E-03	1.10E-05	0.500	6.50E-04	22	2,5
Q2o	Omission (use procedure with checkoff) long List >10 items	3.80E-03	7.90E-06	5.000	1.90E-02	15	5
Q3c	Select wrong control from panel with clearly drawn mimic lines	1.30E-03	1.10E-05	0.500	6.50E-04	22	2,5
Q3o	Omission (use procedure with checkoff) long List >10 items	3.80E-03	7.90E-06	5.000	1.90E-02	15	5
Q4c	Misread display on Digital Readout (<= 4 digits)	1.20E-03	8.80E-07	0.500	6.00E-04	52	2,5
Q4o	Omission (use procedure with checkoff) long List >10 items	3.80E-03	7.90E-06	5.000	1.90E-02	15	5
Q5c	Select wrong control from panel with clearly drawn mimic lines	1.30E-03	1.10E-05	0.500	6.50E-04	22	2,5
Q5o	Omission (use procedure with checkoff) long List >10 items	3.80E-03	7.90E-06	5.000	1.90E-02	15	5
C3o	Q3o*2				3.80E-02		
C3c	Q3c*2				1.30E-03		
C5o	Q5o*0.15				2.85E-03		
C5c	Q5c*0.15				9.75E-05		

* -- HEP is equal to Nominal Mean times Multiplicative Factor
** -- Data is taken from Appendix C, table C-2. Item numbers from table C-2 are quoted as source of data.

Comments

1. Due to the assumed operating crew experience, it is believed that failure to diagnose the event by not responding to the appropriate alarm(s) is less than nominal; thus, a multiplicative factor of 0.1 is applied.

2. Commission errors are believed to be less than nominal due to operator experience and proper labeling of equipment and controls; thus, a multiplicative factor of 0.1 is applied.

3. Low stress level is assumed; a multiplier of 1.0 is applied. (Reference 6, Table 20-16).

4. Moderate stress level is assumed; a multiplier of 2.0 is applied. (Reference 6, Table 20-16).

5. High stress level is assumed; a multiplier of 5.0 is applied. (Reference 6, Table 20-16).

6. Unproceduralized checking by 2 people; a multiplier of 0.1 is applied to item 33 of DATA SOURCE.

7. Step is not proceduralized; a multiplier of 2.0 is applied.

8. Slack time recovery is applied; medium dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + 6N) / 7, where N = 8.1E-02. Thus, a multiplier of 0.21 is applied.

9. Slack time recovery is applied; high dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + N) / 2, where N = 8.1E-02. Thus, a multiplier of 0.54 is applied.

TABLE 3.6.7.1c
QUANTIFICATION OF
STOP RHR PUMPS (WITHIN 30 MINUTES) DURING SLOCA OR MLOCA
OLP

Index	Formula	Calculation	Result
I. DIAGNOSIS			
1.	Q1r*(Q1o+Q1c)	4.05E-02 * (1.90E-02 + 6.00E-04)	7.94E-04
II. ACTION			
2.	Q1r*(Q2o+Q2c)	4.05E-02 * (1.90E-02 + 6.50E-04)	7.96E-04
3.	Q1r*(C3o+C3c)	4.05E-02 * (3.80E-02 + 1.30E-03)	1.59E-03
4.	Q1r*(Q4o+Q4c)	4.05E-02 * (1.90E-02 + 6.00E-04)	7.94E-04
5.	Q1r*(C5o+C5c)	4.05E-02 * (2.85E-03 + 9.75E-05)	1.19E-04
	Total		4.09E-03

3.6.16 OPERATOR ACTION: 1DCHURGXFMRIAHXB

TRANSFER 120V AC TO REGULATED TRANSFORMERS

Loss of two 120V AC panels, 1AY1A AND 1BY1B, causes a reactor trip and failure of power to the SSPS cabinets. Assuming that failure of the two 120V AC panels is due to the power supplies and not the panels themselves, the operator is required tc transfer the panels to the regulated transformers.

Success is defined as transferring 2 of 2 120V AC panels to the regulated transformers within 30 minutes; the actual time to complete the actions is estimated to be 16 minutes.

Reference Procedure [Step]

VEGP 18032-1, Rev. 4; Loss of 120V AC Instrument Power [A13, C13]
VEGP 13431-1, Rev. 6; 120V AC 1E Vital Instrument Distribution System [4.2.1.1; 3; 4]

The applicable procedure steps are provided as a markup in Appendix D

Subtask(s)

1. Respond to 1AY1A & 1AY1B panel alarms
2. Dispatch operator to perform local transfer
3. Ensure regulated transformer breaker closed
4. Open AC breaker from transformer
5. Close AC breaker from regulator transformer source

Note(s):

1. It is assumed that the operators are trained in performing this task.
2. High stress level PSF is applied because of the urgency of the accident scenario.
3. The entire operating crew (including shift supervisor and shift technical advisor) is assumed to be present and slack time is believed to exist. Therefore, unproceduralized checking recovery is applied to all steps.

TABLE 3.6.16a
FAILURE MODEL
TRANSFER 120VAC TO REGULATED TRANSFORMERS
1DCHURGXFMRIAHXB

N	Description	QNd	QNo	QNc	QNm	QNr	Notes
I.	DIAGNOSIS						
1.	Failure to respond to 1 of 2 alarms for failed 1AY1A & B panels	Q1d				Q1r	(see notes 1, 5)
II.	ACTION						
2.	Failure to dispatch operator to transfer 1AY1A or B; (18032-1, steps A13, C13)		C2o			Q1r	(see notes 2, 5)
3.	Failure to ensure Reg. Trans. bkr closed; (13431-1, step 4.2.1.1)		C3o	C3c		Q2r	(see notes 2, 3, 5)
4.	Failure to open instru. dist. panel AC bkr from Trans; (13431-1, step 4.2.1.3)		C4o	C4c		Q2r	(see notes 2, 3, 5)
5.	Failure to close instru. dist. panel AC bkr from Reg. source; (13431-1, step 4.2.1.4)		C5o	C5c		Q2r	(see notes 2, 3, 5)

Calculated Parameter	Formula
C2o	Q2o*2
C3o	Q3o*2
C3c	Q3c*2
C4o	Q4o*2
C4c	Q4c*2
C5o	Q5o*2
C5c	Q5c*2

Notes

1. QNd - Initial operator response during diagnosis

2. QNo - Errors of Omission; where CNo indicates formula for multiple components

3. QNc - Errors of Commission; where CNc indicates formula for multiple components

4. QNm - Failure to use procedure; where CNm = 1-QNm

5. QNr - Unproceduralized checking; where CNr indicates formula for unproceduralized checking multiplied by procedural and recovery checking or verification

where:

"N" (in general) corresponds to the row number of the described step; except in cases where an HEP is repeated.

TABLE 3.6.16b
DATA FOR QUANTIFICATION
TRANSFER 120VAC TO REGULATED TRANSFORMERS
1DCHURGXFMRIAHXB

Failure	Description	Nominal Probabilities Mean	Nominal Probabilities Variance	Mult Factor	* HEP	** Data Source	Comment
Q1d	Respond to 1 of N alarms with 2 annunciator alarming	1.60E-03	1.60E-05	0.500	8.00E-04	40	1,5
Q1r	Special short-term, one-of-a-kind checking with alert factors	8.10E-02	1.00E-02	0.500	4.05E-02	33	5,6
Q2o	Omission (use procedure with checkoff) Short List <=10 items	1.30E-03	8.80E-07	5.000	6.50E-03	14	5
Q2r	Checking routine tasks; checker using written materials	1.60E-01	4.20E-02	5.000	8.00E-01	31	5
Q3c	Select wrong circuit breaker in a group of circuit breakers	6.20E-03	2.20E-05	0.500	3.10E-03	29	2,5
Q3o	Omission (use procedure with checkoff) Short List <=10 items	1.30E-03	8.80E-07	5.000	6.50E-03	14	5
Q4c	Select wrong circuit breaker in a group of circuit breakers	6.20E-03	2.20E-05	0.500	3.10E-03	29	2,5
Q4o	Omission (use procedure with checkoff) Short List <=10 items	1.30E-03	8.80E-07	5.000	6.50E-03	14	5
Q5c	Select wrong circuit breaker in a group of circuit breakers	6.20E-03	2.20E-05	0.500	3.10E-03	29	2,5
Q5o	Omission (use procedure with checkoff) Short List <=10 items	1.30E-03	8.80E-07	5.000	6.50E-03	14	5
C2o	Q2o*2				1.30E-02		
C3o	Q3o*2				1.30E-02		
C3c	Q3c*2				6.20E-03		
C4o	Q4o*2				1.30E-02		
C4c	Q4c*2				6.20E-03		
C5o	Q5o*2				1.30E-02		
C5c	Q5c*2				6.20E-03		

* -- HEP is equal to Nominal Mean times Multiplicative Factor

** -- Data is taken from Appendix C, table C-2. Item numbers from table C-2 are quoted as source of data.

Comments

1. Due to the assumed operating crew experience, it is believed that failure to diagnose the event by not responding to the appropriate alarm(s) is less than nominal; thus, a multiplicative factor of 0.1 is applied.

2. Commission errors are believed to be less than nominal due to operator experience and proper labeling of equipment and controls; thus, a multiplicative factor of 0.1 is applied.

3. Low stress level is assumed; a multiplier of 1.0 is applied. (Reference 6, Table 20-16).

4. Moderate stress level is assumed; a multiplier of 2.0 is applied. (Reference 6, Table 20-16).

5. High stress level is assumed; a multiplier of 5.0 is applied. (Reference 6, Table 20-16).

6. Unproceduralized checking by 2 people; a multiplier of 0.1 is applied to item 33 of DATA SOURCE.

7. Step is not proceduralized; a multiplier of 2.0 is applied.

8. Slack time recovery is applied; medium dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + 6N) / 7, where N = 8.1E-02. Thus, a multiplier of 0.21 is applied.

9. Slack time recovery is applied; high dependency is applied to the HEP for unproceduralized checking; this is evaluated as: (1 + N) / 2, where N = 8.1E-02. Thus, a multiplier of 0.54 is applied.

TABLE 3.6.16c
QUANTIFICATION OF
TRANSFER 120VAC TO REGULATED TRANSFORMERS
1DCHURGXFMRIAHXB

Index	Formula	Calculation	Result
I. DIAGNOSIS			
1.	Q1r*(Q1d)	4.05E-02 * (8.00E-04)	3.24E-05
II. ACTION			
2.	Q1r*(C2o)	4.05E-02 * (1.30E-02)	5.26E-04
3.	Q2r*(C3o+C3c)	8.00E-01 * (1.30E-02 + 6.20E-03)	1.54E-02
4.	Q2r*(C4o+C4c)	8.00E-01 * (1.30E-02 + 6.20E-03)	1.54E-02
5.	Q2r*(C5o+C5c)	8.00E-01 * (1.30E-02 + 6.20E-03)	1.54E-02
	Total		4.66E-02

Question 8

In applying PSFs the consideration of time is important. The submittal is not clear on how "time factors" (available time and required time) were calculated and incorporated in the analysis of the various response- and recovery-type post-initiator human events. Please provide the information requested in 8a and b below for the following human actions:

- OATa	- OFC1
- OATb	- OFC2
- OATc	

a. The available time estimated for the operator action and the bases for the time chosen. Include in your discussion how different available times were calculated for the same task but different sequences.

b. The required time estimated for the operator action and the process used to determine the time required. For example, simulator observations, walk-down inspection of procedures, operator interviews, and so on, could be used to measure or estimate the time necessary for the operator (s) to complete the action.

Response 8a. and 8b.

Time windows (that is, the available time) for post-initiator human actions were generally based on event sequence timing from the IPE success criteria analyses. Times required to perform response-type actions were initially estimated by the HRA and IPE analysts, and subsequently reviewed by SNC personnel familiar with Vogtle operations. Those actions quantified using the SLIM method (e.g., OATa, OATb, OATc) were then presented to the Vogtle operating crews during the SLIM expert sessions. The operators were asked to comment on the timing specified for each action assessed, and where there were any discrepancies between the listed timing and the operators' stated experience for the actions, the operators evaluated the timing performance shaping factor based on their experience and training (e.g., simulator exercises, available job performance measure information, and so forth). For the SLIM assessment, the operators evaluated the procedural steps required and the time available to accomplish a given action, and assigned an appropriate weighting and ranking to the timing performance shaping factor; a specific time to complete was not assessed. If the operators considered the available time to be sufficient to accomplish the task, an appropriate ranking was assigned; however, if the operators considered the available time to be insufficient, credit would not have been taken for the action as defined.

Recovery-type actions were identified and defined in conjunction with the Independent Review Group reviewers and Vogtle operations personnel. The times required to perform such actions were based on input from these operations personnel using, where available, specific timing data from job performance measure (JPM) exercises.

For the specific actions mentioned in this question, the pertinent information is as follows.

Actions OATa, OATb, and OATc are response-type actions quantified using the SLIM method. These are variations of the action to terminate safety injection following different events, each involving somewhat different procedural steps and event-specific timing. The timing information was obtained from the detailed event-specific thermal/hydraulic analyses performed to justify the Vogtle IPE success criteria. Additional information regarding the IPE success criteria analysis methodology is provided in Section 3.1.3 of the Vogtle IPE Report.

Action OATa is defined as the action to terminate safety injection following a secondary side break in order to avoid overfilling the pressurizer. The time window (available time) for this action, per the Vogtle IPE success criteria analysis for secondary side break events, is the time from event initiation to the time the pressurizer would overfill, and is approximately 10 minutes. Action OATb is defined as the action to terminate safety injection in order to avoid overfilling the steam generator, and transfer to normal charging and letdown following a steam generator tube rupture. The available time for this action, per the Vogtle IPE success criteria analysis for steam generator tube rupture events, is the time from successful isolation of the ruptured steam generator to the time the steam generator would overfill, and is approximately 10 minutes. Action OATc is defined as the action to terminate safety injection and transfer to normal charging following a small LOCA. The available time for this action, per the Vogtle IPE success criteria analysis for small LOCA events, is the time from event initiation to the time that normal RHR can be aligned, and is approximately 3 hours. As discussed earlier, since these actions were quantified using SLIM, required action times were not defined, but were implicitly considered by the operators in their assessment of the viability of the task and of the timing performance shaping factor.

Actions OFC1 and OFC2 are recovery-type actions quantified using the THERP method. These both represent local control of the turbine-driven AFW pump using the trip/throttle valve following a station blackout and loss of DC power (either upon depletion of the station batteries or following their failure due to loss of room cooling), but evaluated under two different sets of conditions. Action OFC2 applies to the condition where the batteries become depleted. The time window (available time) for this action is 20 minutes, per the Vogtle IPE success criteria analysis for station blackout. The required time is 12 minutes, per a Vogtle Control Room Operator JPM for this action. Action OFC1 is a version assumed in scenarios also involving failure of the operators to open the inverter room doors, resulting in loss of DC power. Because of the possibility that indications from instrumentation might not be available if equipment has failed due to loss of room cooling, the human error probability for OFC1 was quantified using a high dependency relationship, OFC1 = (1+OFC2)/2. Thus, it was assumed that the action could still be performed under these conditions, but that the operators might need more time and might be under more stress than would be the case if room cooling were available. This resulted in an HEP of 0.515 (i.e., very little credit) for OFC1.

Question 9

It is not clear from the submittal how dependencies were addressed and treated in the post-initiator human reliability analysis (HRA). The performance of the operator is both dependent on the accident in progress and the past performance of the operator during the accident of concern. Improper treatment of these dependencies can result in the elimination of potentially dominant accident sequences and, therefore, the identification of significant events. Please provide a concise discussion and examples illustrating how dependencies were addressed and treated in the post-initiator HRA such that important accident sequences were not eliminated. The discussion should address the following:

a. The 12 human error events modeled as dependent (listed on page 3-124 of the submittal) were identified as being modeled using 5 PSFs (stress in prior event, and time window, slack time, complexity, and type of procedural guidance for the second event). However, definitions and bases for these factors are not described in the submittal. P' ase provide definitions of these factors and the basis for their use in modeling dependent events.

b. Quantification values associated with these factors are not described in the submittal. Please provide descriptions of how these factors were used in quantifying the 12 dependent events.

Response 9a. and 9b.

The dependency evaluation covers positive dependency between events whereby failure on the first task increases the probability of failure on the second task. The evaluation does not cover negative dependency which implies that failure on the first task reduces the probability of failure on the second task; application of negative dependency produces results that may not be realistic.

Dependencies are evaluated by the equations provided by THERP (NUREG/CR-1278, Tables 20-17 and 20-18).

The dependency modeling is addressed as conditional probabilities based on the following set of criteria:

a) Dependencies in manipulating 2 or more of the same type of component, by the same operator in the same procedure step are modeled as follows:

1. Failure to operate 2 of 2 controls (e.g., failure to start 2 of 2 pumps) is modeled with the second action having a low dependency of the first action. The model will reflect base human error probability (BHEP) x 0.05. However, we have applied moderate dependency which results in BHEP x 0.15.

   If the operator manipulates both controls together, then complete dependency is assumed; that is, if one control is missed, the other is missed also.

2. Failure to operate 3 of 3 controls is modeled with the second action having a low dependency of the first action, and the third action having a moderate dependency on the previous actions. The model will reflect BHEP x 0.05 x 0.15.

3. Failure to operate N of N controls (N > or = 4) is modeled with the second action having a low dependency of the first action, the third action having a moderate dependency on the previous actions, and fourth and subsequent actions (each) having a high dependency on previous actions. The model will reflect BHEP x 0.05 x 0.15 x 0.5 x ... x 0.5. In general, we have assigned one high dependency value (0.5) for all fourth and subsequent actions. Therefore, the joint conditional probability, for N > than 4, is evaluated by BHEP x 0.05 x 0.15 x 0.5.

4. Failure to operate M of N controls (2 < M < N) is modeled by applying the appropriate dependency level (shown in 1, 2 or 3 above) based on the value of M. The binomial coefficient of "M out of N" shows up in this evaluation. For example, failure to operate 2 of 4 controls will reflect BHEP x 0.15 x 6.

b) In selecting the critical subtasks for an operator action, each step of the applicable procedure(s) must be examined to determine its significance relative to system success. In most cases, subtasks that are recovery actions of, or redundant to, other previous subtasks are screened out because failure of those subtasks are judged to be dependent on failure of the previous subtasks; total dependency relation between such actions is assumed. This is particularly true in the selection of omission errors; depending on the structure of the procedure, if a step in one column of the procedure is missed, then recovery steps provided in the alternate path will most likely be missed.

   Some operator actions that involve verification of component status or system parameter are screened out because of total dependency relationship between the verification step and a previous step, or because the effects of not performing such verification could be realized in a subsequent step that is judged to be critical.

   In some scenarios, redundant subtasks are modeled using the dependency technique described in (a) above. An example of this is the actions in performing depressurization using the condenser (steam dumps) or using the atmospheric relief valves; the second option is modeled as being dependent on the first.

c) Dependencies between different (top) events are evaluated, based on factors such as: time window, slack time, complexity of tasks, and type of procedural guidance available.

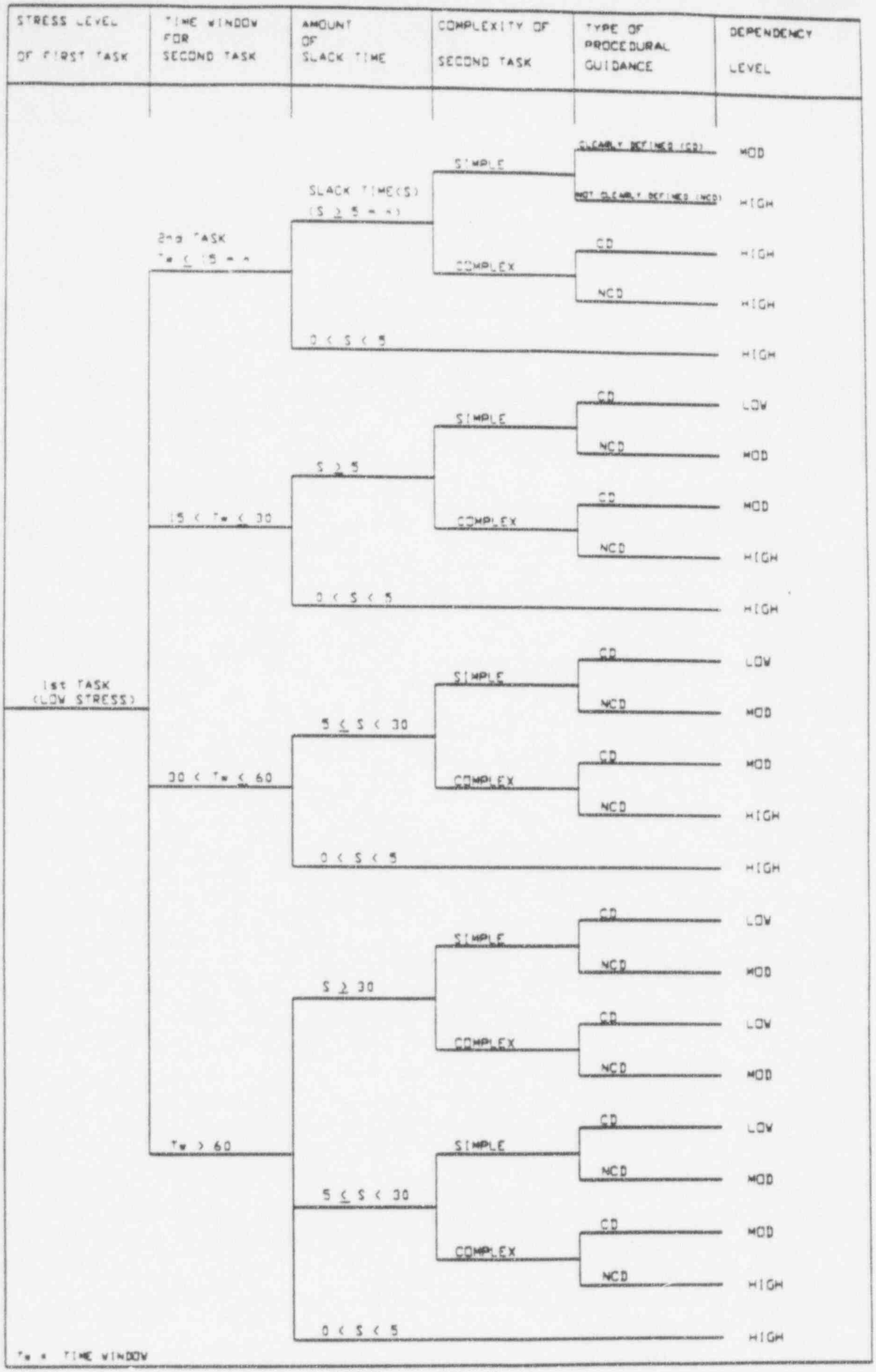
For this type of conditional probability evaluation, the analysis is performed using the event tree provided in Figure 1. The appropriate event tree path, for a given operator action, is agreed upon by the cognizant system analysts and HRA analyst.

The starting point in Figure 1 is to determine the stress level of the first (or preceding) task. If low stress level was used for the first task, then use sheet 1 of 3; if moderate stress was used for the first task, use sheet 2; and if high stress, use sheet 3.

The exercise continues with the aim of determining factors specific to the second task such as time window, slack time, complexity of the tasks (taking into account workload), and the type of procedural guidance. The end result is the deducing of the dependency level for the second task.

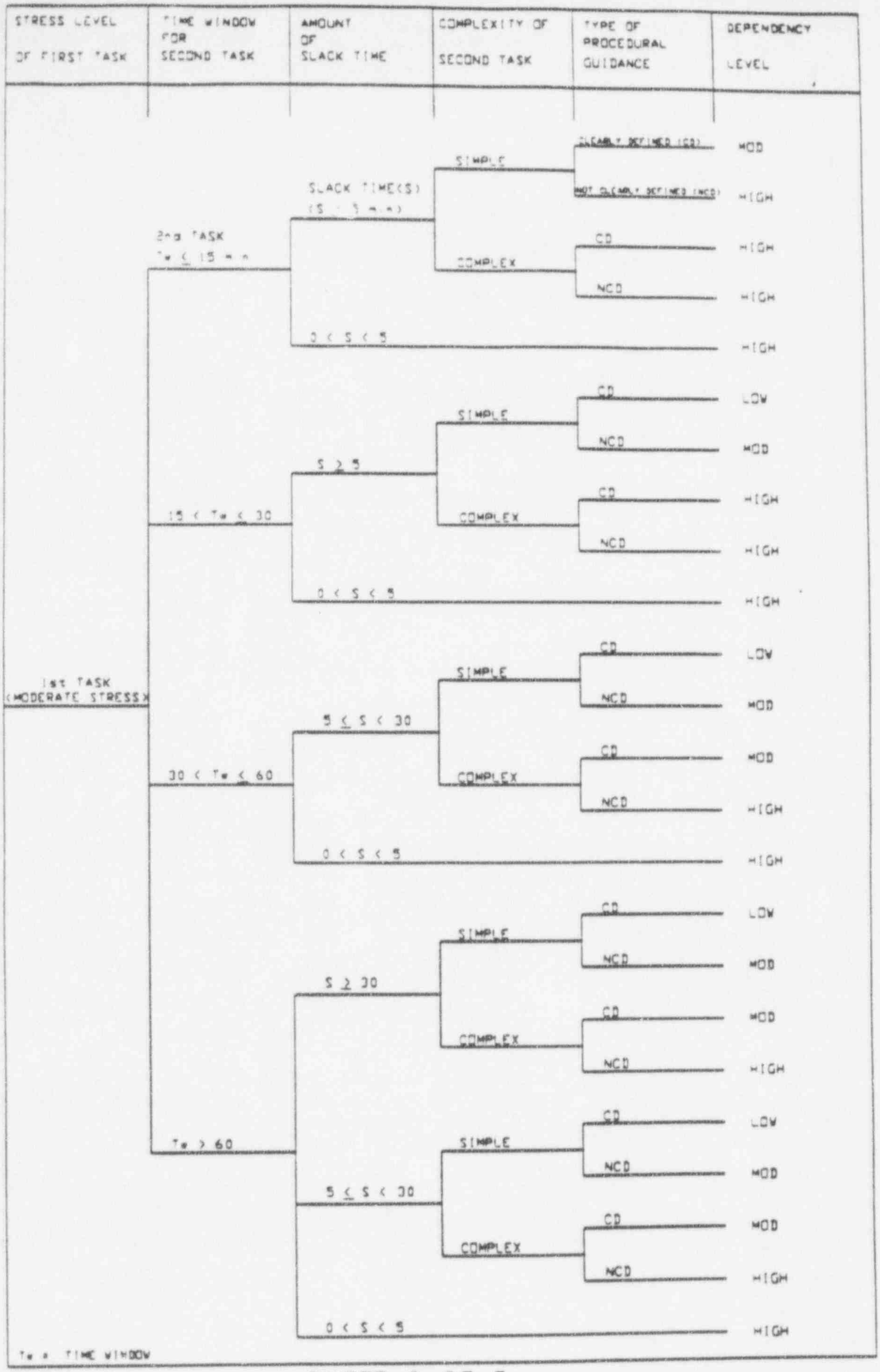
Conditional Probabilities are then documented in Table 1 which summarizes and captures the factors considered in Figure 1 for each task.

FIGURE 1



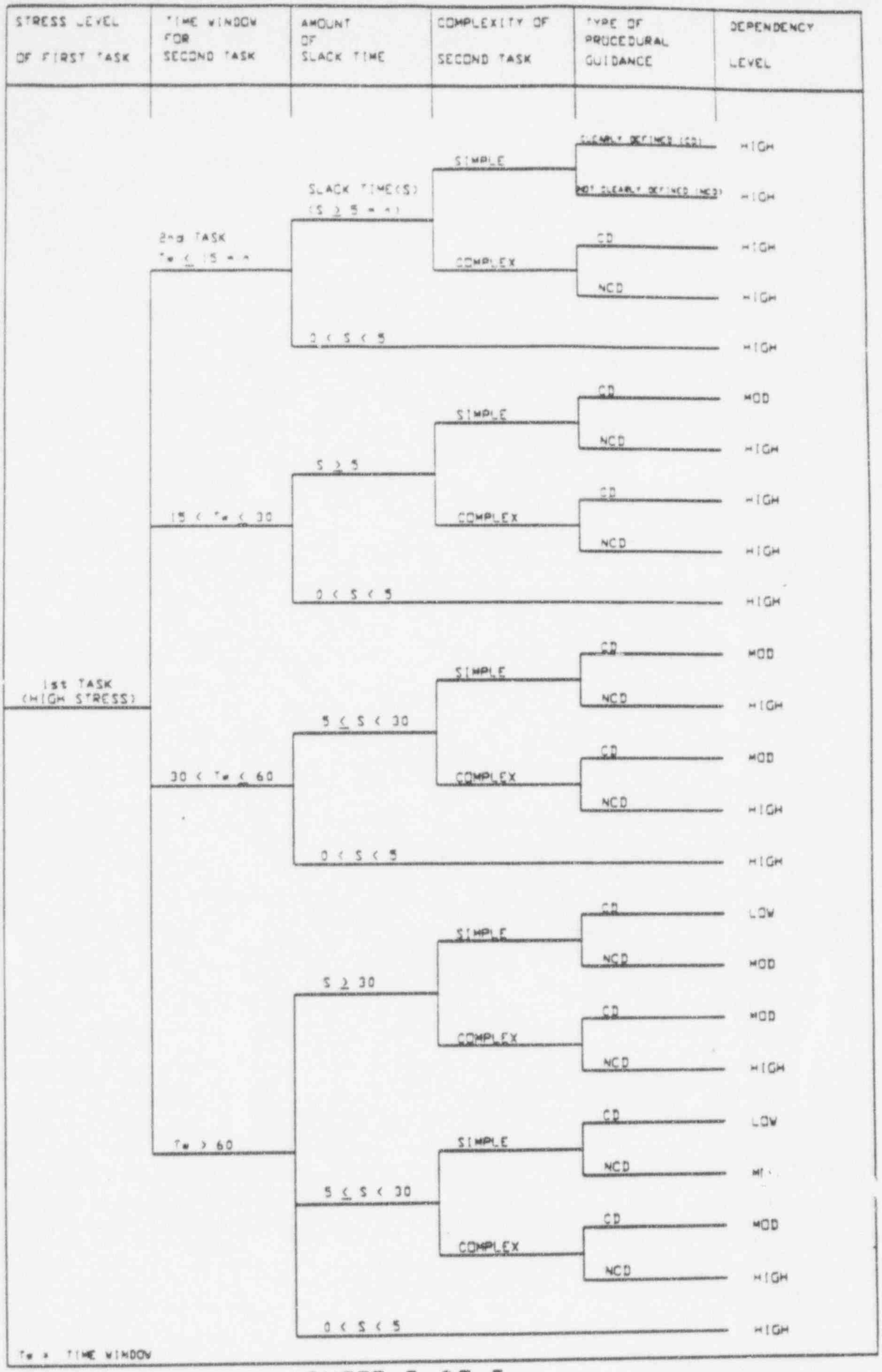
SHEET 1 OF 3

FIGURE 1



SHEET 2 OF 3

FIGURE 1



SHEET 3 OF 3

FIGURE 1 (continued)

Notes

1. If Event Unconditional Failure Probability is less than or equal to 1.0E-02, then apply the conditional failure probability as follows:

   a) 0.05 for low dependency;
   b) 0.15 for moderate dependency; and
   c) 0.5 for high dependency.

2. If Event Unconditional Failure Probability is greater than 1.0E-02, then evaluate conditional failure probability by the applicable equation as follows:

   a) $(1+19N)/20$ for low dependency;
   b) $(1+6N)/7$ for moderate dependency; and
   c) $(1+N)/2$ for high dependency;

Where, N is the unconditional failure probability of the dependent event.

3. Definition of terms (used in Figure 1) are provided as follows:

   a) Time window - Available time to perform the required tasks before system failure occurs;

   b) Actual time - Estimated time to perform the required tasks;

   c) Slack time - "Time window" minus "Actual time";

   d) Simple task - Activities consisting of less than 10 steps, and not involving any specific operator interaction or dependency;

   e) Complex task - Activities consisting of 10 or more steps, and/or involving more than normal operator dependency;

   f) Clearly defined procedure - Steps are such that operators do not have to shuffle between procedures, and/or steps are not confusing or ambiguous.

TABLE 1
CONDITIONAL PROBABILITY EVALUATION SUMMARY

Case	Preceding Event		Dependent Event Characteristics							
Name	Name	Stress	Name	Times		Tasks	Procedure	Dependency level	Uncond. Prob.	Cond. Prob.
				Available	Actual	Simple/ complex	Clear/ Unclear			

APPENDIX

Examples of Dependency Evaluation for Vogtle IPE

This Appendix provides the dependency evaluation among operator actions in the Vogtle IPE. The dependency levels determined from this evaluation are summarized in Table HRA-Q9, below.

Dependency evaluation is conducted for the following events in the Vogtle HRA:

a. OAC (cooldown & depressurize RCS) --
dependent on OAP (depressurize the primary side) during a SGTR event

b. OAD (depressurize secondary side) --
dependent on OAI (isolate ruptured SG) during a SGTR event

c. OAP (depressurize primary side) --
dependent on OAD (depressurize the secondary side) during a medium LOCA

d. OAP (depressurize primary side) --
dependent on OAI (isolate ruptured SG) during a SGTR event

e. OAR (establish cold leg recirculation) --
dependent on OAN (establish normal RHR) during a small LOCA

f. OAS (establish CTMT spray recirculation) --
dependent on OAR (establish cold leg recirculation) during a large, medium or small LOCA, ATWS, transient, station blackout, or SGTR

g. OAT (terminate SI and transfer to normal charging and letdown) --
dependent on OAP (depressurize the primary side) during a SGTR

h. OCR (insert control rods) --
dependent on OMG (trip MG sets) during an ATWS.

i. OFC (Control of TDAFW pump) --
dependent on opening of doors during SBO

j. CBHVAC-LOPD (Open inverter room doors) --
dependent on LOSP initiator

ATWT Sequence

One additional conditional probability evaluation is performed on a sequence of operator actions in ATWT initiating event sequence. This sequence of operator actions and their unconditional failure probabilities, derived from the SLIM methodology, are as follows:

1) ORT - 8.00E-03 (failure to initiate manual reactor trip); followed by
2) OMG - 4.40E-03 (failure to trip MG sets); followed by
3) OCR - 1.43E-02 (failure to insert control rods); followed by
4) OBR - 1.40E-02 (failure to execute emergency boration).

The following dependency relationship is believed to exist among these events:

a) OMG has a high dependency on ORT due to the time constraint
b) OCR has a high dependency on ORT and OMG due to the time constraint
c) OBR has a moderate dependency on OCR due to the actions toward the same goal.

Therefore, the quantification of this sequence of events is as follows:

HEP = 8.00E-03 x 5.00E-01 x 5.07E-01 x 1.55E-01 = 3.14E-04.

This HEP is approximated to 3.00E-04.

TABLE HRA-Q9
CONDITIONAL PROBABILITY EVALUATION SUMMARY

CASE	PRECEDING EVENT		DEPENDENT EVENT CHARACTERISTICS							
NAME (Initiator)	NAME	STRESS	NAME	TIMES		TASKS	PROCEDURE	DEPENDENCY LEVEL	UNCOND. PROB.	COND. PROB.
				AVAILABLE	ACTUAL	SIMPLE/ COMPLEX	CLEAR/ UNCLEAR			
LLOCA	OAR-LPLL	Moderate	OAS-LLF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
MLOCA	OAD-MLA	Low	OAP-MLF	20 min	< 5 min	Simple; 8 steps	Clear	Low	2.00E-02	6.90E-02
MLOCA	OAR-LPMLB	Moderate	OAS-MLFL	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
MLOCA	OAR-HPML	Moderate	OAS-MLFH	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
SLOCA	OAN-SL	Moderate	OAR-LPSLB	130 min	< 5 min	Complex; > 10 steps	Unclear	High	1.57E-03	5.00E-01
SLOCA	OAR-LPSLD	Moderate	OAS-SLF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
SLOCA	OAR-HPSLB	Moderate	OAS-SLFH	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
TRANS	OAR-HPTR	Moderate	OAS-TRF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
SSBO	OAR-HPSSO	Moderate	OAS-SSOF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01

<u>TABLE HRA-Q9 (Continued)</u>
<u>CONDITIONAL PROBABILITY EVALUATION SUMMARY</u>

CASE	PRECEDING EVENT		DEPENDENT EVENT CHARACTERISTICS							
NAME (Initiator)	NAME	STRESS	NAME	TIMES		TASKS	PROCEDURE	DEPENDENCY LEVEL	UNCOND. PROB.	COND. PROB.
				AVAILABLE	ACTUAL	SIMPLE/ COMPLEX	CLEAR/ UNCLEAR			
SSBI	OAR-HPSSI	Moderate	OAS-SSIF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
ATWT	OMG	Moderate	OCR-B	1 min	< 1 min	Simple; 1 step	Clear	High	1.43E-02	5.07E-01
ATWT	OAR-HPATB	Moderate	OAS-ATF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01
SGTR	OAI-SG	Moderate	OAD-SGN	60 min	10 min	Simple; 5 steps	Clear	Low	3.97E-03	5.00E-02
SGTR	OAI-SG	Moderate	OAP-SGN	20 min	< 5 min	Simple; 4 steps	Clear	Low	7.14E-03	5.00E-02
SGTR	OAP-SGN	Moderate	OAT-SGN	10 min	< 5 min	Complex; > 10 steps	Clear	High	3.83E-03	5.00E-01
SGTR	OAP-SGI	Moderate	OAC-SGIN	60 min	10 min	Complex; > 10 steps	Clear	Moderate	8.37E-03	1.50E-01
SGTR	OAP-SGN	Moderate	OAC-SGNN	60 min	10 min	Complex; > 10 steps	Clear	Moderate	8.37E-03	1.50E-01
SGTR	OAR-HPSGB	Moderate	OAS-SGF	10 min	< 2 min	Simple; 8 steps	Clear	Moderate	4.73E-04	1.50E-01

TABLE HRA-Q9 (Continued)
CONDITIONAL PROBABILITY EVALUATION SUMMARY

CASE	PRECEDING EVENT		DEPENDENT EVENT CHARACTERISTICS							
NAME (Initiator)	NAME	STRESS	NAME	TIMES		TASKS	PROCEDURE	DEPENDENCY LEVEL	UNCOND. PROB.	COND. PROB.
				AVAILABLE	ACTUAL	SIMPLE/ COMPLEX	CLEAR/ UNCLEAR			
SBO	N/A	High	OFC	20 min	12 min	Complex;	Unclear	High	2.93E-02	5.15E-01
LOSP	N/A	High	CBHVAC-LOPD	60 min	10 min	Simple; 1 step	Unclear	Moderate	7.33E-02	2.06E-01
ATWT	N/A	High	ORT; OMG; OCR; OBR	~ 10 min	~ 5 min	Simple	Clear	High / moderate	See: ATWT Sequence Section in Appendix	3.00E-04

Question 10

The submittal provides no detailed discussion of HRA performed for post-core damage operator actions reported in the summary descriptions of the dominant sequences. In particular, sequences 2, 8, 10, 11, 15, 18, 19 and 20 identify that "operator action is credited for isolation of certain containment penetrations." However, very limited information is provided as to how this action was modeled using HRA methods. Therefore, please supply the following:

a. Discuss the process used to identify and select this operator action for inclusion in the model. For example, the process may include review of operations procedures, discussion with operators or other plant support personnel on interpretation of procedures, on expected emergency response team activities, and so on. Include the steps taken to assure that selection of post-core damage recovery actions was based on careful examination of plant conditions, procedures, and practices.

b. The process used to quantify the human error probabilities of post-core damage human events was reported as SLIM. SLIM involves the assessment of plant-specific PSF information as a basis for interpolating between "anchor point" human error probabilities. Please explain how anchor points were selected for post-core damage human error probabilities and how the selection and evaluation of PSFs for the post-core damage actions were made.

c. How were dependencies addressed and treated in the post-core damage HRA? The performance of the operator is both dependent on the accident in progress and the past performance of the operator during the accident sequence of concern. Improper treatment of these dependencies can result in the elimination of potentially dominant Level 2 sequences and, therefore, the identification of significant events. Please provide a discussion, with examples illustrating how dependencies were addressed and treated such that important Level 2 sequences were not eliminated. If the IPE did not address such dependencies in the quantification, please justify this omission.

Response 10

The Vogtle IPE did not include post-core damage operator actions. Only those actions that the operating crew would perform prior to core damage were included in the IPE model. Moreover, only those actions for which written procedures exist were specifically modeled.

The operator action, OCI-Manually Isolate Containment, which was the source of this question, would be performed by the operators prior to core damage. This operator action is specifically identified in the Vogtle Emergency Operating Procedures, E-0 Reactor Trip or Safety Injection (Procedure No. 19000, Revision 9), Step 7. The Vogtle Emergency

Operating Procedures are structured such that the operators are required to verify Containment Isolation Phase A has occurred following either manual or automatic Safety Injection which would proceed any accident sequence that would result in core damage. If Containment Isolation Phase A did not occur, the operators are instructed to manually actuate Phase A Containment Isolation which is the action specifically modeled by the Vogtle IPE.

The operator action, OCI-Manually Isolate Containment, is not modeled as a post-core damage operator action. Although this operator action would be addressed very early in any core damage accident sequence as discussed above, it was purposely placed at the end of the Plant Response Trees (PRTs) for several reasons. First, the operator action to manually initiate Containment Phase A Isolation is not a core damage mitigation feature and therefore has no impact or influence on the out-come of the accident sequences in terms of preventing or mitigating core damage. Second, by placing this operator action at the end of the accident sequence, the PRT structures are simplified by not having to address this operator action for success sequences. Third, placing this operator action at the end of the PRTs provides for an orderly sequencing of only the core damage mitigation features which aides in the understanding and use of the PRT models. Lastly and most importantly, this operator action was specifically included to aid in the determination of the resulting core damage category assigned to each core damage accident sequence. This provides an important link between the Level 1 Plant Analysis and the Level 2 Containment Analysis.

Since the Vogtle IPE did not include post-core damage operator actions, responses to parts a, b, and c are not applicable.

BACK-END QUESTIONS

Question 1

The Vogtle Units 1 and 2 IPE back-end results in the submittal showed that steam generator tube rupture (SGTR) would lead to more than 10 percent release of volatiles with a frequency of 1.56E-6 per reactor year. However, you have defined four SGTR functional sequences, SGE02BH, SGL11BH, SGL20BH, and SGE158BH, that fall below the reporting criteria and differ mainly by the core and containment cooling status defined in Table 3.1-4, page 3-30. Because the containment cooling status is generally unimportant for SGTR sequences, please justify your position on not combining these sequences into one functional sequence.

Response 1

The process of establishing a plant damage state encompassed the assignment of a designator reflecting the ECCS status as well as that for the containment heat removal. This designator was applied to all the PRT end states independent of the initiating event. This facilitated a consistent approach in establishing the link between Level I and Level II. The plant damage states then become the Level II functional sequences. Functional sequences are screened to form bins used for sequence selection in the source term analysis, this process is delineated in Section 4.7.2 of the submittal. The process recognized that the steam generator tube rupture cases were not dependent upon the ECCS injection or the status of containment heat removal. The four damage states or functional sequences SGE02BH, SGL11BH, SGL20BH and SGE15BH were combined as a single source term bin (ref. Table 4.7-3). Thus, although the process did not begin with the combination of these sequences into a single functional sequence, the process followed did combine them, thereby precluding their elimination due to screening on frequency.

Question 2

In using the back-end screening criteria, the submittal has not listed the sequences that met the screening Criterion 3 ("any functional sequence that has a core damage frequency greater than or equal to 1E-6 per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the PWR-4 release category of WASH-1400") because this criterion was bounded by Criterion 1 ("any functional sequence that contributes 1E-6 or more per reactor year to the core damage frequency (CDF)") (page 31 o the submittal). However, because of the potential risk importance, please identify whether any sequence met Criterion 3. If a sequence did meet criterion 3, describe the sequence and its contribution to predicted radionuclide releases.

Response 2

Sequences which meet criterion 3 have a frequency greater than 1E-06 and a release greater than that specified in WASH-1400 as PWR-4. All functional sequences with a frequency greater than the specified cutoff frequency were identified through criterion 1. Releases meeting the PWR-4 results are equated to a volatile fission product release greater than 10%. Referring to Table 4.7-4, these would be release categories J, M, V and W (the other release categories in Table 4.7-4 exceeding 10% volatiles and not included in the above listing did not involve containment failure. These were cases of either a bypass or impairment). Assignment of release categories to the source term bins is contained in Table 4.7-7. A review of this table indicates that there are no bins with the release categories listed above, thus there are no sequences which resulted in a release of greater than that of PWR-4 with a frequency of 1E-06, no sequences satisfied criterion 3.

Question 3

In response to the request from the Internal Review Group (IRG) to provide the technical basis for selection of 2-inch screening criterion for addressing containment isolation failures in the Vogtle IPE, Table 5.3-1, page 5-16 of the submittal, notes the following:

> A systematic review of the Vogtle containment penetrations concluded that there are no penetrations which connect directly to the containment atmosphere, penetrate the containment boundary, and are less than 2 inches in diameter that constitute potential isolation failures.

What was the basic used for identifying "potential isolation failures?" What would be the release over 48 hours from a 2-inch containment isolation failure?

Response 3

Potential containment isolation failures as referred to in Table 5.3-1 are based upon the configuration of the line penetrating containment. A potential containment isolation failure would meet the following criteria:

1) The line must connect directly to the containment atmosphere, and

2) Penetrate the containment and end outside containment, (several small lines loop out of containment through a sampling type device and return without being exposed to conditions outside containment), and

3) Remain normally open during plant operation, and

4) Receive an automatic signal to close and isolate.

A two inch line meeting these criteria would constitute a potential containment isolation failure. The review process for two inch lines is summarized in the submittal sub-section titled Containment Isolation Failure (under section 4.4.3 Failure Modes). It is stated that eight penetrations from Table 6.2.4-1 of the FSAR were identified which connect directly to containment atmosphere. Six are closed to the environment outside containment and the remaining two are isolated during normal plant operation. Hence, the conclusion noted above and on page 5-16 of the submittal, that there are no potential isolation failures of two inch diameter or less containment penetration lines.

The release for a two inch line would be dependent upon the accident and timing of events. A sensitivity case for a failure to isolate was analyzed for a three inch line. These results are summarized in Table 4.7-12 of the submittal as sequence 11-N13. The three inch failure released 4.3% of the volatile fission products compared to 2.8% for the 8 inch line (base case). A two inch isolation failure would be comparable to that for the three inch line and the results would not necessitate a change in release category. Modeling of the line losses (not included in our analysis) would act to further reduce the predicted release.

Question 4

As noted in Section 4.4.2, page 4-15 of the submittal, a 14.3-percent concentration of hydrogen for the Vogtle containment was calculated by assuming 100-percent oxidation of all zirconium and the lower core plate. Using a procedure of Sherman and Berman, you concluded that failure of the Vogtle containment as a result of hydrogen detonation was very unlikely. This conclusion is based on the condition that no obstacles exist in the path of the expanding unburned gases. However, obstacles in the path would cause detonation at lower concentrations thus increasing the likelihood of containment failure. Please describe whether, or how, you considered obstacles in the containment before arriving at such a conclusion.

Response 4

Deflagration to detonation transitions (DDT) were considered separately for the Vogtle annular, lower, and upper compartments in FAI/91-72, "Vogtle Electric Generating Plant Units 1 and 2 Phenomenological Evaluation Summary on the Probability and Consequences of Deflagration and Detonation of Hydrogen in Support of the Individual Plant Examination." All three containment regions were analyzed based on their geometric configurations according to the methodology of Sherman and Berman. Additionally, since the lower and annular compartments were considered as channels with transverse venting, they were also assessed according to their mixture reactivity.

The mixture reactivity assessment utilized scaling of detonation cell widths between the FLAME facility (small scale) and the reactor geometry. Results presented in FAI/91-72 indicate that the necessary detonation cell width is larger than the physical boundaries of either the annular or lower compartment. For instance, the minimum channel size required at the reactor scale to accelerate a flame to DDT given a 14.3% hydrogen concentration is 360 ft x 480 ft. This compares to the annular and lower compartment dimensions between the 171 -9 and 213 elevations of 21 x 40 and 23 x 41 , respectively. Based on the scaling assessment, there is no potential for DDT in the lower and annular compartments of the Vogtle containment.

The Sherman and Berman methodology provides a procedure, based on engineering judgement, to estimate the potential for DDT. The procedure assumes that the potential for DDT can be evaluated based on the mixture intrinsic flammability (detonation cell width) and type of geometry. Five classes of mixture sensitivity are defined ranging from class 1, most detonable, to class 5, least detonable. The mixture class can be readily assigned given a hydrogen mole fraction. Five geometry classes are also defined, ranging from 1, very favorable to DDT and featuring large geometries with obstacles and partial containment, to class 5 very unfavorable to DDT and featuring large scale and complete unconfinement. The combination of mixture class and geometry class leads to a matrix of result classes which qualitatively describe the DDT potential. The result class matrix, taken from FAI/91-72 is shown in Table 4-1, below.

Table 4-1
Matrix of Result Classes

Geometric Class	MIXTURE CLASS				
	1 Most Detonable	2	3	4	5 Least Detonable
1 (Very favorable to DDT)	1	1	2	3	4
2 (Favorable to DDT)	1	2	3	4	5
3 (Neutral to DDT)	2	3	3	4	5
4 (Unfavorable to DDT)	3	4	4	5	5
5 (Very unfavorable to DDT)	4	5	5	5	5

Result Class Description

Result Class 1.	DDT is highly likely.
Result Class 2.	DDT is likely.
Result Class 3.	DDT may occur.
Result Class 4.	DDT is possible but unlikely.
Result Class 5.	DDT is highly unlikely to impossible.

Based on a hydrogen molar concentration of 14.3%, the mixture class for any of the containment regions was selected as class 4: detonation unlikely. Based on containment walkdowns and a review of plant drawings, the lower and annular compartments were assigned geometric class 4: geometries unfavorable to flame acceleration. Examples are large volumes with hardly any obstacles and a large amount of venting transverse to the flame path. DDT will usually not occur in a class 4 geometry. Both the annular and the lower compartment can be considered as channels with transverse venting due to extensive grating which allows good communication with the upper compartment. Similarly, upward flame propagation in both compartments will also be "vented" sideward because of the ring shape of the compartments. According the Table 4-1, a mixture class of 4 and a geometry class of 4 lead to a result class of 5: DDT is highly unlikely to impossible.

Due to its large open volume, the upper compartment was considered an unconfined geometry at large scale (geometric class 5) which is very unfavorable to flame acceleration. The combination of mixture class 4 and geometric class 5 again leads to result class 5: DDT is highly unlikely to impossible.

Since the Sherman and Berman procedure is subjective, the sensitivity of the results to engineering judgement can be investigated by assigning each containment compartment a more conservative geometry class. That is, suppose that the lower and annular compartments are designated as geometric class 3: geometries that yield moderate flame acceleration, but are neutral to DDT. An example of this geometry is a large tube without obstacles. The result class changes to class 4: DDT is possible but unlikely. However, when this sensitivity result is considered in conjunction with the mixture reactivity assessment presented above, it is still sound to conclude that DDT is a highly unlikely event.

If the upper compartment geometry class changes from 5 to 4, the result class remains class 5: DDT is highly unlikely to impossible.

In summary, the conclusion that DDT is highly unlikely is based both on an independent scaling assessment for the lower and annular compartments, and on the qualitative procedure of Sherman and Berman. Inputs for these assessments are derived from containment walkdowns and review of plant drawings. Finally, it has been shown above that there is sufficient margin in the analysis to account for the subjective nature of the procedure.

Question 5

You have assumed that only 2 percent of the entrained core debris would make it past the 90-degree turn from the instrument tunnel to the annual compartment (a debris mass of 2,800 lbs) (Section 4.4.2, page 4-16 o the submittal). This appears to be a relatively small fraction of melt participating in a direct containment heating (DCH). You have based this assumption in part on the special feature of the Vogtle cavity and instrument tunnel that enhances melt de-entrainment. Please provide the basis for assuming a value of 2 percent, including the role of the 90-degree turn from the instrument tunnel.

Response 5

FAI/91-122, "Vogtle Electric Generating Plant Units 1 and 2 Phenomenological Evaluation Summary on Direct Containment Heating in Support of the Individual Plant Examination," provides the basis for determining that only 2% of the entrained debris mass would exit the reactor cavity instrument tunnel and reach the annular compartment. The technical basis is as follows.

Sandia National Laboratory [Walker, 1987] developed a model for estimating the likelihood of debris particles not deflecting with the flow due to a change in flow direction, thus impacting structural boundaries of the flow path. The model was developed for the Zion seal table and instrument tunnel structure where the steam and debris escaping from the cavity makes a 90 degree turn to escape into the lower compartment. An analogous equation was used to describe de-entrainment of debris particles as gas makes a 90 turn to exit the Vogtle reactor cavity.

The Zion and Vogtle cavity configurations are compared in Figure 5-1. The results of this plant-specific calculation indicate that only 2% of the entrained debris mass will reach the annular compartment.

Whereas the model of Walker accounts for a single 90 turn, core debris leaving the Vogtle reactor vessel will have to make two 90 turns before exiting the instrument tunnel to the annular compartment (see Figure 5-1). Also, at Vogtle, there is a concrete platform which extends part way across the instrument tunnel, forming a "lip" which further enhances debris de-entrainment. Thus, the 2% figure obtained from the Walker model is considered to be conservative.

<u>Reference:</u>

Walker, J. V., 1987, "Reactor Safety Research Semi-Annual Report," NUREG/CR-5039, SAND87-2411.

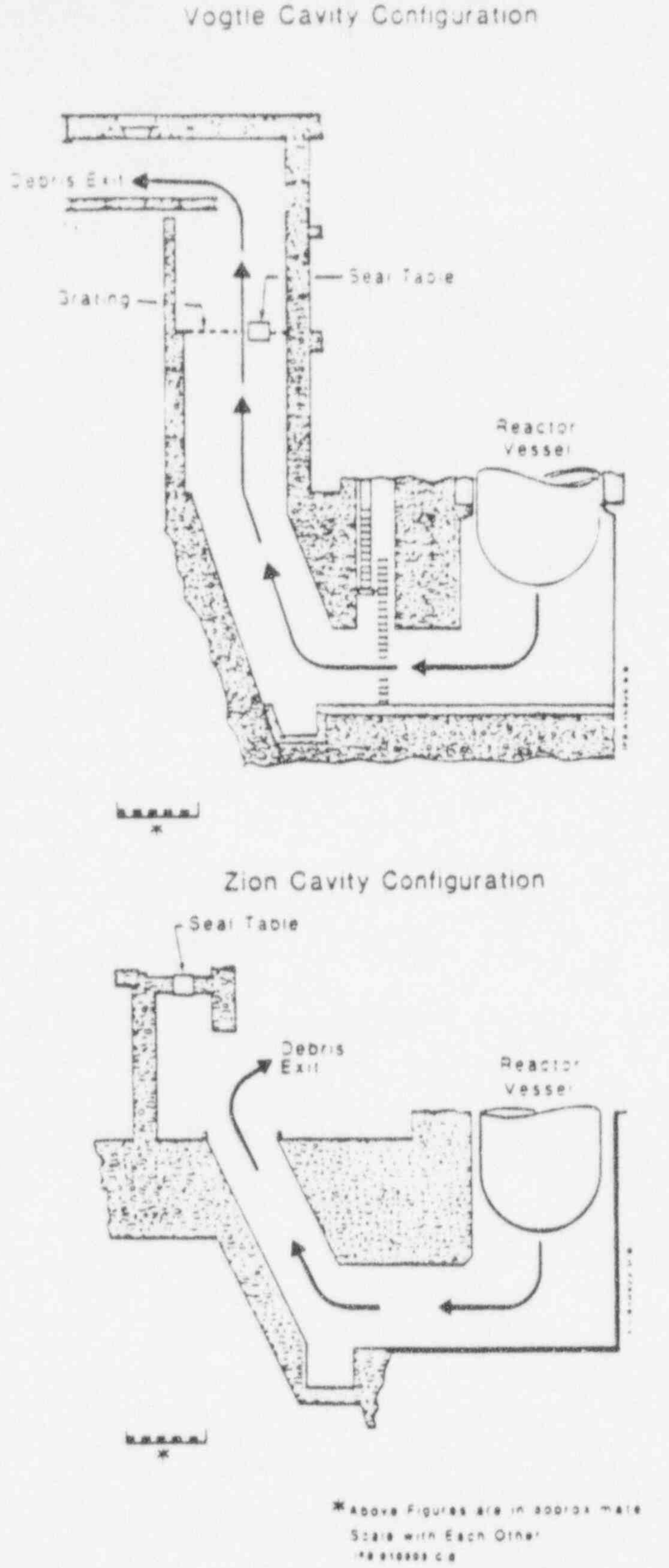


Figure 5-1 Comparison of Vogtle and Zion cavity configuration.

Question 6

What was the amount of core material that was assumed to be released from the failed vessel for each of the sequences analyzed? Describe the sensitivity analysis performed on the impact of the quantity of core material released. In particular, what is the impact of releasing 100 percent of the core material on containment performance and containment failure?

Response 6

The amount of core material released from the failed vessel for each analyzed sequence is assumed to be identical and equal to 100% of the total core inventory (fuel, cladding, support and internal structures) of 350,000 lbm. Detailed MAAP results are contained in FAI/92-58, "Vogtle Electric Generating Plant Source Term Notebook," Vol. 3.

Thus, the impact on containment due to 100% release is implicit in the analyzed source term sequences. No additional sensitivity sequences were necessary, nor performed, to quantify the effect of complete core release on containment performance.