V.C. Summer Nuclear Station

Post-Examination Comments

January 10, 2006
Initial Written Examination

Examination 2005-301

January 25, 2006

Enclosure 1
Post-Examination Comments

I. Introduction

The post-examination review identified certain issues in relation to specific questions addressed in detail in this enclosure. The comments were developed using the criteria set forth in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors" (Rev. 9), which is referenced in 10 C.F.R. § 55.40 as establishing the criteria for preparing written examinations. NUREG-1021, Appendix A, "Overview of Generic Examination Concepts," and Appendix B, "Written Examination Guidelines," provide detailed guidance for the examination preparer, and are useful in evaluating questions following completion of the examination. Certain concepts apply to the majority of the individual comments. These are discussed first, and then comments are provided on individual questions.

II. General Concepts

A. Level of Knowledge

The examination appears to contain a number of questions that require a level of knowledge based on memorization of procedure steps or plant parameters, or that challenge the examinee on functions outside the scope of an operator's responsibility to perform or to memorize. While NUREG-1021 guidance discusses the importance of knowledge testing (Appendix B, Section B.1), it notes that the knowledge tested should relate to the ability to perform a job, and that it not require an operator to remember trivial information embedded in or covered by procedures (Appendix B, Section C.1.a). The guidance also notes that authors tend to underestimate the difficulty of a concept and, therefore, it is important for examination preparers to address the appropriate level of knowledge expected of the examinee (Appendix B, Section C.1.e).

NUREG-1021, Appendix A, explains that "[f]or a test to be considered valid, it must be shown to measure that which it is intended to measure," and that operator examinations are intended to measure the knowledge and ability of examinees, "such that those who pass will be able to perform the duties of a reactor operator (RO) or senior reactor operator (SRO) to ensure the safe operation of the plant" (Appendix A, Section C). Appendix A identifies three principal facets for examiners to use to establish the validity of NRC examinations:

- Content Validity – establishing a link to job duties and using a sampling plan;
- Operational Validity – testing actual or conceivable mental or psychomotor activity performed on the job; and
- Discrimination Validity – ensuring that the test discriminates between examinees who are capable and those who are not in order to make "reliable and valid

distinctions at the minimum level of competency," but not to identify levels of competency or the "most qualified."

Based on this guidance, we are concerned that certain questions focused on activities outside the scope of an operator's job (e.g., performing pre-job dose estimates) or focused on steps in procedures that we do not require operators to memorize (note that the January 10, 2006, examination was a closed-book examination and no procedures were provided as reference material during the examination). For example, some questions related specifically to an action directed in one or more procedures, where an operator is expected to follow the procedure rather than commit the actions to memory. On this basis, we are requesting that certain questions be deleted, as explained in the individual comments below.

Further in support of this concern, our licensing basis and implementing processes establish the expectations for procedural adherence. Consistent with the NRC regulatory requirements, the V.C. Summer Final Safety Analysis Report (FSAR), Section 13.5.2, "Control Room Operating Procedures," states that "Control Room operating procedures are those procedures that are performed by the licensed Control Room Operator or under his direction and control. They are a preplanned method for the conduct of operations to minimize reliance on memory." Section 13.5.2.2 of the FSAR also defines the scope of procedures and actions necessary for responding to emergencies—i.e., Emergency Operating Procedures (EOPs)—and specifies that "[t]hose sections of the procedure that require immediate response action from the operating crew are committed to memory." Section 13.5 explains that "[s]ome procedural steps such as immediate action steps for emergency procedures are required to be committed to memory. Routine actions are performed in accordance with approved written procedures."

Note 6.13.c of Operations Administrative Procedure, OAP-103.4, "EOP/AOP User's Guide," distinguishes between AOP steps and EOP steps, noting that "AOP immediate operator actions are not required to be specifically committed to memory." SAP-123, Step 6.3.2, of Station Administrative Procedure SAP-123, "Procedure Use and Adherence," explains that certain procedures may be performed from memory – but this does not relieve the performer from procedural adherence – but that those procedures marked as "Continuous Use" must be performed "exactly" according to the procedure and in the step sequence specified (unless the steps are specified as not sequence critical).

Our comments on individual questions include several examples that appear to require knowledge of specific procedural steps or parameters that need not be committed to memory. These and other questions may not meet the NUREG-1021 criteria for content and operational validity (NUREG-1021, Appendix A, Section C). In addition, the guidance and criteria in NUREG-1021, Appendix B, "Written Examination Guidelines," explains that the "most effective tests of knowledge include questions and test items that measure the application of knowledge that directly relates to an individual's job" (Section B.1) and that a generic principle of question development is to "[e]nsure that the concept being measured has a direct, important relationship to the ability to perform the job"

(Section C.1.a). These memory and knowledge questions, and questions related to job functions, are numbers 9, 11, 26, 30, 41, 45, 69, 77, 84, 87, 93, 94, and 96.

B. Consistency of Examination Structure

NUREG-1021, Appendix A, discusses the importance of consistency in preparing examinations. Our review of the examination questions indicates a number of problems in this regard:

1. A significant number of questions on the SRO written examination did not evaluate any of the 7 items required by 10 CFR 55.43(b) for SRO written examinations. The links to 10 CFR 55.41(b) for these test items were not established at a level of knowledge or ability required specifically by SROs.

2. A large number of questions that did not test the knowledge or ability required by the intended K/A topic. This has resulted in an examination that did not meet the requirements of NUREG 1021, ES-401. K/A mismatch is considered an unacceptable flaw in accordance with ES-401, Section E.2.d.

3. After completing a post examination review, we believe that the requirements NUREG 1021, ES-401-6, Written Examination Quality Checklist, Items 3, 9, and 10 were not adequately met. This may have been a contributor to the following examination deficiencies:

    a. Several topics were tested multiple times on the RO and the SRO examination. The SRO examination contained three test items in one topic area in Tier 1.

    b. A significant number of required topics in tier 1, and one K/A Category in tier 2 were not examined at all as required by NUREG 1021.

    c. Duplicate K/A topics evaluated by more than 1 test item.

    d. Several questions evaluated knowledge at the generic fundamentals level; information examined at a level and context that an operator would not apply as part of their job.

    e. A high number of questions requiring only memorization of fact or fundamental thought, resulting in an examination that we believe provided marginal evaluation of the higher cognitive thought processes required of operators on the job.

Enclosure 2, Section 1, reflects our efforts to assign each question in its proper tier, group, and K/A. Enclosure 2, Section 2, documents our assessment of:

- SRO questions 77*, 80*, 83, 85, 86, 88, 89, 91, 93*, 95, 96*, 98, 99, and 100, and

- RO questions 2, 4, 5, 7, 12, 14, 15, 16, 18, 21*, 24, 25, 26*, 28, 29*, 30*, 31, 33, 35, 37, 46, 49, 51, 53, 55*, 56, 58, 62, 63, 67, 69*, 72, 73, and 74.

  (* These questions are also contained in Enclosure 1 comments on individual questions with specific recommendations to delete the question or accept alternate answers.)

C. Multiple or No Correct Answers

In some cases, a question may include multiple answers that should be accepted as correct, based on plant-specific features, operating conditions, or other considerations not addressed or misstated in the stem. In addition, a limited few questions may not include a correct answer. Where these concerns are identified, justification is provided with specific cites to the supporting references, with a copy of the references themselves attached to the discussion. This is inconsistent with guidance for multiple-choice questions in NUREG-1021, Appendix B, Section C.2. The questions discussed in individual comments which were identified as having multiple correct answers or no correct answer include 11, 21, 29, 45, 55, 75, 77, 80, 92.

III. Post-Examination Comments on Individual Questions

Post-examination comments on individual questions follow.

The first item addresses JPA-001, "Review Work Package for SFP HEX 'A.'" During the on-station portion of the examination process, job performance measures ("JPM") were conducted with the applicants. One such JPM involved review of work documents for determination of Equipment Out Of Service ("EOOS") implications. The discussion contains a recommendation for the conclusion that Step 5 of JPA-001 is not a critical step.

Following the JPA-001 discussion, specific questions are addressed. The material includes the examiner's key, which includes the question, the explanation for the correct answer, and the relationship of the question to the K/A deemed applicable to the question. Comments on the questions then are provided, along with any reference material applicable to the justification for the recommended action on the particular question.

JPA-001

JPA-001: Review of Work Package for SFP HEX 'A'

Recommendation:

Step 5 of JPA-001 should not be a critical step.

Justification:

Step 5 of this administrative Job Performance Measure (JPM) relates to the examinee's review of the green Preventive Maintenance Task Sheet (PMTS). The examinee was expected to review the PMTS and use Attachment VI, "Scheduling Package Checklist," of OAP-102.1, "Conduct of Operations Scheduling Unit," to ensure all pertinent package data is included and is correct. One aspect of this review is to conduct a risk assessment of the impact that the activity (tagout of the Spent Fuel Pool (SFP) heat exchanger), including the impact on Equipment Out Of Service (EOOS). This EOOS determination is initially made by the Operations Scheduling SRO. When the Operations Scheduling SRO completes the evaluation, he stamps the PMTS to indicate his EOOS impact conclusions. It then is verified by the duty Shift Supervisor (SS) along with Work Control Center personnel. The actual critical portion of the SS review is the risk assessment itself, not the application of the stamp. The SS would conduct the risk assessment when the out-of-service component was entered into the EOOS program.

Therefore, the absence of the EOOS stamp is not critical, as the risk evaluation would be conducted regardless of the presence, or absence, of the stamp. In addition, the Spent Fuel Cooling Heat Exchanger is not considered within the EOOS profile. Therefore, the EOOS stamp is not relevant to the activity assigned for the JPM.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

OPERATIONS ADMINISTRATIVE PROCEDURE

OAP-102.1

CONDUCT OF OPERATIONS SCHEDULING UNIT

REVISION 5

Original Signed By: D. A. Baker — 02/26/04
DISCIPLINE SUPERVISOR — DATE

Original Signed By: G. A. Lippard — 03/01/04
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	07/12/04		E	P	04/23/05	
B	P	10/04/04		F	P	05/09/05	
C	P	02/02/05		G	P	10/06/05	
D	P	02/24/05					

INFORMATION USE

Procedure May Be Performed From Memory.
User Retains Accountability For Proper Performance.

6) Directing the preparation of the Refueling Outage Removal and Restoration Log per SAP-205 and providing SRO review of that document.

7) Directing the preparation of the Refueling Outage Operations Retest Book and providing the SRO review for that document.

8) Determining Refueling Outage Operations Activity durations and getting those durations into the Refueling Scheduling.

9) Planning all Operations Refueling Outage task sheets and updating them in RDb.

10) Coordinating changes in SAP-201, Danger Tagging , with Operations and the other disciplines.

CHG B

11) Providing SRO reviews for Online work when Online Operations Scheduling manpower is low.

12) Provide On-shift, Shift Supervisor, coverage during times of vacation, sick leave, and other shift needs.

13) Provide Plant Experience input to the Planning and Scheduling organization.

14) Maintaining an active SRO license unless otherwise authorized by the Manager, Operations.

15) Completing Special Projects assigned by the Manager, Operations, Operations Supervisor or Work Control Supervisor.

b. The Operations Scheduling Supervisor is responsible for the following: CHG C

1) Directing the preparation of the daily work package.

2) Attending the T-7, T-4, and T-2 Work Week Meetings and assigned project planning meetings. CHG B

3) Performing review of work documents in the planning stages.

CHG D

4) If possible, complicated maintenance preplans and work activities:

   a) Should be reviewed with the Manager, Operations or the Operations Supervisor to ensure adequate reviews and approvals are obtained. CHG B

   b) Should receive a review from the WCC SRO and the WWM that will be on duty to ensure safe and timely completion of the activity. CHG C

5) When required, downgrading priority work documents or authorization to begin work as described below:

   a) Only personnel who are involved with tie WCC or Ops Planning and Scheduling On-Line Work or WCC personnel can downgrade a work document or authorize work to begin. CHG C

   b) Priority 1 or 2 work orders cannot be downgraded until the WCC SRO or Shift Engineer has been notified and concurs with downgrade. CHG C

   c) Reasons for downgrading the Work Order shall be documented in the Planner's Section of the Work Order.

6) Performing a reactivity management task screening per Enclosure I.

7) Assist in determining the most effective time for repairs:

   a) On-line:

      (1) Repairs can wait until the next Association Code or FEG for the component or system. CHG D

      (2) Non-scheduled equipment outage activities require prompt attention, but normal planning and scheduling activities are utilized to generate a complete work package.

   b) Off-line:

      (1) Reactor Trip Package, component should be worked when plant conditions would support its removal from service.

(2) Refueling Outage, maintenance activity is of such a nature only outages of sufficient duration are desired for repair.

8) Reviewing work package tagouts and R&Rs for completeness and accuracy.

9) Reviewing WPOs and assigned retests.

10) Reviewing work documents for MRT program applicability, stamps, and work priority.

11) Performing an Integrated Risk Assessment of scheduled work activities utilizing the EOOS computer program and available FEGs and FIDs. (Refer to Enclosure G and description of Fields in Enclosure H). CHG D

12) Problem solving between the Work Control Center and the Scheduling Group. CHG C

13) Maintaining a shift schedule when required during plant outages.

14) Maintaining the schedule for vacation, training, and proficiency watches to ensure the Operations Scheduling Unit is adequately manned during normal working hours. CHG B

15) Designating a Non-Supervisory SRO to fulfill the responsibilities of this position when absent.

16) Routing the Daily Work Package to the Work Control Center for implementation. CHG C

17) Maintaining an active operator license unless authorized otherwise by the Operations Supervisor.

18) Assisting the Discipline Planner in determining Risk/Plant Effect.

19) Monitoring the turnover training of personnel that have been assigned to Operations Scheduling Unit. All new personnel should receive hands-on training for their duties. At the conclusion of turnover training, verify new personnel are ready to assume their responsibilities.

b) Determination and assignment of the requirement for an operations retest.

c) Determination of Tech. Spec applicability.

2) Generation and maintenance of the Outage R&R and Retest Books.

3) Assisting with the generation of non-scheduled work packages when required.

4) Generation and maintenance of the Reactor Trip Package.

5) Attending assigned Scheduling Department and project planning meetings.

6) When required, supporting a shift schedule during plant outages.

7) Maintaining operator qualifications unless authorized otherwise by the Operations Supervisor.

f. Work Control Center personnel (WCC SRO, or SE) with regard to this procedure: CHG C

1) Discuss next days work package with the Operations Scheduling Representative and the Duty SS. CHG C

2) Review the impact of the work package on plant risk prior to removal from service.

3) Review the impact of testing (Preventive or Surveillance) on the availability of SSCs. (See Enclosure E).

4) Recalculate the EOOS Risk Assessment profile when plant conditions change:

a) Adjust Environmental Variances within EOOS program based on sound operator judgment.

b) Emergent work or equipment failures are of such a nature that risk values may be impacted.

c) Maintenance activities are delayed beyond expected completion times, which may now overlap other scheduled activities.

8) All work documents will have the EOOS Stamp placed upon them to alert Operations and Maintenance Personnel whether a task or removal of a component from service may or may not effect the operation of the plant.

   a) If the following occur the "Y" on the EOOS stamp should be circled indicating a potential adverse effect on plant operations:

      (1) The task or component is contained within the EOOS Risk Assessment Program and is entered per Enclosure E, or

      (2) Performing a task or removal of a component from service requires compensatory action(s) to negate effect or for the component to remain functional.

   CHG B

   b) If task or component is not contained in the EOOS Risk Assessment Program and compensatory actions are not required, the "N" is circled indicating no effect on EOOS Risk Management Program or plant operations.

9) The Equipment Lineup Request, Attachment I, should be included noting any special conditions or equipment lineup changes required for the tagout or work activity.

10) An EOOS Risk Assessment printout should be included with each daily work package. The printout should indicate any impact(s) on safety functions, or risk significant combinations caused by the planned activities. If necessary, amplifying notes on why an activity does or does not impact a safety function will be attached.

11) If available, an Integrated Risk Assessment along with associated FEGs and FIDs should be performed using Enclosure G. CHG D

12) A Scheduling Package Checklist, Attachment VI, should be completed for each work package.

13) After all forms are completed, they should be placed in a folder or plastic sleeve along with the associated work documents to ensure the package remains intact.

e. Work packages should be reviewed, routed, and executed as follows:

1) All work (daily, trip, and short duration outage) packages should be reviewed by an SRO. CHG C

2) The Operations Supervisor should be informed of additions or deletions to the Trip Package and any conflicts that may prevent or have prevented scheduled activities from being performed.

3) The daily work package will then be routed to the WCC SRO.

4) The WCC SRO will verify the daily work package reflects what is on the schedule to be worked.

C02 → 5) The WCC SRO shall discuss with the Duty Shift Supervisor all items scheduled for the next working day including any special plant conditions, equipment lineups, MRT restrictions, or EOOS assumptions required to support the scheduled activities. CHG C

6) The WCC SRO or Shift Engineer will update the EOOS Program prior to tagout/work release to ensure conditions have not changed such that an activity could now result in a Moderate, Elevated, or High Risk Level. See Enclosure E for information on the use of the EOOS Program.

7) The WCC SRO will update the Control Room and Auto Log as tagouts are hung or as EOOS changes due to scheduled activities or emergent work.

8) Tagouts and equipment alignments should be completed prior to 0600 each day to support scheduled activities.

9) Tagouts that are hung in the water treatment building should involve the duty chemist to ensure components are correctly removed from and restored to service. CHG B

10) Feedback should be submitted to the Operations Scheduling Unit on Attachment VII. Examples for feedback are:

   a) During or following review, package was found to be inadequate.

   b) Scope of work within the work package changed requiring tracking not originally planned.

USE OF EOOS:

General:

The assessment is performed for the maintenance train. Further review is required if simultaneous work on risk significant systems on both trains is required. This is not expected due to the train week philosophy.

The users are expected to be the Operations personnel in the Operations Scheduling Unit (for use during POW, POD, and general work package review), and the SS/SE(for review of emergent work, or in progress work with changing plant conditions). *The user should be a licensed operator since knowledge of the EOPs is necessary.*

The majority of the review process is conducted by the OPS Scheduling Unit personnel. They are expected to prevent risk significant combinations of activities prior to release of the work package to the Control Room. The primary task of the shift personnel is to ensure conditions have not degraded such that scheduled work has a higher risk impact. The following conditions are examples of changing conditions that would require a review of planned or in progress work by the on shift personnel:

1) Severe Weather - If an ice storm, hurricane, or other weather condition occurs that could increase the likelihood of a Loss of Offsite Power (LOSP).

2) Equipment Failure - If equipment fails that has some impact on safety functions.

3) Plant conditions - If conditions exist such that the likelihood of a plant transient is high. This is strictly a judgment call by the SS/SE. Examples could include special testing, unstable secondary systems, or forced shutdowns. These events come under the EOOS ENVIRONMENTAL VARIABLES heading. Plant evolutions are activities that significantly increase the potential for a plant transient or trip (e.g. Turbine Control VLV Testing, Main Feedwater Pump maintenance, operation with two Circulating Water Pumps, etc...). Critical Testing includes the Category 1, 2, & 3 tests in SSP-001, and other test activities that create similar transients.

If a significant risk impact is identified for elective maintenance that has not yet been released to the field, the SS/SE should hold or reschedule if possible. If a situation arises after the work activity has started, the EOOS program can be used to assist in prioritizing the restoration of equipment.

Performing Integrated Risk Assessment

Integrated risk assessment (exclusive of the EOOS program calculation) may be performed as follows on routine scheduled work or individual components:

By noting the **Criticality** or **FID** (of a component), **Impact** (of a repetitive task), **Work Order Class,** and **notebooks** (when indicated), the Integrated Risk Assessment with an activity or component can be determined. This information in addition to OE, professional experience, and judgment performs the Integrated Risk Assessment. CHG D

The Operations SRO Scheduling representative determines the level of management involvement. He is initially assisted in the risk assessment by input from the discipline planners who alert him to risk significant activities or components identified in the planning process.

The above mentioned component and activity attributes are found in RDb and are accessed as described below:

1) Log into RDb
2) Arrow to VIEW
3) ENTER
4) Arrow to BROWSE
5) ENTER
6) Arrow to MULTIPLE SELECTION
7) ENTER
8) Input work order number
9) ENTER
10) Select work order (use arrow keys)
11) Arrow to REVIEW WORK ORDER
12) ENTER
13) Note the work order **CLASS** (left side of screen)
14) KEYPAD 6
15) Arrow to DETAILED INFORMATION
16) ENTER
17) Note **CRITICALITY or FID** and **IMPACT** values CHG D
18) REMOVE (DELETE)
19) REMOVE (DELETE)
20) Arrow to Browse Job Steps

SCHEDULED WORK APPROVAL/DENIAL

Scheduled Work/Activity Date ______________________________

Description of Work/Activity to be Performed __________________________________

____

____

I. This Moderate Risk, Elevated Risk, High Risk, or Cross Train activity is approved for work provided the required plant conditions are available on the scheduled due date. The following items were considered for making this approval:

____

____

____

_________________ Operations Scheduling Supervisor

_________________ Operations Supervisor (Moderate Risk or Cross Train) CHG B
In the absence of the Operations Supervisor:
Operations Scheduling, Shift Supervisor

_________________ GMNPO/MDS (Elevated Risk)

_________________ PSRC (High Risk)

II. This work activity/package cannot be performed on the scheduled date due to the following reason(s):

____

____

____

_________________ SRO (WCC or On Shift) CHG C

_________________ Operations Scheduling Supervisor

III. Recommended re-schedule date or plant conditions:

____

____

SCHEDULING PACKAGE CHECKLIST

SYSTEM/COMPONENT ____________________

Date Scheduled: ______________________ Tagout #: ___________________________________

	YES	N/A		
			Correct train work week: Work Week (circle) A1 A2 B1 B2	
			Is tagout as requested?	
			All work within tagout boundary?	
			Instruments affected by tagout evaluated for impact on system?	
			Power secured?	
			DC Control Power breakers in REALIGNMENT Section or tagged as required?	
			Install sequence correct?	
			Work document numbers on WPO and tagout?	
			WPO Index number?	
			Restoration lineup verified per SOP?	
			Component worked in REALIGNMENT Section?	
			Switch and/or Fuse Hold tags included in package, if necessary?	
			Each Discipline is assigned to the correct requested tags on the tagout. (X in Box)	
			Tagout has index number (and # is on all work documents)?	
			System/Electrical drawings are verified to be: Current revision, marked with tagout index #, reviewed, and included? If the latest drawing revision, CHAMPS, or equipment ID are not available, generate a CER and get Ops Management approval prior to proceeding.	CHG G
			Electrical feeder list reviewed and included, if necessary?	
			Vent and drain information sheet included, if necessary?	
			Tech Spec Cross Reference List included, if necessary?	
			R&R written, if necessary?	
			Equipment Lineup Request, Included if needed:	
			Special conditions/equipment lineup changes (including welding boundary purge path)?	
			SOP, EMP, MMP, or other procedure used to coordinate work?	
			Tagout requires coordination between groups (including contact)?	
			Shift Test Specialist notified of any work that may require Fire Protection related compensatory actions.	
			Emergent or FIN Team Work Review Requirements	CHG G
C03 →			This work reviewed for impact on the integrity of the Control Room Pressure Boundary. If the Control Room boundary is impacted: • Are compensatory measures provided by Engineering Services? or • Already provided by the applicable procedure?	CHG C
C01 →			Tagout involves risk (include management approval)?	CHG G
			~~Reviewed for Reactivity Management~~ and stamped, if necessary?	
			Reviewed for Maintenance Rule and stamped, ~~if necessary?~~	
			EOOS Assessment: **Moderate** ☐ **Elevated** ☐ **High** ☐ (Att. II, Ops signoff)	CHG G
			~~Retest reviewed?~~	
			GTP-214 reviewed?	
			GTP-702 reviewed?	
			Shop sign-on sheet included in package?	

☐ Remarks (if checked see attached) CHG G

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS
COPY NO.____________

STATION SCHEDULING PROCEDURE

SSP-001

PLANNING AND SCHEDULING ON-LINE MAINTENANCE ACTIVITIES

REVISION 16

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE

INFORMATION USE

Procedure May Be Performed from Memory.
User Retains Accountability for Proper Performance.

6.0 RESPONSIBILITIES

6.1 Alara Representative

6.1.1 Reviews WO and ensures dose and area contamination reduction are evaluated during assignment of SRWP, RWP and coverage requirements

6.2 Engineering Services

6.2.1 Provides previews for ECR, Building Services, Corrective Maintenance and Engineering program support WO's when requested.

6.3 Occupational Risk Management Planning Representative

6.3.1 Assists in the planning of all work package plans where safety questions or concerns arise.

6.3.2 Develop Safety Plans and store them in the Safety Notebook.

6.4 Operations Scheduling Supervisor or SRO

6.4.1 Validates priorities and verifies approval of WO.

6.4.2 Determines Risk Level and Equipment Impacts and ensures they are evaluated, used FID to support the evaluation.

6.4.3 Performs an EOOS and Integrated Risk Assessment of planned activities.

6.4.4 Ensures Control room Pressure Boundary components are controlled by SAP-603 requirements.

6.5 Planning Supervisor

6.5.1 Ensures that adequate time, technical resources and support services are readily available to planning personnel to ensure package quality and schedule adherence is maintained.

6.5.2 Ensures that work packages are reviewed for quality and technical content that are consistent with station requirements and expectations.

6.5.3 Ensures that areas of responsibility for overall implementation of work package and scheduling meet milestone expectations.

7.7.7 The Operations Scheduling Unit will identify activities that incur elevated risk per EOOS and OAP-102.1.

7.8 Main Control Board Discrepancy/Distraction WO Administrative Controls.

7.8.1 When a Main Control Board Discrepancy is identified, a Maintenance Work Request (WO) will be generated and assigned a priority of 2 or higher.

A. An Out-of-Service (OOS) sticker will be placed on the appropriate item or component on the MCB to identify the WO.

B. The discipline planner will change the MCB field (impact) to a "1" for tracking purposes.

C. Priority WO's will be worked in accordance with SAP-601.

7.8.2 If an MCB WO is used to bypass an annunciator window until the repair can be made to the channel, it will be assigned an MCB “2” for tracking purposes.

7.8.3 When a Main Control Board Distraction is identified, a WO will be generated and the priority will be left blank.

A. An Out-of-Service (OOS) sticker will be placed on the appropriate item or component on the MCB to identify the WO.

B. The Screening Committee will assign the WO a priority.

C. The discipline planner will change the MCB field (impact) to a "3" for tracking purposes.

D. MCB WO's will be processed as follows:

1. MCB WO's will be planned within the following week. The WO should be scheduled to work within 2 weeks of initiation or the next train week if applicable, provided parts are available and plant conditions allow.

2. If work on associated equipment is planned and scheduled to start within approximately T-4 weeks of initiation of the MCB WO, then the work can be scheduled with the other planned activities, subject to the Operations Scheduling Representative concurrence.

7.12 Risk Assessment of On-Line Maintenance Activities

7.12.1 When the Plan of the Week is completed, normally at T-2 weeks, an assessment of plant risk will be performed by the WWM using the Equipment Out Of Service (EOOS) software and Integrated Risk Assessment. EOOS validation of the Plan of the Week freezes the schedule.

7.12.2 If any schedule change is made to risk significant equipment the assessment will be re-performed. This assessment will have the following:

A. The scheduler's screen of EOOS will reflect the proposed plant system line up.

B. The risk significant equipment will be shown as unavailable in EOOS using the duration and start times from the Plan of the Week Schedule.

7.12.3 The results of EOOS assessment's of plant risk will be communicated by the following:

A. The Supervisor, Operations Planning Group and the appropriate Work Week Manager will be informed of the results.

B. The results will be enclosed in the Plan of the Week agenda to be discussed at the T-1 Week meeting.

C. The results will be enclosed in the Plan of the Day with the Major Work Items List.

7.12.4 If any Urgent or Emergent Work Orders on risk significant equipment are generated the assessment will be re-performed. This assessment will have the following:

A. The scheduler's screen of EOOS will reflect the current plant system line up.

B. The risk significant equipment will be shown as unavailable in EOOS using the duration and start times from the Plan of the Week Schedule

7.12.5 If the results of the EOOS or Integrated Risk assessment indicate elevated risk level as identified by OAP-102.1, the Planning and Scheduling Department will do the following contingency planning activities:

A. Schedulers, Operations Schedulers, and WWM's will identify other plant configurations or modes where the maintenance could be performed with less risk.

WORK PACKAGE ORGANIZER

NOTE: If there is an R&R associated with this WPO - attach a copy of completed WPO to R&R.
This WPO impacts EOOS Risk Assessment Calculation **YES / NO**

MRF/ECR OPERABILITY REQUIRED YES / NO MRF/ECR NUMBER ASSOCIATION CODE/FEG_______________ PAGE 1 OF ____

SCHEDULED DATE ______________ SCHEDULED COMPLETION DATE ____________ TRAIN ________ WPO # __________

R&R # ________________ TAGOUT # ____________ FBRPSN # __________ EQUIPMENT # ____________ SYS _

SCHEDULING GROUP SECTION												PERFORMER SECTION *		OPERATIONS REVIEW		
Equipment ID	TASK #	RESP. GROUP	R & R REQ.		RED TAGS REQ		OPS RETEST REQ.		MAINT. RETEST		PROC. NO.	WORK COMPLETE OR READY FOR RETEST (signature)	RETEST * COMPLETE (signature)	OPS RETEST		SS, SE OR CRS REVIEW COMPLETE (signature)
			Y	N	Y	N	Y	N	Y	N				PROC	STTS	
Comments/Retest:																
Comments/Retest:																
Comments/Retest:																
Comments/Retest:																
Comments/Retest:																
Comments/Retest:																

* PERFORMER SIGNATURE INDICATES TASK IS COMPLETE OR READY FOR RETEST. WHEN RETEST IS COMPLETE, THE PERFORMER WILL SIGN THE RETEST COMPLETE COLUMN.

2. SPECIAL INSTRUCTIONS ____

3. RESPONSIBLE SUPERVISOR FOR WORK:ELECT ____________ MECH ____________ I&C ____________ CIVIL ____________
4. LEAD GROUP ______________________________

Question 9

9. 007A2.05 002

The Unit is shutdown in Mode 4.

- RCS pressure is 375 psig
- RCS temperature is 305 ^{o}F

The crew is performing GOP-6 to place RHR Train A in service for cooldown and to establish RHR Train B as the protected train. When MVG-8701A and MVG-8702A, RCS LP A TO PUMP A, are opened, annunciator PRT LVL LO/TEMP/LVL/PRESS HI actuates.
Operators note the following:

- PRT Temperature: 105 ^{o}F
- PRT Level: 69%
- PRT Pressure: 12 psig

Which ONE of the following identifies the correct operator response?

A. This is an expected alarm and operators should continue placing the RHR Train A inservice, then restore the PRT parameters using SOP-101.
B. Operators should secure from placing the RHR Train A in service since PRT temperature limits have been exceeded, operators should cool the PRT using SOP-101.
C. Operators should secure from placing the RHR Train A in service since PRT level limits have been exceeded, operators should lower PRT level using SOP-101.
D. Operators should secure from placing the RHR Train A in service since PRT high pressure limits have been exceeded, operators should vent the PRT using SOP-101.

Feedback

DISTRACTORS:

A INCORRECT This is not an expected alarm.
B INCORRECT PRT LVL LO/TEMP/LVL/PRESS HI annunciator actuates when PRT temperature reaches 113 degrees F. Although temperature is somewhat elevated, it has not reached that temperature yet.
C INCORRECT PRT LVL LO/TEMP/LVL/PRESS HI annunciator actuates when PRT level reaches 83%. Although level is somewhat elevated, it has not yet reached that level.
D CORRECT PRT LVL LO/TEMP/LVL/PRESS HI annunciator actuated when pressure reached 8 psig.

REFERENCES:

1 AB-02, REACTOR COOLANT SYSTEM, rev 10, 04/18/02, page 49, fig AB2.4
2 SOP-101, REACTOR COOLANT SYSTEM, rev 25, 07/23/04, pages 70 - 74.
3 ARP-001, PANEL XCP-616, rev 6, page 29.

K/A CATALOGUE QUESTION DESCRIPTION:
- Pressurizer Relief Tank/Quench Tank System (PRTS); Ability to (a) predict the impacts of the following malfunctions or operatios on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Exceeding PRT high-pressure limits.

Question #9 (007A2.05 002)

Recommendation:

Delete the question.

Justification:

Normal parameters of the Pressurizer Relief Tank (PRT) would be similar to the following:
Pressure: 5#
Level: 60-70%
Temperature: 80-90F

Upset parameters given:
Pressure: 12#
Level: 69%
Temperature: 105F

PRT limits (as listed in the ARP-001-XCP-616.06B)
Pressure: 8#
Level: 83%
Temperature 113F

(Reference attached)

Operators are expected to know nominal operating parameters of systems; however, this question inappropriately requires a level of knowledge of specific alarm setpoints which are contained in Annunciator Response Procedures (ARP). Accordingly, this question is simply testing the examinees ability to identify which of the three PRT parameters (temperature, level, and pressure) are outside of its alarm setpoint. The information contained in the stem regarding the crew performing GOP-6 to place RHR Train A in service for cooldown and to establish residual heat removal (RHR) Train B as the protected train is information that is not needed to answer the question. Consequently, this contributes to an examinee considering information which is not applicable to answering the question.

This question should be deleted due to the psychometric problems with the stem and the level of knowledge, difficulty, and detail requiring the operator to memorize specific set points, sequence of events, parameters or conditions which are not immediate actions needed for safe operation of the plant.*

* See discussion in Section II.A.

PANEL XCP-616
ANNUNCIATOR POINT 4-4

PRT LVL LO/ TEMP/LVL/ PRESS HI	**SETPOINT:** 113°F (Hi Temp) 83% (Hi Level) 8 psig (Hi Press) 64% (Lo Level)	**ORIGIN:** ITB00471 ILB00470A IPB00472A ILB00470B

PROBABLE CAUSE:

1. Pressurizer relief or safety valves actuated or leaking.
2. Excess nitrogen pressure.
3. Tank leak or excessive pump down with RCDT pump.

C01→ 4. RHR suction relief valve lifted while the RHR Loop suction valves are unisolated.

AUTOMATIC ACTIONS:

1. The rupture disc will rupture at 90 psig.

CORRECTIVE ACTIONS:

1. Determine the cause of the alarm by monitoring PI-472, PRESS PSIG,TI-471, TEMP °F, and LI-470 , LEVEL %.
2. Determine if a Pressurizer PORV or safety has lifted or is leaking.
3. Determine if an RHR suction relief valve is lifting and which one it is.

SUPPLEMENTAL ACTIONS:

1. Refer to SOP-101 for PRT operations.
2. If necessary, switch RHR trains to isolate a leaking suction relief valve.

REFERENCES:

1. E-302-602.
2. B-804-616, Sh. 2.
3. B-208-082, RC-85.
4. SOP-101.

valves operate on 480 VAC power (see Table AB2.4 for a list of power supplies). All valves fail as-is on a loss of electrical power.

Each isolation valve has its own three position (CLOSE, neutral, OPEN) hand control switch on the MCB, which spring-returns to the midposition. Once the control switch is momentarily taken to OPEN, the motor energizes and rotates in the open direction. It continues to drive the valve open, even if the control switch is taken to CLOSE and held there, until the valve opens fully and the motor stops. Then the CLOSE position is enabled and the motor direction can be reversed to close the valve. Once CLOSE is momentarily selected, the OPEN position is disabled until the valve is closed fully and the motor is deenergized. This arrangement prevents shorting the motor windings through the open and close connectors.

Pressurizer Relief Tank (Figures AB2.4 and AB2.18)

Any steam or water discharged by the lifting of a PORV or code safety valve is passed to the Pressurizer Relief Tank (PRT) through a common 12-inch discharge line that serves all six valves (Figure AB2.4). The PRT condenses and cools any discharge from the PORVs or safety valves. Discharges from other relief valves inside the Reactor Building are also piped to the PRT, as is any discharge from the Reactor Vessel Head Vent System. Normally, the tank is partially filled with water at or near Reactor Building ambient temperature and contains a predominantly nitrogen atmosphere maintained at a pressure of 3-5 psig by a nitrogen pressure regulator. Sparging nozzles beneath the water surface discharge steam into the water volume (Figure AB2.18). The mixing that results condenses and cools the discharged steam. The PRT is equipped with a spray supplied from the Primary Makeup Water System, a drain to the suction of the Reactor Coolant Drain Tank (RCDT) pumps, and a connection to the waste gas header. There is a hole in the inlet line (from the Pressurizer) above the normal water level that allows nitrogen to the PRT to be used to place the RCS on a "PRT FLOAT" during cold shutdown.

PRT Design

The PRT has an internal volume of 1300 cubic feet and is constructed of austenitic stainless steel. The tank is sized to quench a postulated discharge of pressurizer steam equivalent to 110 percent of the volume above the programmed pressurizer water level for full power (60 percent of level span). The tank is not designed to accept a continuous discharge from the Pressurizer. Design temperature is 340°F; normal operating temperature is 120°F or less. The volume of water in the tank is capable of absorbing the heat from the postulated discharge, assuming an initial temperature of 120°F and increasing to 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by aligning the Reactor Coolant Drain Tank (RCDT) pumps to take a suction on the PRT drain connection. The liquid circulates through the RCDT heat exchanger and returns cooled to the PRT via the spray header. The Reactor Coolant Drain Tank and its pumps and heat exchanger are components of the Waste Disposal System.

PRT Rupture Discs

The tank is protected from a discharge exceeding the design value by two rupture discs which relieve to the Reactor Building atmosphere. The rupture discs have a combined capacity of 1.6 million lbm/hr, saturated steam. This exceeds the combined capacity of the pressurizer safety valves. The tank design pressure of 100 psig is twice the calculated pressure resulting from the maximum safety valve discharge described above. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added. The rupture discs are set to release within the range of 86-100 psig (nominal release pressure is 91 psig).

The discharge line from the code safety valves (and PORVs) to the PRT is large enough to prevent back pressure at the safety valves from exceeding 20 percent of the setpoint pressure (2485 psig) at full flow. (The maximum back pressure expected during discharge is 350 psig.)

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

SYSTEM OPERATING PROCEDURE

SOP-101

REACTOR COOLANT SYSTEM

REVISION 26

SAFETY RELATED

Original Signed By: S. Lathren
DISCIPLINE SUPERVISOR

08/05/05
DATE

Original Signed By: R. F. Ray
APPROVAL AUTHORITY

08/08/05
DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	11/02/05					

CONTINUOUS USE

Continuous Use Of Procedure Required.
Read Each Step Prior To Performing.

SYSTEM INFORMATION

5. The Pressurizer is a vertical, cylindrical vessel. Electric heaters are installed through the bottom head while the spray nozzle, power operated relief valves and safety valves are connected to the top head of the vessel.

a.	Safety valve set pressure	2485 psig
b.	Power operated relief valve capacity	210,000 lb/hr each
c.	Heaters	18 KW/heater

1)	Control Group	21 Heaters	376 KW
2)	Backup Group A	30 Heaters	540 KW
3)	Backup Group B	27 Heaters	484 KW
	Total	78 Heaters	1400 KW

6. The Pressurizer Relief Tank is a horizontal, cylindrical vessel. Steam from the Pressurizer safety and relief valves is discharged into the tank through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The PRT has a rupture disc to release excess pressure in the tank and prevent tank damage.

a.	Tank volume	1,300 ft^3
b.	Rupture disc release pressure	90 psig
c.	Total rupture disc release capacity	1.6×10^6 lb/hr

7. Removing one Pressurizer Safety Valve creates an opening in the RCS of 21.15 square inches. A safety valve is a 6 inch valve with inlet piping of 5.19 inches inside diameter. Removing one safety valve adequately meets the 2.7 square inch RCS opening required per Technical Specification 3.4.9.3.

K. VENTING THE PRT TO THE WASTE GAS SYSTEM VIA THE RCDT

1.0 INITIAL CONDITIONS

☐ 1.1 Waste Gas System is in operation per SOP-119.

2.0 INSTRUCTIONS

☐ 2.1 Close XVD07920-WG, PZR RELIEF TANK NITROGEN SUP ISOL VLV (FB-412).

☐ 2.2 Open XVD08025-RC, PRT VENT HDR ISOL TO WPS (FB-412).

☐ 2.3 Close XVD07154-WL, RCDT HYDROGEN INLET HEADER VALVE (FB-412).

CAUTION 2.4

XVT00100-WG, RCDT/PRT VENT CONN ISOL/THROTTLING VLV, must be opened slowly to prevent tripping the operating Waste Gas recombiner on high hydrogen concentration.

☐ 2.4 Throttle open XVT00100-WG, RCDT/PRT VENT CONN ISOL/THROTTLING VLV (FB-412).

NOTE 2.5

RCDT pressure should be maintained between 3 psig and 5 psig at all times.

☐ 2.5 Monitor PI-472, PRT PRESS PSIG, and adjust XVT07152-WL, RCDT GAS HEADER PRESSURE CONTROL VALVE (FB-412), as necessary to lower PRT pressure to desired pressure.

☐ 2.6 When a sufficient PRT purge has been completed or conditions exist to secure the venting, perform the following (FB-412):

☐ a. Adjust XVT07152-WL, RCDT GAS HEADER PRESSURE CONTROL VALVE, to 5 psig.

☐ b. Close XVT00100-WG, RCDT/PRT VENT CONN ISOL/THROTTLING VLV.

☐ c. Open XVD07154-WL, RCDT HYDROGEN INLET HEADER VALVE.

☐ d. Close XVD08025-RC, PRT VENT HDR ISOL TO WPS.

☐ e. Open XVD07920-WG, PZR RELIEF TANK NITROGEN SUP ISOL VLV.

END OF SECTION

L. FILLING AND VENTING THE PRT TO A POLY BOTTLE

1.0 INITIAL CONDITIONS

☐ 1.1 A **Pre-Job Brief** has been conducted per OAP-100.3.

☐ 1.2 The PRT is in a normal lineup per SOP-101.

☐ 1.3 The Reactor Make-up Water system is in service per SOP-106.

☐ 1.4 The PRT has been vented to atmosphere.

NOTE 2.0
a. PRT level should not exceed 95%.
b. PRT pressure should not exceed 60 psig.

2.0 INSTRUCTIONS

☐ 2.1 Install a poly bottle on XVD00007-RC, UPSTREAM TEST VENT ISOL FROM XVD08033-RC (FB-412).

2.2 Isolate the PRT as follows (FB-412):

☐ a. Close XVD08025-RC, PRT VENT HDR ISOL TO WPS.

☐ b. Set XVT08034-RC, PRT N2 PRESSURE REGULATOR, to 0 psig.

2.3 Raise level in the PRT to between 90% and 95% as follows:

☐ a. Close XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPES MU SUP VLV (FB-412).

b. Open the following valves:

☐ 1) PVD-8028, PRT RMWST MU.

☐ 2) PVD-8030, PRT RMWST MU.

Step 2.3 continued

NOTE 2.3.c
Nonessential water flow in excess of 170 gpm will isolate PVD-1920A and PVD-1920B, MU WTR TO NON-ESSEN LOADS.

☐ c. Throttle open XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPES MU SUP VLV (FB-412), to obtain the desired fill rate.

☐ d. When PRT level reaches between 90% and 95%, close XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPES MU SUP VLV (FB-412).

NOTE 2.4
HP and Chemistry should be contacted for a gas sample prior to venting the second time.

2.4 If venting the PRT for the first time proceed to Step 2.5, otherwise, sample the PRT oxygen concentration as follows:

☐ a. Ensure PRT pressure is greater than the Shutdown Waste Gas Decay Tank to which the sample gas will be directed.

☐ b. Have Chemistry personnel install a sample vessel for the PRT.

☐ c. Open one of the following valves (AB-388):

1) XVD07886A-WG, GAS DECAY TANK G OUTLET ISOLATION VALVE.

2) XVD07886B-WG, GAS DECAY TANK H OUTLET ISOLATION VALVE.

☐ d. Open XVD07935-WG, PZR RELIEF TANK NITROGEN SUPPLY VALVE (AB-388).

☐ e. Open XVT08091-RC, PRT INLET ISOL TO SAMPLE VESSEL (FB-412).

Step 2.4 continued

☐ f. Start sample flow through the sample vessel by throttling open XVT08036-RC, PRT OUTLET ISOL TO SAMPLE VESSEL (FB-412).

☐ g. When directed by Chemistry personnel, close XVT08036-RC, PRT OUTLET ISOL TO SAMPLE VESSEL (FB-412).

h. Close the following valves:

☐ 1) XVT08091-RC, PRT INLET ISOL TO SAMPLE VESSEL (FB-412).

☐ 2) XVD07886A-WG, GAS DECAY TANK G OUTLET ISOLATION VALVE (AB-388).

☐ 3) XVD07886B-WG, GAS DECAY TANK H OUTLET ISOLATION VALVE (AB-388).

☐ 4) XVD07935-WG, PZR RELIEF TANK NITROGEN SUPPLY VALVE (AB-388).

☐ 2.5 Throttle open XVD00007-RC, UPSTREAM TEST VENT ISOL FOR XVD08033-RC (FB-412).

☐ 2.6 After the PRT is vented, close XVD00007-RC, UPSTREAM TEST VENT ISOL FOR XVD08033-RC (FB-412).

☐ 2.7 Lower PRT level to between 60% and 65% per SOP-108.

☐ 2.8 Repeat Steps 2.2 through 2.7 until oxygen Concentration in the PRT is less than 5%.

2.9 When PRT venting is complete, perform the following:

☐ a. Remove the poly bottle installed on XVD00007-RC, UPSTREAM TEST VENT ISOL FROM XVD08033-RC (FB-412).

☐ b. Cap XVD00007-RC, UPSTREAM TEST VENT ISOL FOR XVD08033-RC (FB-412).

Step 2.9 continued

c. Close the following valves:

☐ 1) PVD-8028, PRT RMWST MU.

☐ 2) PVD-8030, PRT RMWST MU.

☐ d. Open XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPES MU SUP VLV (FB-412).

☐ e. Set XVT08034-RC, PRT N2 PRESSURE REGULATOR (FB-412), to between 3 psig and 5 psig.

END OF SECTION

M. FILLING AND VENTING THE PRT TO THE WASTE GAS SYSTEM

1.0 INITIAL CONDITIONS

☐ 1.1 A **Pre-Job Brief** has been conducted per OAP-100.3.

☐ 1.2 No gaseous release is in progress.

☐ 1.3 Gas Decay Tanks G and H are not in service.

☐ 1.4 Waste Gas is in a normal lineup per SOP-119.

NOTE 1.5
To avoid sending a large volume of oxygen to the Waste Gas System, it is desirable to raise PRT level initially to between 90% and 95% while it is still aligned to the Outage Vent System.

☐ 1.5 The PRT Outage Vent has been restored to normal per Section IV.

NOTE 2.0
a. PRT level should not exceed 95%.
b. PRT pressure should not exceed 60 psig.

2.0 INSTRUCTIONS

2.1 If required, fill the PRT as follows, otherwise proceed to Step 2.4 to align nitrogen to the PRT:

☐ a. Close XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPES MU SUP VLV (FB-412).

☐ b. Open PVD-8028, PRT RMWST MU.

☐ c. Open PVG-8030, PRT RMWST MU.

Step 2.1 continued

NOTE 2.1.d
Nonessential water flow in excess of 170 gpm will isolate PVD-1920A and PVD-1920B, MU WTR TO NON-ESSEN LOADS.

☐ d. Throttle open XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPE MU SUP VLV (FB-412), to obtain desired fill rate.

☐ e. When PRT pressure is greater than the selected Waste Gas shutdown tank pressure, proceed to Step 2.2.

NOTE 2.2
HP and Chemistry should be contacted for gas sample prior to venting.

2.2 Vent the PRT to Waste Gas as follows:

☐ a. Close XVD07920-WG, PZR RELIEF TANK NITROGEN SUP ISOL VALVE (FB-412).

☐ b. Open XVD07935-WG, PZR RELIEF TANK NITROGEN SUPPLY VALVE (AB-388).

☐ c. Open one of the following (AB-388):

1) XVD07886A-WG, GAS DECAY TANK G OUTLET ISOLATION VALVE.

2) XVD07886B-WG, GAS DECAY TANK H OUTLET ISOLATION VALVE.

☐ d. If a gas sample is not desired, open XVD08025-RC, PRT VENT HDR ISOL TO WPS (FB-412).

Step 2.2 continued

e. If a PRT gas space sample is required, perform the following (FB-412):

☐ 1) Have Chemistry personnel install a sample vessel.

2) Open the following:

☐ a) XVT08091-RC, PRT INLET ISOL TO SAMPLE VESSEL.

☐ b) XVT08036-RC, PRT OUTLET ISOL TO SAMPLE VESSEL.

☐ 3) Open the Sample Vessel Inlet and Outlet Valves.

☐ 4) Close XVD08025-RC, PRT VENT HDR ISOL TO WPS.

☐ 5) Close the Sample Vessel Outlet Valve.

6) Close the following:

☐ a) Sample Vessel Inlet Valve.

☐ b) XVT08091-RC, PRT INLET ISOL TO SAMPLE VESSEL.

☐ c) XVT08036-RC, PRT OUTLET ISOL TO SAMPLE VESSEL.

☐ 7) Open XVD08025-RC, PRT VENT HDR ISOL TO WPS.

☐ 8) Have Chemistry personnel remove the sample vessel.

2.3 When PRT filling is complete, perform the following:

☐ a. Close PVD-8028, PRT RMWST MU.

☐ b. Close PVD-8030, PRT RMWST MU.

☐ c. Open XVD01945-MU, PRT & RCPS #3 SEAL STANDPIPES MU SUP VLV (FB-412).

2.4 When PRT pressure is stable, align nitrogen to the PRT as follows:

☐ a. Ensure XVT08034-RC, PRT N2 PRESSURE REGULATOR (FB-412), is set to between 3 psig and 5 psig.

☐ b. Close XVD08025-RC, PRT VENT HDR ISOL TO WPS (FB-412).

☐ c. Close XVD07886A-WG, GAS DECAY TANK G OUTLET ISOLATION VALVE (AB-388).

☐ d. Close XVD07886B-WG, GAS DECAY TANK H OUTLET ISOLATION VALVE (AB-388).

☐ e. Close XVD07935-WG, PZR RELIEF TANK NITROGEN SUPPLY VALVE (AB-388).

☐ f. Open XVD07920-WG, PZR RELIEF TANK NITROGEN SUP ISOL VALVE (FB-412).

☐ g. Lower PRT level to between 60% and 65%, per SOP-108.

☐ 2.5 Repeat Steps 2.1 through 2.4 until oxygen Concentration in the PRT is less than 5%, as determined by gas sample.

☐ 2.6 When PRT oxygen concentration is less than 5%, ensure PRT level is 65% or greater.

END OF SECTION

Question 11

11. 007K1.01 002

A small break LOCA has occurred. Pressurizer PORVs are being used to reduce RCS pressure per EOP-2.1, "Post-LOCA Cooldown and Depressurization."

- Containment pressure is 14 psig.

Which ONE of the following represents the maximum pressure that could be reached inside the Pressurizer Relief Tank (PRT) before the PRT rupture disc ruptures?

A. 90 psig
B. 100 psig
C. 114 psig
D. 128 psig

Feedback

DISTRACTORS:

A INCORRECT — With the RB at atmospheric pressure, the rupture discs are set to release within the range of 86 - 100 psig (nominal release pressure is 90 psig). Plausible since this is the nominal pressure.

B INCORRECT — With the RB at atmospheric pressure, the rupture discs are set to release within the range of 86 - 100 psig (nominal release pressure is 90 psig). Plausible since this is the minimum pressure plus containment pressure and also the max pressure with normal containment parameters.

C CORRECT — With the RB at 14 psig, the rupture discs set pressure would subsequently be affected causing the release range to increase to between 100 - 114 psig (nominal release pressure is 104 psig).

D INCORRECT — With the RB at 14 psig, the rupture discs set pressure would subsequently be affected causing the release range to increase to between 100 - 114 psig (nominal release pressure is 104 psig). Plausible if the applicant tries to take into account psig and psia adding 14 psi.

REFERENCES:

1. AB-2, "Reactor Coolant System," page 46.

K/A CATALOGUE QUESTION DESCRIPTION:
Pressurizer Relief Tank/Quench Tank System (PRTS); Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: Containment system.

<u>Question # 11 (007K1.01 002)</u>

Recommendation:

Delete this question.

Justification:

This question is misleading and includes an inappropriate distractor in the question and the answer choices. The containment pressure given in the question stem and the multiple choice answers are all given in units of gauge pressure. Thus, an examinee could conclude that the answer is intended to address the maximum <u>indicated</u> pressure before the rupture disc ruptures (in this case, 100 psig). As explained further below, the indicated pressure will be impacted by the containment pressure.

Operators are expected to know the nominal release pressure (90 psig) of the Pressurizer Relief Tank (PRT) rupture disc: However, this question inappropriately requires a detailed knowledge of the <u>design</u> range (84-100 psig) of possible release pressure. This design range is an engineering issue, not an operationally important aspect of PRT indications/parameters. The K/A for this question does not require knowledge of this information. Accordingly, the question does not match the K/A.

Also, the question could have more than one correct answer. Choice C could be the correct answer for the reason already provided in the Feedback section. Specifically: With the RB at 14 psig, the rupture disc set pressure would subsequently be affected causing the release range to increase to between 100 - 114 psig (nominal release pressure is 104 psig). However, Choice B is also correct if the examinee reasoned that since the stem and choices were constructed using psig vs psia, the writer was referring to indicated pressure. Given that the bourdon tube supplying indicated PRT pressure is inside the Reactor Building (RB), it would be sensing a ΔP between the RB and the PRT. If RB pressure was 0 psig and the PRT pressure increased to 100 psig (upper limit of the release range with a nominal 90 psig setpoint), then maximum indicated PRT pressure would be 100 psig. If RB pressure were 14 psig and PRT pressure were 114 psig, then the bourdon tube would sense a ΔP of 100 psi and, again, maximum indicated pressure at release would be 100 psig. Therefore, regardless of existing pressure in the RB, when the rupture disc releases, the maximum <u>indicated</u> PRT pressure will be 100 psig.

This question should be deleted due to two reasons. The first reason is that it requires a detailed level of knowledge regarding the <u>design</u> range of the rupture disc that is not required per the K/A, and secondly, due to the terminology (psig vs psi) used in the question, the question can have more than one correct answer.

NUCLEAR OPERATIONS TRAINING

AUXILIARY BUILDING SYSTEM

AB-2

REACTOR COOLANT SYSTEM

REVISION 10

Recommended: Original signed by: Gerald Merchant **Date:** 4/18/02

Approved: Original signed by:Douglas O. Watson **Date:** 4/18/02

Senior Instructor Development

PRT Design

The PRT has an internal volume of 1300 cubic feet and is constructed of austenitic stainless steel. The tank is sized to quench a postulated discharge of pressurizer steam equivalent to 110 percent of the volume above the programmed pressurizer water level for full power (60 percent of level span). The tank is not designed to accept a continuous discharge from the Pressurizer. Design temperature is 340°F; normal operating temperature is 120°F or less. The volume of water in the tank is capable of absorbing the heat from the postulated discharge, assuming an initial temperature of 120°F and increasing to 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by aligning the Reactor Coolant Drain Tank (RCDT) pumps to take a suction on the PRT drain connection. The liquid circulates through the RCDT heat exchanger and returns cooled to the PRT via the spray header. The Reactor Coolant Drain Tank and its pumps and heat exchanger are components of the Waste Disposal System.

PRT Rupture Discs

The tank is protected from a discharge exceeding the design value by two rupture discs which relieve to the Reactor Building atmosphere. The rupture discs have a combined capacity of 1.6 million lbm/hr, saturated steam. This exceeds the combined capacity of the pressurizer safety valves. The tank design pressure of 100 psig is twice the calculated pressure resulting from the maximum safety valve discharge described above. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added. The rupture discs are set to release within the range of 86-100 psig (nominal release pressure is 91 psig).

The discharge line from the code safety valves (and PORVs) to the PRT is large enough to prevent back pressure at the safety valves from exceeding 20 percent of the setpoint pressure (2485 psig) at full flow. (The maximum back pressure expected during discharge is 350 psig.)

NUCLEAR OPERATIONS TRAINING

AUXILIARY BUILDING SYSTEM

AB-2

REACTOR COOLANT SYSTEM

LESSON PLAN

TIME: 2 HOURS

REVISION 10

Recommended: Original signed by: Gerald Merchant **Date:** 4/18/02

Approved: Original signed by: Douglas O. Watson **Date:** 4/18/02
Senior Instructor Development

e. Small air accumulator

Letter CGSS-0012-spe. If valve bumped off seat - should close block valve and full stroke PORV to assure proper reseating

1) Provides enough volume to overcome Spring & lift valve off seat instantly, but not full open (628 cc each)

8. Motor-operated isolation valves (MVG-8000 A/B/C)
   a. CLOSE, OPEN MCB
   b. 480 V MCC: 1DA2X, 1DB2X, 1DB2Y
9. Pressurizer relief tank (PRT)

Objective AB-2-09 Figure AB2.8

   a. Quenches discharges underwater
   b. Sized to quench discharge of 110% of steam volume above full load water level (60% of span)
   c. Design temperature 340°F; design pressure, 100 psig; rupture disc setting approximately 91 psig;
10. Combined rupture disc capacity exceeds combined capacity of safety valves
11. Service Connections
   a. N_2 atmosphere to prevent explosive H2-O2 mixture
   b. Makeup water
   c. Drain to liquid waste processing
   d. Connection to waste gas processing
   e. Connection to PASS

Objective AB-2-19

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ____________

EMERGENCY OPERATING PROCEDURE

EOP-2.1

POST-LOCA COOLDOWN AND DEPRESSURIZATION

REVISION 12

SAFETY RELATED

Original signed by Baker — 11/14/03
DISCIPLINE SUPERVISOR — DATE

Original signed by Lippard — 11/15/03
APPROVAL AUTHORITY — DATE

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
7 **Check if the SI System is in service:** • Any Charging Pump is running with flow indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. ☐ OR • Any RHR Pump is running in the SI Mode. ☐	7 IF SI has been terminated, THEN **GO TO Step 13.** ☐

NOTE - Step 8

If no RCP is running, the Reactor Vessel Head Upper Plenum may void during depressurization resulting in a rapidly increasing PZR level.

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
8 **Depressurize the RCS to refill the PZR:**	
a. Establish Normal PZR Spray: • Using RCP A: 1) Open PCV-444D, PZR SPRAY. ☐ 2) Close PCV-444C, PZR SPRAY. ☐ OR • Using RCPs B AND C: 1) Open PCV-444C, PZR SPRAY. ☐ 2) Close PCV-444D, PZR SPRAY. ☐	a. Cycle one PZR PORV as necessary to depressurize the RCS. ☐
b. Verify PZR level is GREATER THAN 30% [50%]. ☐	b. WHEN PZR level is GREATER THAN 30% [50%], THEN **COMPLETE Step 8.** ☐ **CONTINUE WITH Step 9.** Observe the **CAUTION** and **NOTE** prior to **Step 9.** ☐
c. Stop RCS depressurization. ☐	

Question 21

21. 022A1.04 002

Given the following:

- The plant is operating at 75% reactor power.
- The Reactor Building sump was pumped down to the Waste Holdup Tank twenty minutes ago.

Which one of the following may provide an alarm for a 0.7 GPM leak from the reactor coolant system to the Reactor building?

A. Reactor Building Sump level.
B. Reactor Building Radiation level.
C. Reactor Building Temperature.
D. Reactor Building Cooling Unit condensate drain flow.

Feedback

DISTRACTORS:

A INCORRECT Sump level would provide indications of leaks >10GPM or > 1 GPM
B INCORRECT Radiation level will cause an alarm when leakage exceeds 1 gpm for one hour
C INCORRECT Temperature may not increase until leakage is excessive.
D INCORRECT Condensate drain flow will alarm when condensed leakage exceeds 0.5 GPM.

REFERENCES:

1. GS-7, "Leak Detection."

K/A CATALOGUE QUESTION DESCRIPTION:
- Containment Cooling (CCS); Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Cooling water flow.

Question #21 (0022A1.04 002)

Recommendation:

Delete the question.

Justification:

The given answer, Choice D, is incorrect for the following reasons:

The Reactor Building Cooling Units (RBCU) condensate drains high flow alarm *does* actuate at 0.5 gpm; however, assuming that both trains of RBCUs equally condensed ½ of the given 0.7 gpm reactor coolant system (RCS) leak (which is a reasonable assumption), then the actual condensate drain flow for each train of RBCUs would be 0.35 gpm. This is below the alarm setpoint. Additionally, a portion of the given 0.7 gpm leak may impinge on other equipment/components in the RB and drain into the RB sump, and would not contribute to the drain flow from either of the trains.

The distractors; Choices A, B, and C, are incorrect for the reasons already stated in the Feedback section.

Also note that this question does not match the selected K/A in that it does not relate to "exceeding design limits," nor does it relate to containment cooling "CCS [Containment Cooling System] controls: cooling water flow." The condensate return flow from the RBCU is not the cooling water *to* the RBCU, which is what the K/A applies to. This item also does not match the K/A, because it does not measure whether changes can be predicted in cooling water flow from the operation of system controls. Rather, this item tests a failure of RCS boundary and effect on RBCU drain flow.

The question should be deleted because none of the choices are the correct answer.

NUCLEAR OPERATIONS TRAINING

GENERAL SYSTEMS

GS-7

LEAK DETECTION SYSTEM

REVISION 8

Recommended:	Original signed by Rusty Quick	Date:	11/18/02
Approved:	Original signed by Douglas O. Watson Senior Instructor Development	Date:	02/05/03

TRAINING MATERIAL ERRATA SHEET

GS-7, Revision 8

TDB Commitment Number: TMD*04-GS-7

Significant changes to lesson text:

Add the following objectives to GS-7:

GS-7-18, DISCUSS, in detail, all the recommendations of SOER(s) 85-5 as they pertain to operator training.

GS-7-19, DESCRIBE how SOER(s) 85-5 applies to your job (the cause) and/or its significance to plant operations (the effect).

GS-7-20, DETERMINE which of the seven Human Performance tools could have been used to prevent or mitigate the events of SOER(s) 85-5.

GS-7-21, Using selected operating experiences related to this course, DESCRIBE their applicability to your job (the cause), their significance to plant operations (the effect), and which of the seven Human Performance tools could have been used to prevent or mitigate the events.

Add the following documents to the references in both the handout and lesson plan:

SEN-184, Fire System Water-Hammer Results in Flooding of ECCS equipment

SOER 85-1, Reactor Cavity Seal Failure

SOER 85-5, Internal Flooding of Power Plant Buildings

CER-00-1171, CRDM Cooling System Leakage Through Relief Valves During STP.

CER-03-3989, SW Relief Valve Leakage Due to Water-Hammer During STP.

Approved: Original signed by Douglas O. Watson Date: 2/21/05
Program Lead Instructor

Forward Copy to Document Control

received, control room personnel determine the actual leak rate by using the IPCS and STP-114.002.

If known NON-RCS leakage is entering the sump, the IPCS constant can be adjusted to alarm only on an <u>increase</u> in the inleakage rate (ex - with ½ gpm of Industrial Cooling leakage, the alarm constant could be set to 1.5 gpm).

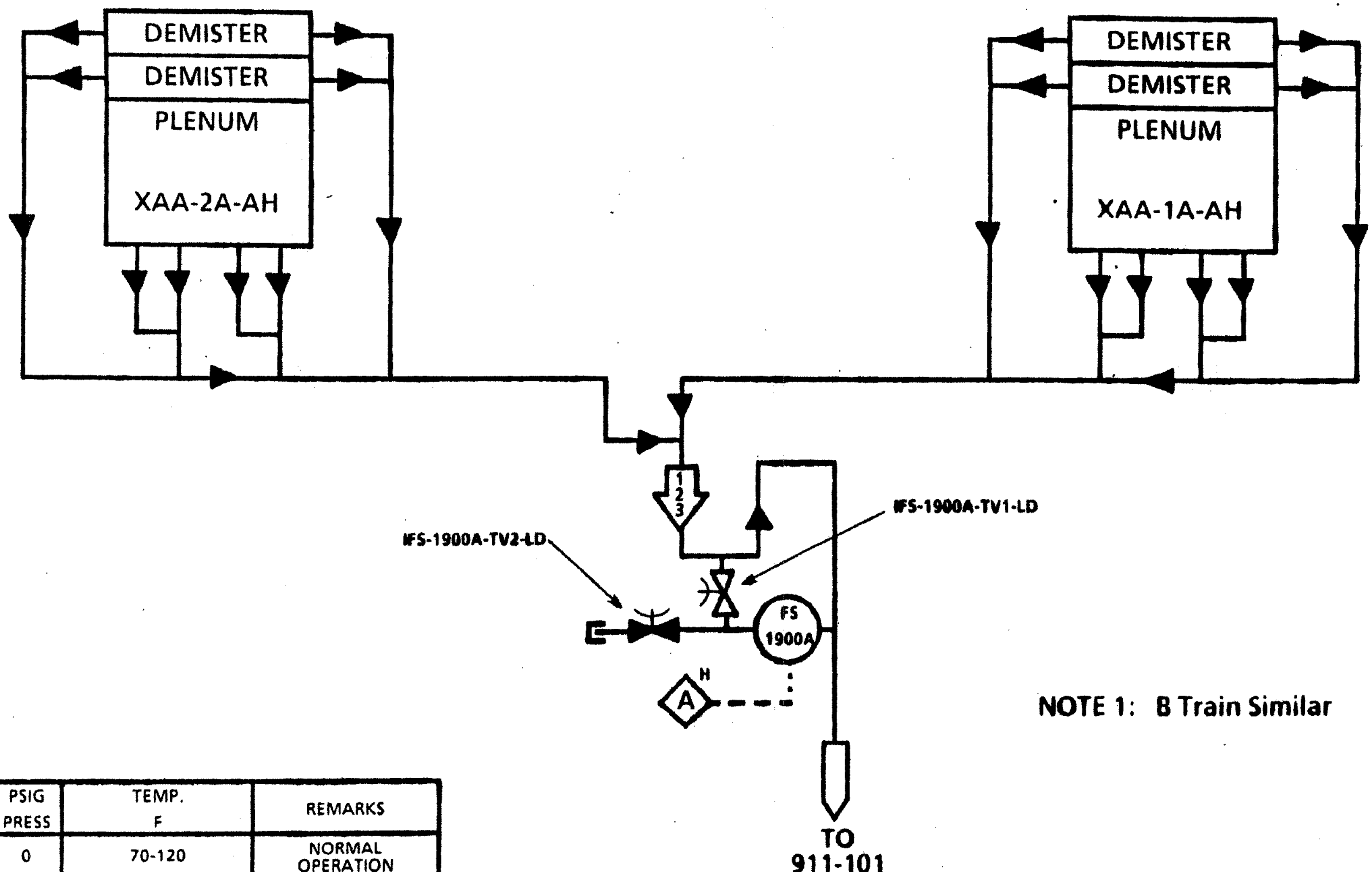
The Reactor Building sump level rising above 402'0" (LS-1963, -1964) actuates a 10-minute timer. If level reaches 402'6" before the timer expires, a "LEAK DETECT REACTOR BUILDING SUMP A/B LEAKAGE > 10 GPM annunciator" actuates on XPN-6032, -6033. Any alarm on XPN-6032 (Figure GS-7.12) actuates a "LD TRBL RB INCORE SMP LVL HI" on the MCB. Additionally, any alarm on XPN-6033 (Figure GS7.13) actuates "LD RB SMP LVL HI" annunciator on the MCB. If the level rises to 405'0" (LS-1963, -1964), a "REACTOR BUILDING SUMP A/B LEVEL HIGH" annunciator actuates on XPN-6032, -6033.

In response to these alarms, plant procedures direct control room personnel to determine the source of the leakage. The sump is pumped down to the Floor Drain Tank by opening the two containment isolation valves and starting either sump pump (XPP-115A or B). These pumps will automatically stop at 401'9". The leak rate determination will not be valid for 15 minutes after the sump has been pumped. Leak rates from LT-1963 and -1964 and the current leak detection setpoint can be obtained from the IPCS by displaying group SUMSUM.

<u>Reactor Building Cooling Units Drain Flow</u>

Other than activity detection, one of the most sensitive leakage detection systems is the Reactor Building cooling unit condensate drain flow alarm (Figure GS7.2). Flow switches (FS-1900 A/B) located in the common condensate drain header from the Reactor Building cooling units actuate annunciator "RBCU 1B/2B (1A/2A) DRN FLO HI" in the control room if flow rate exceeds 0.5 gpm. A backup to this detection method is

REACTOR BUILDING COOLING UNIT DRAINS



NOTE 1: B Train Similar

	GPM	PSIG PRESS	TEMP. F	REMARKS
1	< .5	0	70-120	NORMAL OPERATION
2	100	30	70-120	COOLING COIL TUBE RUPTURE
3	230	50	283	DBA CNDS REMOVAL

Figure GS7.2

Question 26

26. 026AA1.01 002

Plant conditions are as follows:

- A plant event occurred simultaneously with a Component Cooling System malfunction.
- ALL ESF and ALL Reactor Coolant pumps are running.
- Component Cooling System temperature is going UP.
- RCP A seal water outlet temperature is 230 °F and increasing very slowly.
- CHG pump A oil cooler outlet temp is 148 °F and increasing very slowly.
- RHR pump A seal water heat exchanger temperature is 185 °F and increasing very slowly
- Spent Fuel Pool temperature is 122 °F and increasing very slowly

Which ONE of the above components has exceeded a MAXIMUM temperature LIMIT per AOP-118.1, Total Loss of Component Cooling Water, or System Operating Procedures?

A. RCP A
B. CHG pump A
C. RHR pump A
D. Spent Fuel Pool

Feedback

DISTRACTORS:

A Incorrect. The maximum temperature limit for RCP seal water outlet is 235°F per AOP-118.1, Step 1 caution. Plausible because the maximum temperature limit for RCP lower seal water bearings is 225°F.

B Incorrect. The maximum temperature for CHG pump oil cooler outlet is 150°F per AOP-118.1, Total Loss of Component Cooling Water, Attachment 3. The running Charging pump would be secured within ONE (1) minute of losing CCW, so it is realistic to believe that oil temperature may be decreasing

C Incorrect. Could not find a temperature limit associated with RHR pump seal water HX in SOP-115, Residual Heat Removal or AOP-118.1 (Step 17). Note that CCW Loop A Essen Load Temp Hi does not alarm on high CC temp until 205°F. Plausible because other setpoints, below 185°F, are associated with this alarm.

D Correct. The SFP administrative temperature limit per SOP-123 is 120°F. Plausible because the CCW supply temp to the SFP HX is limited to 105°F during refueling operations.

REFERENCES:

1 AOP-118.1, Total Loss of Component Cooling Water.
2 SOP-123, Spent Fuel Cooling System
3 SOP-115, Residual Heat Removal
4

<u>Question #26 (026AA1.01 002)</u>

Recommendation:

Delete the question.

Justification:

This question involves recognition of a temperature limit being exceeded following a loss of component cooling capacity during a plant engineered safety features (ESF) actuation.

Upset parameters given:
'A' RCP seal water outlet – 230°F and rising
'A' CHG pump cooler outlet – 148°F and rising
'A' RHR seal water heat exchanger – 185°F and rising
Spent Fuel Pool – 122°F and rising

Component limits:
'A' RCP seal water outlet – 235°F
'A' CHG pump cooler outlet – 150°F
'A' RHR seal water heat exchanger – 205°F
Spent Fuel Pool – 120°F

This question is inappropriately testing the operator's memorization of alarm/limits on equipment that all have alarms or compensatory monitoring in accordance with procedures in these given conditions.*

The monitoring of key ESF equipment would be the primary focus of the control room staff upon an ESF actuation. Secondary responsibilities would remain with the crew responding to various annunciators using the appropriate Annunciator Response Procedures (ARP) as alarms are received.

The Spent Fuel Pool temperature limit of 120°F is not addressed in AOP-118.1, as the question suggests. This limit is addressed only in SOP-123, Spent Fuel Cooling, as a step when placing a loop of SF Cooling in service. Response to this alarm would be in accordance with the ARP.

The RHR pump seal cooler alarm limit as addressed in this question also is not specifically referenced in AOP-118.1. Step 17 of AOP-118.1 requires monitoring RHR Pump temperatures on specific moduflash points. This is in turn requires locating the appropriate alarm on XCP-601 2-4, which then lists the alarm setpoint and instrument.

The Charging Pump temperature setpoint is only addressed on Attachment 1 of AOP-118.1 after Alternate Cooling has been established. At this point, the building operator would be monitoring pump setpoints locally per this attachment.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Component Cooling Water (CCW); Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: CCW temperature indications.

The reactor coolant pump Seal Water Outlet temperature setpoint of 235°F is addressed in the AOP. But this point is not required for memorization per OAP-103.4 guidelines.

In addition, this question does not match the K/A as it does not examine the impact a loss of component cooling water (CCW) has on CCW temperatures. Instead, it tests the impact that a loss of CCW has on the Spent Fuel Pool temperature.

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ____________

ABNORMAL OPERATING PROCEDURE

AOP-118.1

TOTAL LOSS OF COMPONENT COOLING WATER

REVISION 2

SAFETY RELATED

Original signed by Hitt A. Crider — 10/2/00
DISCIPLINE SUPERVISOR — DATE

Original signed by Dan Gatlin — 10/11/00
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLED DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLED DATE

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing.

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION

OPERATOR ACTIONS

CAUTION

- Any running Charging Pump should be stopped within one minute of a loss of Component Cooling Water flow to its oil coolers.
- Any running Reactor Coolant Pump should be stopped if any of the following conditions exist:

  a. Component Cooling Water flow to the motor bearing coolers can NOT be restored within ten minutes.

  b. Motor Bearing temperature exceeds 195°F.

  c. Lower Seal Water Bearing temperature exceeds 225°F.

  d. Seal Water Outlet temperature exceeds 235°F.

NOTE

If a Reactor trip occurs, this procedure should be continued concurrently with the recovery actions of **EOP-1.0, REACTOR TRIP/SAFETY INJECTION ACTUATION.**

1 **Determine the cause for the loss of CCW:**

a. Check for annunciators on XCP-601, 602, and 603. ☐

b. **REFER TO the appropriate ARPs.** ☐

c. Attempt to correct the cause for loss of CCW. ☐

2 **Establish either train of CCW as the Active Loop. REFER TO SOP-118, COMPONENT COOLING WATER.** ☐

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
*16 **Check if RCS temperatures are stable OR decreasing.** ☐	*16 **REFER TO** the applicable AOP for present plant conditions: • **AOP-115.3, LOSS OF RHR WITH THE RCS INTACT.** ☐ • **AOP-115.4, LOSS OF RHR WHILE REFUELING.** ☐ • **AOP-115.5, LOSS OF RHR WITH THE RCS NOT INTACT (MODE 5).** ☐
*17 **Monitor RHR Pump temperatures on the following moduflash points:** • TM-7246, RHR PP A SEAL (Moduflash M3/SW Pt #4). ☐ • TM-7256, RHR PP B SEAL (Moduflash M4/SW Pt #4). ☐	
*18 **Monitor other CCW System loads:** a. Monitor the temperatures of other operating components cooled by the CCW System: • Spent Fuel Pool. ☐ • RCDT. ☐ • Waste Gas Compressors. ☐ • Hydrogen Recombiners. ☐ • Sample Coolers. ☐ • Recycle Evaporator. ☐ • Waste Evaporator. ☐ • Excess Letdown Heat Exchanger. ☐ b. At Shift Supervisor discretion, remove loads from service as necessary to prevent equipment damage. **REFER TO the appropriate system SOPs.** ☐	

CHARGING PUMP TEMPERATURE MONITORING

COMPONENT/ LOCATION	INSTRUMENT	EXPECTED TEMP (°F)	MAX TEMP (°F)	TIME ____	TIME ____	TIME ____	TIME ____	TIME ____	TIME ____
XPP0043A CHARGING/SI PUMP A	ITI17550A CHG/SI PP A GEARBOX LUBE OIL TEMP IND	115-145							
	ITI07551 PUMP A OIL CLR OUTLET	120-150	150						
(AB-388)	ITI07552 THRUST BRG TEMP	130-155							
	ITI17555A CHARGING/SI PUMP A OIL RSVR TEMP IND								
XPP0043B CHARGING/SI PUMP B	ITI17550B CHG/SI PP B GEARBOX LUBE OIL TEMP IND	115-145							
	ITI07561 PUMP B OIL CLR OUTLET	120-150	150						
(AB-388)	ITI07562 THRUST BRG TEMP	130-155							
	ITI17555B CHARGING/SI PUMP B OIL RSVR TEMP IND								
XPP0043C CHARGING/SI PUMP C	ITI17550C CHG/SI PP C GEARBOX LUBE OIL TEMP IND	115-145							
	ITI07571 PUMP C OIL CLR OUTLET	120-150	150						
(AB-388)	ITI07572 THRUST BRG TEMP	130-155							
	ITI17555C CHARGING/SI PUMP C OIL RSVR TEMP IND								

CCW LOOP A ESSEN LOAD TEMP HI M3	**SETPOINT:**	**ORIGIN:**
	155°F	TS-7038 (alarm input only)
	205°F	TM-7246
	205°F	TM-7037
	120°F	TS-7054 (PID-TD9004)
	205°F	TM-7052

PROBABLE CAUSE:

1. Insufficient Service Water flow through the Component Cooling heat exchanger.
2. Leak from the Reactor Coolant System to Component Cooling in the RHR heat exchanger in conjunction with a Component Cooling Surge Tank high level and RM-L2A or 2B alarm.
3. Leak from the Reactor Coolant System to Component Cooling in the RHR pump cooler in conjunction with a Component Cooling Surge Tank high level and RM-L2A or 2B alarm.
4. Insufficient Component Cooling flow through the RHR heat exchanger or RHR pump cooler.
5. Valve misalignment.

AUTOMATIC ACTIONS:

1. None.

> **NOTE**
>
> This alarm has reflash capabilities.

CORRECTIVE ACTIONS:

1. Observe the appropriate temperature indicators to determine which high temperature is alarming.
2. Provide additional Service Water cooling through the Component Cooling heat exchanger as necessary.
3. Isolate leakage from the Reactor Coolant System to Component Cooling as required by isolating and securing equipment.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

SYSTEM OPERATING PROCEDURE

SOP-123

SPENT FUEL COOLING SYSTEM

REVISION 14

SAFETY RELATED

Original Signed By: D. A. Baker — 06/22/05
DISCIPLINE SUPERVISOR — DATE

Original Signed By: G. A. Lippard — 06/27/05
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	09/08/05					
B	P	11/02/05					

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing

III. NORMAL OPERATIONS

A. STARTUP OF SPENT FUEL COOLING LOOP A

1.0 INITIAL CONDITIONS

☐ 1.1 Valve lineup is complete per Attachment IA.

☐ 1.2 Electrical lineup is complete per Attachment IIA.

☐ 1.3 I&C is available to perform Instrument lineups per Attachment IIIA as system conditions permit.

☐ 1.4 Control Panel lineup is complete per Attachment IVA.

☐ 1.5 Component Cooling Water is aligned to Spent Fuel Heat Exchanger A per SOP-118.

2.0 INSTRUCTIONS

☐ 2.1 Verify Spent Fuel Pool level is in the skimmer (461'6") as indicated on LI-7431 or LI-7433, POOL LEVEL FEET.

☐ 2.2 Start XPP0032A-SF, SPENT FUEL PIT COOLING PUMP A (AB-412). **(PEER ✓)**

☐ 2.3 Throttle XVT06658-SF, SPENT FUEL HEAT EXCHANGER A OUTLET VLV (AB-388), to establish a flow of less than or equal to 1800 gpm as indicated on FI-7449, HX A FLOW GPM.

☐ 2.4 Throttle XVB09628A-CC, SPENT FUEL HT EXCH A CC WTR OUTLET VLV (AB-388), to maintain pool temperature less than 120°F as indicated on TI-7435, POOL TEMP °F.

END OF SECTION

PANEL XCP-609
ANNUNCIATOR POINT 1-3

SFP TEMP HI	SETPOINT: 120°F	ORIGIN: ITB07437

PROBABLE CAUSE:

1. Insufficient cooling for the quantity of Spent Fuel being stored in the Spent Fuel Pool.
2. Loss of Component Cooling Water to the Spent Fuel Heat Exchanger.
3. Valve misalignment.

AUTOMATIC ACTIONS:

1. None.

CORRECTIVE ACTIONS:

1. Determine Spent Fuel Pool temperature on TI-7435, TEMP °F, and TI-7437, TEMP °F.
2. Determine Spent Fuel Heat Exchanger flow on FI-7449, HX A FLOW GPM, or FI-7459, HX B FLOW GPM.

SUPPLEMENTAL ACTIONS:

1. Dispatch an operator to verify proper operation of the operating Spent Fuel Cooling loop.
2. Line up and start the standby Spent Fuel Cooling loop per SOP-123, if necessary.

REFERENCES:

1. B-804-609.
2. B-208-093, SF-10.
3. D-302-651.
4. SOP-123.

Question 29

29. 027AK1.02 002

The crew has just completed a 6% per hour power ascention from 75% to 90% rated thermal power. A Pressurizer Safety Valve has just been identified as leaking to the PRT.

Which ONE of the following describes the affect on RCS temperature and Pressurizer water temperature, associated with the above events? (Assume no operator actions.)

A. RCS temperature will initially rise from its steady state value. Pressurizer water temperature will initially rise from its steady state value.
B. RCS temperature will initially rise from its steady state value. Pressurizer water temperature will initially lower from its steady state value.
C. RCS temperature will initially lower from its steady state value. Pressurizer water temperature will initially rise from its steady state value.
D. RCS temperature will initially lower from its steady state value. Pressurizer water temperature will initially lower from its steady state value.

Feedback

K/A MATCH ANALYSIS

The K/A requires the testing of knowledge of operational implications of a PCS malfunction, which is accomplished with the safety valve leaking. The safety valve has a pressure control function to keep the RCS from encroaching on the RCS pressure safety limit. The K/A also requires testing the implications of expansion of liquids as temperature increases. Therefore, the K/A is met because the RCS temperature goes up, causing the liquid volume to expand. The safety valve leaks, reducing the pressure in the pressurizer. Both of these items cause an insurge into the pressurizer, which lowers the pressurizer temperature (operational implication).

ANSWER CHOICE ANALYSIS:

A. Incorrect. Xenon is burning out due to the power ascention, which will cause RCS temperature to increase. The rise in RCS temperature will cause the RCS volume to expand coupled with the Safety Valve leaking causes an above average insurge into the Pzr, which will drop the average water temperature in the Pzr initially beyond the heaters capacity to compensate.
B. Correct. See analysis for A above.
C. Incorrect. See analysis for A above.
D. Incorrect. See analysis for A above.

REFERENCES:

1. NONE

K/A CATALOGUE QUESTION DESCRIPTION:

027 Pressurizer Pressure Control System (PZR PCS) Malfunction Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases.

<u>Question # 29 (K/A 027.AK1.02 002)</u>

Recommendation:

Delete the question.

Justification:

The question states that a 6% per hour power ascension from 75% to 90% rated thermal power has been completed and that a Pressurizer Safety Valve has just been identified as leaking. The question then asks for the affects on RCS temperature and Pressurizer water temperature assuming <u>no operator actions</u>. At this point the plant is in a steady state condition at 90% power. Xenon is burning out due to the power ascension, but with no operator actions being taken, the rod control system will account for the addition of positive reactivity and maintain the reactor coolant system (RCS) temperature constant. Therefore, with no change in RCS temperature, the RCS volume will not change. Since each of the four choices include RCS temperature either increasing or decreasing, there are no correct answers.

Additionally, the question has a psychometric flaw which caused significant confusion of the examinees. The leaking safety valve does not necessarily cause "an above average insurge'" as suggested in the Feedback section. Since the leak rate was <u>not</u> provided in the stem, the question stem provides insufficient information for the examinee to determine the impact of the leaking safety valve. Given historical safety valve leakage at VCSNS, there would be NO impact. If an examinee's experience allowed him/her to apply VCSNS history, the examinee would dismiss the given information as having negligible impact. If the examinee did not apply VCSNS history, then the given information would introduce confusion as described in the next paragraph.

If an examinee makes an assumption that RCS temperature initially rises, then he/she would assume that an insurge would occur. With an insurge, it may be reasonable to assume that pressurizer water temperature would decrease. If an examinee assumed that the leaking safety would cause the insurge to be *larger*, then he/she may also assume that the pressurizer water temperature would decrease. In either case, the water temperature decreases and the leaking safety has no impact. If the examinee tries to consider the effect of the leaking safety on the operation of the Pressurizer Pressure Control System (PCS), then the examinee would not be able to reach a definitive conclusion. Even if a reasonable leak rate were given in the stem, the dynamics of the PCS keeping up with the leak could not be resolved, nor should an operator be expected to determine if the leak was beyond the capacity of the PCS.

This question does not match the K/A in that it does not involve "Pressurizer Pressure Control Malfunctions". A leaking safety valve is not a control malfunction. In addition, the safety valves are not part of the Pressurizer Pressure *Control* System. Rather, they are part of the Pressurizer over-pressure *protection* components. Additionally, the question is actually a GFE question, which simply examines thermodynamic and reactivity processes rather than the response of the control system.

5.2.2.2 Mounting of Pressure Relief Devices

The NSSS supplier provides protection pads for the pressurizer. The architect engineer designs the supports for the safety relief valves and the attached piping in accordance with the installation guidelines and suggested physical layout criteria provided by the NSSS supplier. The architect engineer determines the reactions on the valves and attached piping to limit the piping reaction loads to acceptable values.

5.2.2.3 Report on Overpressure Protection

The pressurizer is designed to accommodate pressure increases (as well as decreases) caused by load transients. The spray system condenses steam to prevent the pressurizer pressure from reaching the setpoint of the power operated relief valves during a step reduction in power level equivalent to 10% of full rated load.

The spray nozzle is located in the top head of the pressurizer. Spray is initiated when the pressure controlled spray demand signal is above a given setpoint. The spray rate increases proportionally with increasing compensated error signal until it reaches a maximum value. The compensated error signal is the output of PID (Proportional plus Integral plus Derivative) controller, the input to which is an error signal based on the difference between actual pressure and a reference pressure.

The pressurizer is equipped with power operated relief valves which limit system pressure for a large power mismatch, and thus prevent actuation of the fixed high pressure reactor trip. The relief valves are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power operated relief valves if excessive leakage occurs. The relief valves are designed to limit the pressurizer pressure to a value below the high pressure trip setpoint for all design transients up to and including the design percentage step load decrease with steam dump but without reactor trip.

Output signals from the pressurizer pressure protection channels are isolated and used for pressure control. These are used to control pressurizer spray and heaters and power operated relief valves. Pressurizer pressure is sensed by fast response pressure transmitters with a time response of better than 0.2 seconds.

In the event of a complete loss of heat sink, e.g., no steam flow, protection of the RCS against overpressure is afforded by pressurizer and steam generator safety valves along with any of the following reactor trip functions:

1. Reactor trip on turbine trip (if the turbine is tripped at above 50% power); | 99-01

2. High pressurizer pressure reactor trip;

3. Overtemperature ΔT reactor trip;

4. Low feedwater flow reactor trip or;

5. Low-low steam generator water level reactor trip.

The ASME Code pressure limit is 110% of the 2485 psig design pressure. This limit is not exceeded as discussed in Reference [5]. The report describes in detail the pressure relief devices, location, reliability, and sizing. Transient analysis data is provided for the worst cases that require safety valve actuation as well as those cases which do not.

A detailed functional description of the process equipment associated with the high pressure trip is provided in Reference [6].

The upper limit of overpressure protection is based upon the positive surge of the reactor coolant produced as a result of turbine trip under full load, i.e., a 100% load mismatch assuming that the core continues to produce full power with no steam dump. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2500 psia and a total accumulation of 3%.

The actual installed capacity of the safety valves is always greater than the capacity calculated from the sizing analysis and is indicated so by the ratio of safety valve flow to peak surge rate being greater than 1.0. Note that no credit is taken for the relief capability provided by the power operated relief valves during this surge.

The RCS design and operating pressure together with the safety, power relief and pressurizer spray valve setpoints and the protection system setpoint pressures are listed in Table 5.2-7.

System component whose design pressure and temperature are less than the RCS design limits are provided with overpressure protection devices and redundant isolation means. System discharge from overpressure protection devices is collected in the pressurizer relief tank.

As specified in NUREG 0660, item II.K.3 Table C.3 item 3, SCE&G will report safety/relief valve failures and challenges during startup testing and plant operation.

5.2.2.4 ECCS Check Valve Testing

Check valves on the discharge side of low and high head safety injection systems, and the safety injection accumulator subsystems that are classified as ASME Code, Category AC are listed in Table 5.2-7a. 00-01

The ECCS check valve system, shown by Figure 5.2-8, provides the means for leak testing individual check valves, classified Category AC, on the discharge side of the low head safety injection systems and safety injection accumulator subsystems independent 00-01 of other valves. Testing of these valves will be performed during plant startup following each refueling shutdown. Reactor coolant pressure will be at least 1000 psig. 00-01

NUCLEAR OPERATIONS TRAINING

INSTRUMENTATION AND CONTROL SYSTEM

IC-3

PRESSURIZER PRESSURE AND LEVEL CONTROL

REVISION 7

Recommended Original Signed by Hitt A. Crider Date 11/30/04

Approved Original Signed by Douglas O. Watson Date 12/01/04
Senior Instructor, Development

The Pressurizer Pressure and Level Control System is completely automatic under normal operating conditions. The system can also be manually controlled by the operator in the control room.

GENERAL DESCRIPTION

The Pressurizer Pressure and Level Control System (Figure IC3.1) is used to control the pressure in the pressurizer (and RCS) and the level in the pressurizer. The system is made up of two separate subsystems, the pressure control system and the level control system.

Pressure Control Subsystem (Figure IC3.2 and IC3.9)

The pressurizer pressure control components consist of the pressurizer heaters, spray valves, and power-operated relief valves. The pressurizer control system operates these components to maintain the desired system pressure, which is normally 2235 psig.

The heart of the pressure control system is the master pressure controller. The master pressure controller receives a signal corresponding to actual pressurizer pressure from one of the pressure measurement channels.

A signal corresponding to the constant desired (reference) pressure is internally generated within the master pressure controller. Reference pressure adjustment is made at the auto/manual station for the master pressure controller on the main control board. Reference pressure is normally set at 2235 psig. The reference signal is subtracted from the actual signal (P_{act}) to obtain a pressure error signal.

Question 30

30. 034A1.02 002

Operators are filling the refueling cavity using Spent Fuel Cooling Pump B per SOP-123, Spent Fuel Cooling System.

In order to move irradiated fuel per Technical Specification 3.9.9, Refueling Cavity level must be maintained GREATER than or equal to _________, however, Refueling Cavity level must not EXCEED _________, otherwise water may enter the Refueling Cavity ventilation intakes.

A.	461'	461' 3.5"
B.	461'	461' 8"
C.	461' 6"	461' 8"
D.	461 8"	462'

Feedback

Notes 436' 7.43": is RV level for shielding considerations during head lift. 457' 6": is Fuel Transfer Canal miminum level to minimize airborne activity. 461': equals 23' above RV flange. Is Fuel Xfer Canal Lvl Hi/Lo low alarm setpoint 461' 3.5": Is Refuel Cav Lvl Hi/Lo low alarm setpoint 461' 6": Is normal refueling cavity skimmer trough level 461' 7": Is Refuel Cav Lvl Hi/Lo high alarm setpoint 461' 8": Is location of Refueling Cavity ventilation intakes 462' 0": Is Fuel Xfer Canal Lvl high alarm setpoint.

DISTRACTORS:

A. INCORRECT 1. RC level is at the minimum required to move irradiated fuel (must be 23' above the top of the reactor pressure vessel flange = Reactor Cavity level of 461'). 2. RC level is **less** than the level of the RC ventilation intakes (461' 8").

B. CORRECT RC level must be a minimum of 461' but should not exceed 461'8".

C. INCORRECT 1. RC level is **above** the minimum needed to move fuel. 2. RC level is at the level of the RC ventilation intakes..

D. INCORRECT 1. RC level is **above** the minimum needed to move fuel. 2. RC level is **above** the level of the RC ventilation intakes..

REFERENCES:

1 SOP-123, Spent Fuel Cooling System
2 GOP-7, Core Refueling
3 XCP-609, 2-6, Refuel Cav Lvl Hi/Lo
4 XCP-612, 1-6, Fuel Xfer Canal Lvl Hi/Lo
5 Summer Tech Spec 3.10

K/A CATALOGUE QUESTION DESCRIPTION:
- Fuel Handling Equipment System (FHES); Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal.

Question #30 034A1.02 002

Recommendation:

Delete the question.

Justification:

This question relates to values in the plant Technical Specifications. Knowledge of the precise Refueling Cavity level which corresponds to the ventilation louver level is not a parameter operators are expected to memorize. The precise level (461' 6") is addressed in the Infrequent Operations section of SOP-123, Spent Fuel Cooling System; Section IV.AA (Page 76). This information would be covered during the pre-job brief prior to raising level in the Refueling Cavity. During this evolution, an operator is stationed in the Reactor Building for monitoring level while on the headset with the Control Room. There is also a CRT mounted in the Control Room for remote indication. In summary, operational relevance is limited to knowing that the pre-job brief would address the importance of the maximum Refueling Cavity level and what to monitor as level is raised.

In addition, this question is inappropriate for an RO question, because it relates to Technical Specification values. Testing on Technical Specifications is not in accordance with 10 CFR 55.41. Rather, in accordance with regulations, testing on Technical Specifications is limited to SRO exams, as reflected in 10 CFR 55.43(b)(2) - Facility operating limitations in the Technical Specifications and their bases. This position is consistent with the absence of cross-references to 10 CFR 55.41 in the generic K/A items that are associated with Technical Specifications (refer to generic K/As 2.1.33, 2.2.22, 2.2.33, 2.2.24, 2.2.25)

Also, this question does not match the K/A in that it does not ask about Fuel Handling Systems controls. Rather, it asks about level in the Refueling Cavity as related to filling the cavity with the Spent Fuel Cooling System.

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted. In addition, this question does not relate to a 10 CFR 55.41 item for an RO examination.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO._________

GENERAL OPERATING PROCEDURE

GOP-7

CORE REFUELING (MODE 5 TO MODE 6, DEFUEL, AND REFUEL TO MODE 6)

REVISION 9

SAFETY RELATED

Original Signed By: D. A. Baker
DISCIPLINE SUPERVISOR

09/09/03
DATE

Original Signed By: G. A. Lippard
APPROVAL AUTHORITY

09/09/03
DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	03/25/04					
B	P	07/23/04					
C	P	11/24/04					
D	P	04/19/05					

CONTINUOUS USE

Continuous Use Of Procedure Required.
Read Each Step Prior To Performing.

2.3 Perform the following:

☐ a. Open XVG06662-SF, REFUEL WTR STG TK SPENT FUEL ISOL VALVE (YD-170'W).

☐ b. Open XVG06664-SF, REFUEL WTR STG TK SF HDR B SUCT ISOL (AB-412).

☐ c. Open XVG06667-SF, SF HDR B CASK LOADING AREA ISOL VALVE (AB-388).

N02→ ☐ d. Unlock and open XVG06668-SF, FUEL TRANSFER CANAL SF HDR ISOL VALVE (FB-436).

2.4 When ready to commence the transfer, perform the following:

☐ a. Open XVG06651-SF, SPENT FUEL COOLING PUMP B SUCTION VALVE (AB-412).

☐ b. Start XPP0032B-SF, SPENT FUEL PIT COOLING PUMP B (AB-412). **(PEER ✓)**

☐ c. Throttle XVT06659-SF, SPENT FUEL HEAT EXCHANGER B OUTLET VLV (AB-388), to establish flow between 600 gpm and 1800 gpm as indicated by FI-7459, HX B FLOW GPM.

NOTE 2.5

The Refueling Cavity ventilation intakes are two inches above the water (461' 8") when the Refueling Cavity is filled to the normal skimmer trough level at 461'6".

2.5 Continue filling until the levels in the Fuel Transfer Canal and the Spent Fuel Pool have equalized as indicated by the following:

☐ a. LI-7405, FUEL XFER CANL LEVEL FEET.

☐ b. LI-7431, POOL LEVEL FEET.

☐ c. LI-7433, POOL LEVEL FEET.

2.6 As the desired level is approached, not to exceed 461' 6", secure the transfer as follows (AB-412):

☐ a. Stop XPP0032B-SF, SPENT FUEL PIT COOLING PUMP B.

☐ b. Close XVG06651-SF, SPENT FUEL COOLING PUMP B SUCTION VALVE.

INITIALS/DATE

NOTE 3.15

a. The preferred method for filling the Refueling Cavity is using the Spent Fuel Cooling System.

b. Steps 3.15.a and b may be omitted when filling the Refueling Cavity using RHR.

3.15 Raise Refueling Cavity water level as follows: ______/______

a. Remove the Fuel Transfer Tube blind flange. ☐

b. Open XVM06737-SF, FUEL TRANSFER TUBE VALVE (FB-436). ☐

c. Raise Refueling Cavity water level by one of the following methods:

1) Using Spent Fuel Cooling Pump B, fill the Fuel Transfer Canal and/or Refueling Cavity per SOP-123, Spent Fuel Cooling System. ☐

2) Transfer borated water from the RWST to the Refueling Cavity using RHR Pump A(B) and overflowing the Reactor Vessel per SOP-115, Residual Heat Removal. ☐

d. Perform the following tests during the Cavity fill:

CAUTION 3.15.d.1)

Performing STP-230.006A when the Refueling Cavity water level is less than 437.2 ft will increase Reactor Building airborne activity.

1) When Refueling Cavity water level has been raised to at least six feet above the Vessel Flange (443′ – 7.43″), perform STP-230.006A, ECCS/Charging Pump Operability Testing (Refueling). ☐

INITIALS/DATE

> **NOTE 3.15.d.2)**
>
> The performance of STP-205.017 will raise Refueling Cavity level 15 inches when the Fuel Transfer Tube blind flange is removed and XVM06737-SF, FUEL TRANSFER TUBE VALVE, is open.

2) When Refueling Cavity water level is established between 452.2 ft and 457.8 ft, perform STP-205.017, Accumulator Check Valve Flow Test. ☐

3.16 If required, align the Incore Sump Pumps to the Refueling Cavity via the RCDT and Spent Fuel Purification System per SOP-108, Liquid Waste Processing System. ______/______

3.17 Complete STP-110.001, Pre-Core Alterations Verifications. ______/______

3.18 Within 2 hours prior to the start of the movement of control rods, verify at least 23 feet of water exists over the top of the Reactor Vessel Flange (greater than 461 feet). ______/______

3.19 Unlatch all Control Rod Drive Shafts per FHP-611.2, Control Rod Drive Shaft Unlatching Tool. ______/______

3.20 Remove the Reactor Vessel Upper Internals per GMP-100.007, Maintenance Support For Refueling. ______/______

3.21 Verify the Fuel Transfer System is ready for operation per GMP-100.007, Maintenance Support For Refueling, as follows: ______/______

a. Ensure the water level in the Spent Fuel Pool, Transfer Canal, and Refueling Cavity are as close to equal as practical. ☐

b. Complete Attachment III.B of GTP-702, Surveillance Activity Tracking And Triggering. ☐

c. Remove the Weir Gate between the Transfer Canal and the Spent Fuel Pool per GMP-100.012, Crane Operations - Fuel Handling Building. ☐

Task

Task - Task Name and Brief Description

FILLING THE FUEL TRANSFER CANAL AND REFUELING CAVITY per SOP 123 section IV.D

FILL THE FUEL TRANSFER CANAL AND REFUELING CAVITY VIA SPENT FUEL COOLING PUMP B

Equipment XPP0032B **System** SF

Completion Criteria

Transfer is complete, with desired levels in both the Canal and Cavity

Identify each Participant, Who is in charge, and each persons Tasks / Responsibilities

Paritcipants	*Location*	*Tasks / Responsibilities*
Cavity watcher	RB 463	Monitor Cavity level
ABLL	XVT06659-SF	Throttle valve on AB 388
ABUL	SF Pump B, AB 412	Start B SF Pump, assist with XVT06659-SF
FTC watcher	FB 463	Monitor Fuel Transfer Canal level

Seq. of Events - Brief description of normal sequence

1) Remove transfer tube flange and open gate valve (XVM06737-SF).
2) Fill FTC and cavity to less than cavity upper deck (<437.2') per SOP.
   Don't start pump at first to avoid excessive splashing from Transfer Tube into Cavity.
   Get HP's opinion on running cavity fans during evolution.
3) After Head Lift, fill cavity/FTC to at least 6' above Flange (443.62).
3) (If desired) Perform STP230.006A CHG/SI flow tests. (approx. 4 hours)
4) (if desired) Perform STP205.017 Accumulator flow tests when level is 452.2' to 452.8'.
   This raises level 15 inches. (approx. 2 hours)
5) If desired to continue with the SF system, continue cavity/FTC fill to >23' using B SF Cooling pump.
   Reduce fill rate as level approaches cavity ventilation intake.
6) Align SF Purification as directed.

Discuss any expected Deviations from the normal sequence

Tech Specs (ODCM, FPER...) - (for info only, not necessarily all inclusive)

-- 3.9.7.2	Maintain >2800 gpm RHR flow in Mode 6 with both loops available.
-- 3.1.2.5	Minimum RWST level after completion ~12.2% (9690 gallons instrument inaccuracies + 51500 T.S. limit) or operable BAT.

EOOS/Functionality Impact

Comp. Req's - Compensatory Requirements resulting from task

Equipment - Special equipment needed and purpose

-- Binoculars for cavity watcher to see rising level if needed.

Communication - Communication Methods

-- Radio and plant page system

Trends - Expected Trends or Results

-- Flow rates for XVT06659-SF:
5 turns open ~950 GPM
8 turns open ~1775 GPM

-- Volumes, times & flow rates:
Empty to ~1' below cavity upper deck (436.2'): RWST volume used ~12.5%, 61820 gal.
Time to fill @ 1775GPM ~35 min.

-- Volumes, times & flow rates:
~1' below cavity upper deck to ~6' above flange (443.62'): RWST volume used ~15.2%, 76362 gal.
Time to fill @ 1775GPM ~43 min.

-- Volumes, times & flow rates:
6' above flange to 461.5': RWST volume used ~39%, 196378 gal (10983.1gal/foot, FTC/cavity combined).
Time to fill @ 1775GPM ~ 1 hour 51 min.

Stop criteria - Criteria and Method to stop the evolution and place plant in a safe condition

-- Rad levels or leakage rising unexpectedly.
Check RB sump inleakage on GRPDIS SUMSUM.
Check for increased leakage from the bowl of any RCP undergoing seal work.

Concerns - Anticipated Problems or Hazards

-- Performing STP230.006A from < 6' above flange (<443.62') will result in airborne activity in the RB.

--

--

Contingencies - Contingencies for Anticipated Concerns

-- Multiple means of cavity level verification.

Safety

Industrial Safety (PPE, specific hazards related to the task)

-- RWST pit & Refueling cavity are Confined Spaces. Ensure appropriate samples/permits obtained if flange requires removal or pit needs to be entered.

-- Refueling cavity is a fall hazard prior to fill. Ensure appropriate fall protection used if moving within 6' of cavity wall.

Radiological - ALARA

-- Airborne activity, increased dose rates near surface of cavity water.

-- Notify HP prior to starting so that appropriate coverage can be arranged.

Reactivity Management Concerns
- Inadvertent dilution is major concern during Mode 6 condition.

Environmental or Chemical Concerns

Human Performance

Errors - Identify Error Likely Situations
- Numerous tasks require close coordination and good communications.
- Cavity overflow into ventilation duct.

Irreversible - Identify Irreversible Actions

STAR (Self checks, Peer checks)
- Placekeeping.
- Proper communications.

Procedures - Procedure Used, Latest Rev and Change, Place Keeping methods

-- SOP-123	Cavity fill and purification per section IV
-- GOP-7	Overall cavity fill guidance
-- STP205.017	Accumulator flow testing.
-- STP230.006A	CHG PP flow testing
--	

Other Resources

Lessons - Lessons Learned / Operating Experience
- Numerous occurences of cavity water flowing into vent duct. Multiple means of level verification include: MCB indicators, local operator with binoculars, camera on 'bobber' and/or local level indicator. (Note high and low level alarm setpoints are close together and may not provide reliable indication.) CER 01-0288
- Numerous CERs on FME into canal.
- Numerous CERS on improper fall protection in vicinity of cavity.

Training - Is "Just in Time" training needed prior to performing this evolution?
- Are there opportunities for OJT during the evolution?

Industry Events - Related Industry events
- OE-16236 High airborne activity due to high flowrates out of vessel prior to having established a level above the

vessel flange.

- OE 498-030531-1 RWST level low-low due to cavity fill coincident with latched SI generated due to testing resulted in sump valves opening and RWST water being flushed into containment sump.
- OE 272-990923-1 High dose rates at surface of refueling cavity water (1 Rem/Hr) due to poor RCS cleanup.
- OE EAR-PAR 00-010 Cavity overfill due to failed instrumentation and operators not checking redundant indication.

Information Links

Solicit input from each participant Pre and Post evolution

Participant and Location

Cavity watcher

ABLL

ABUL

FTC watcher

Have you answered the four Key Questions?

1. *What's the **Worst** that could happen?* ☐
2. *What are the **Critical Steps** ?* ☐
3. *What are the **Error Likely** situations ?* ☐
4. *What are our **Defenses** ?* ☐

Last update by

Rob Ray 10/18/2003 2:01:05 PM

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

SYSTEM OPERATING PROCEDURE

SOP-123

SPENT FUEL COOLING SYSTEM

REVISION 14

SAFETY RELATED

Original Signed By: D. A. Baker — 06/22/05
DISCIPLINE SUPERVISOR — DATE

Original Signed By: G. A. Lippard — 06/27/05
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	09/08/05					
B	P	11/02/05					

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing

C. SPENT FUEL POOL PURIFICATION

1.0 INITIAL CONDITIONS

☐ 1.1 The Spent Fuel Pool level is in the skimmer (461'6").

☐ 1.2 The Spent Fuel Pool temperature is less than 140°F.

2.0 INSTRUCTIONS

☐ 2.1 Ensure Spent Fuel Purification is secured per Section IV.

2.2 Open the following valves:

☐ a. XVD06669-SF, SPENT FUEL POOL PUR HDR ISOL VALVE (FB-436).

☐ b. XVD06674-SF, SPENT FUEL POOL PUR HDR SUPPLY VALVE (FB-436).

☐ c. XVD06692-SF, SF PUR HDR SF HEADER B SUP ISOL VALVE (AB-388).

☐ 2.3 Start XPP0014, SPENT FUEL PURIFICATION PUMP (AB-412).

☐ 2.4 To prevent channeling of the Spent Fuel Cooling Demineralizer, throttle XVD06690-SF, SPENT FUEL PURIFICATION HDR ISOL VALVE (AB-436), to establish a flow of less than 180 gpm as indicated by IFI07425, SPENT FUEL PUR FILT OUTLET FLOW IND (AB-436).

☐ 2.5 When skimming operations are complete, stop XPP0014, SPENT FUEL PURIFICATION PUMP (AB-412).

2.6 Close the following valves:

☐ a. XVD06669-SF, SPENT FUEL POOL PUR HDR ISOL VALVE (FB-436).

☐ b. XVD06674-SF, SPENT FUEL POOL PUR HDR SUPPLY VALVE (FB-436).

☐ c. XVD06692-SF, SF PUR HDR SF HEADER B SUP ISOL VALVE (AB-388).

END OF SECTION

D. FILLING THE FUEL TRANSFER CANAL AND/OR THE REFUELING CAVITY VIA SPENT FUEL COOLING PUMP B

1.0 INITIAL CONDITIONS

☐ 1.1 A **Pre-Job Brief** has been conducted per OAP-100.3.

☐ 1.2 The RWST level is sufficient to ensure Technical Specification 3.1.2.6 (Modes 1, 2, 3, and 4) or 3.1.2.5 (Modes 5 and 6) is not violated during water transfer.

☐ 1.3 One of the following conditions exists (FB-436):

a. If only the Fuel Transfer Canal is to be filled, XVM06737-SF, FUEL TRANSFER TUBE VALVE, is closed or the blind flange is installed.

b. If the Refueling Cavity is to be filled simultaneously with the Fuel Transfer Canal, XVM06737-SF, FUEL TRANSFER TUBE VALVE, is open and the blind flange is removed.

N01↓ ☐ 1.4 The Spent Fuel Pool slide gate is installed and the boot is inflated to 30 psig.

☐ 1.5 The Cask Loading Area slide gate is installed and the boot is inflated to 30 psig.
N01↑

☐ 1.6 If the Refueling Cavity is to be filled, the temporary cavity drain has been installed and tagged closed.

☐ 1.7 Spent Fuel Cooling Loop B is shutdown per Section III.

2.0 INSTRUCTIONS

☐ 2.1 Ensure Spent Fuel Purification is secured per Section IV.

☐ 2.2 Close XVG06661-SF, SF COOLING PUMP B SF POOL HDR ISOL VLV (AB-388).

AA. RAISING THE REFUELING CAVITY LEVEL VIA THE SPENT FUEL PURIFICATION PUMP

1.0 INITIAL CONDITIONS

☐ 1.1 In order to minimize the release of airborne activity, one of the following conditions has been met:

a. The Refueling Cavity level is 457′ 6″ or greater.

b. If Refueling Cavity level is below 457′ 6″, the use of this procedure has been evaluated by Operations and Health Physics Management.

2.0 INSTRUCTIONS

☐ 2.1 Ensure Spent Fuel Purification is secured per Section IV.

2.2 Unlock and open the following valves:

☐ a. XVD06698-SF, REFUEL CAVITY SF PUR HDR ISOL VLV (IRC) (RB-412).

☐ b. XVD06697-SF, SF PUR HEADER REFUEL CAVITY ISOL VALVE (FB-412).

2.3 Open the following valves:

☐ a. XVD06675-SF, REFUEL WTR STG TK SF PUR HDR SUP VALVE (AB-412).

☐ b. XVT06701-SF, REFUEL WTR STG TK SF PUR HDR SUP ISOL (YD-170'W).

☐ 2.4 Start XPP0014, SPENT FUEL PURIFICATION PUMP (AB-412).

☐ 2.5 To prevent channeling of the Spent Fuel Cooling Demineralizer, throttle XVD06690-SF, SPENT FUEL PURIFICATION HDR ISOL VALVE (AB-436), to establish flow less than 180 gpm as indicated by IFI07425, SPENT FUEL PUR FILTER OUTLET FLOW IND (AB-436).

☐ 2.6 If required, due to low RWST level, throttle XVD06745-SF, SF PURIFICATION PUMP DISCHARGE VALVE (AB-412), to maintain suction pressure greater than or equal to 6 psig, as indicated on PI-7411, PURIFICATION PUMP SUCTION PRESS (AB-412).

NOTE 2.7

The Refueling Cavity ventilation intakes are 2 inches above the water (461' 8") when the Refueling Cavity is filled to the normal skimmer trough level at 461'6".

☐ 2.7 When desired level is reached, stop XPP0014, SPENT FUEL PURIFICATION PUMP (AB-412).

☐ 2.8 Ensure XVD06745-SF, SF PURIFICATION PUMP DISCHARGE VALVE, is fully open (AB-412).

2.9 Close the following valves:

☐ a. XVD06698-SF, REFUEL CAVITY SF PUR HDR ISOL VLV (IRC) (RB-412).

☐ b. XVD06697-SF, SF PUR HEADER REFUEL CAVITY ISOL VALVE (FB-412).

2.10 Lock the following valves:

☐ a. XVD06698-SF, REFUEL CAVITY SF PUR HDR ISOL VLV (IRC) (RB-412).

☐ b. XVD06697-SF, SF PUR HEADER REFUEL CAVITY ISOL VALVE (FB-412).

2.11 Close the following valves:

☐ a. XVD06675-SF, REFUEL WTR STG TK SF PUR HDR SUP VALVE (AB-412).

☐ b. XVD06701-SF, REFUEL WTR STG TK SF PUR HDR SUP ISOL (YD-170'W).

☐ 2.12 Place Spent Fuel Purification in service as directed by the Shift Supervisor.

END OF SECTION

Question 41

41. 058AA2.01 002

The Unit is operating in Mode 1 when the following annunciators actuate:

-INV 1/2 TROUBLE
-INV 1/2 AC INPUT LOSS

Which ONE of the following describes what could have caused BOTH the annunciators and the power source to the inverter based on that condition?

A. Bus volts on XIT-5901 dropped to 120 VDC; Power is now being supplied from alternate source 1FA
B. Bus volts on XIT-5901 dropped to 110 VDC; Power is now being supplied from alternate source 1FA
C. Bus volts on XIT-5903 dropped to 110 VDC; Power is now being supplied from alternate source 1FB
D. Bus volts on XIT-5903 dropped to 120 VDC; Power is now being supplied from alternate source 1FB

Feedback

DISTRACTORS:

A INCORRECT XIT-5901 could be a source of either alarm, but voltage would have had to have dropped to = 120VDC.
B CORRECT Per ARP-001, XCP-636, Annunciator Point 1-5, page 7. If the cause of the alarms are low DC voltage, the INV 1/2 TROUBLE alarm cannot actuate at 110 VDC without the INV 1/2 AC INPUT LOSS alarm at 120VDC since nominal DC bus voltage is about 135 VDC. Also, you cannot get a DC bus voltage this low without first having a low AC source voltage, which will cause the swap to the alternate AC source
C INCORRECT This would be correct if annunciator INV 3/4 had actuated.
D INCORRECT This would be correct if annunciator INV 3/4 had actuated

REFERENCES:

1 ARP-001, Panel XCP-636, Annunciator Point 1-5, page 7.
2 ARP-001, Panel XCP-637, Annunciator Point 1-5, page 7.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of DC Power; Ability to determine and interpret the following as they apply to the Loss of DC Power: That a loss of dc power has occurred; verification that substitute power sources have come on line.

Question #41 (058AA2.01 002)

Recommendation:

Delete the question.

Justification:

Knowledge of the specific DC voltage setpoint which would generate BOTH annunciators is not operationally significant and need not be committed to memory.* The *types* of parameters (AC, DC), which could actuate the annunciators *are* operationally significant.

The question does not test the stated K/A in that it does not test that a "loss of DC power has occurred" because that conclusion is given in all four choices. It does not test the operator's ability to determine "that substitute power sources have come on line" because the given information of both annunciators in alarm *should* (if the loss of DC had not been provided in all four choices) allow the examinee to determine only that a loss of DC may be the initiating event. The two annunciators are not, by themselves, evidence that substitute power is supplying the inverter. It is possible that the two annunciators actuate, the Main Control Board (MCB) operator dispatches an Auxiliary Operator (AO), and the AO finds that the inverter has failed and is supplying no output AC power. In order to reach a conclusion that alternate power had come on line, information such as the following should have been provided:

- "all automatic actions occurred as designed;"
- "the "AO reports that local indications at the inverter which would provide the *indications* that alternate power was on-line and providing input to the inverter;" or
- "the MCB operator notes indications on the MCB which would allow the examinee to determine alternate power was restored."

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.

* See discussion in Section II.A.

PANEL XCP-636
ANNUNCIATOR POINT 1-6

INV 1/2 AC INPUT LOSS	SETPOINT:	ORIGIN:
	AC OUTPUT volts ≤114VAC	AS-4-XIT-5901 or 5902
	DC BUS volts ≤120VDC	OB-2-XIT-5901 or 5902

CHG C

PROBABLE CAUSE:

1. Loss of Alternate source (1FA) or one of the following breakers open, as indicated by the ALTERNATE AC FAIL light being lit:
   a. APN1FA 19, CKTBRK-FEEDER FOR XIT5901.
   b. ALT. AC SOURCE Breaker on XIT-5901.
   c. APN1FA 23, CKTBRK-FEEDER FOR XIT5902.
   d. ALT. AC SOURCE Breaker on XIT-5902.
2. Loss of Normal source (1DA2Y) or one of the following breakers open, as indicated by the DC BUS ON BACKUP light being lit:
   a. XMC 1DA2Y 06ABL, INVERTER 1 NSSS FEED XIT5901-EV.
   b. NORMAL AC SOURCE Breaker on XIT-5901
   c. XMC 1DA2Y 06ABR, INVERTER 2 NSSS FEED XIT5902-EV
   d. NORMAL AC SOURCE Breaker on XIT-5902

AUTOMATIC ACTIONS:

1. Inverter will switch to the backup source without loss of AC output.

CORRECTIVE ACTIONS:

1. Verify that Vital Instrument Panels APN-5901 and APN-5902 are energized by dispatching an operator to check the output ammeter on XIT-5901 and XIT-5902.

SUPPLEMENTAL ACTIONS:

1. If the inverter is not powering the associated APN or is not connected to the battery, refer to Tech Spec 3.8.3 for LCO requirements.
2. Determine the cause of the tripped breaker and correct.
3. Reset the tripped breaker when conditions permit.
4. Place the MAN BYPASS SWITCH to NORMAL when conditions permit.

PANEL XCP-636
ANNUNCIATOR POINT 1-5

INV 1/2 TROUBLE	SETPOINT:	ORIGIN:
	DC BUS volts ≤110VDC	DCB-1-XIT-5901 or 5902
	Static Switch ON ALTERNATE	TAP-3-XIT-5901 or 5902
	AC OUTPUT volts ≤114VAC	UPSO-5-XIT-5901 or 5902

CHG C

PROBABLE CAUSE:

1. Static Switch isolated as indicated by AC OUTPUT FAIL light being lit.
2. Loss of Backup Source (1HA) as indicated by DC BUS LOW VOLTAGE light being lit.
3. UPS Static Switch selected to Alternate source as indicated by ON ALTERNATE light being lit.

AUTOMATIC ACTIONS:

1. UPS transfers to Alternate Source (1FA) on a loss of Inverter (Preferred) voltage.

CORRECTIVE ACTIONS:

1. Verify that Vital Instrument Panels APN-5901 and APN-5902 are energized by dispatching an operator to check the output ammeter on XIT-5901 and XIT-5902.

SUPPLEMENTAL ACTIONS:

1. If the inverter is not powering the associated APN or is not connected to the battery refer to Tech Spec 3.8.3 for LCO requirements.
2. If the inverter has apparently failed identify and correct the cause as follows:
   a. Notify Electrical Maintenance for assistance.
   b. When the inverter is repaired, return it to service per SOP-310.
   c. When conditions permit, place the MAN BYPASS SWITCH to NORMAL.

REFERENCES:

1. 1MS-37-098-1.
2. B-804-636, Sh. 2.
3. B-208-039, EV-01.
4. E-206-062, Sh. 1 and 3.
5. SOP-310, 120 VAC Instrument and Control System.
6. V.C. Summer Tech Specs.
7. 1MS-94B-1317. CHG C

Question 45

45. 062A3.04 002

Inverter XIT5901 is being returned to service after power to the inverter had been lost. All initial conditions required to return the inverter to service have been met.

Step III.N.2.1 of SOP-310, 120 VAC INSTRUMENT AND CONTROL SYSTEM, requires the operator to "Close the NORMAL AC SOURCE Breaker on the inverter front."

Which ONE of the following describes the status of the DC BUS CHARGED light as it relates to the operation of the above breaker?

A. It should light immediately upon closing the breaker and indicates charged capacitors.
B. It should extinguish immediately upon closing the breaker and indicates that the DC bus is no longer supplying power to the inverter.
C. It should light within 10 seconds of closing the breaker and indicates charged capacitors.
D. It should extinguish within 10 seconds of closing the breaker and indicates that capacitors have completed charging.

Feedback

DISTRACTORS:

A. INCORRECT It should initially be dim. Upon closing the breaker, it should light within 10 seconds and indicates charged capacitors.
B. INCORRECT It should light within 10 seconds after closing the breaker to indicate charged capacitors.
C. CORRECT The light should initially be extinguished. After closing the breaker, it should light within 10 seconds and indicates charged capacitors.
D. INCORRECT It should light within 10 seconds.

REFERENCES:

1. SOP-310, "120 VAC Instrument and Control System," pages 22 - 24 and page 1 of Enclosure A.

K/A CATALOGUE QUESTION DESCRIPTION:
- AC Electrical Distribution System; Ability to monitor automatic operation of the ac distribution system, including: Operation of inverter (e.g. precharging synchronizing light, static transfer).

<u>Question #45 (062A3.04 002)</u>

Recommendation:

Delete the question.

Justification:

There is no correct answer to this question because Step 2.2 of Section III.N. of SOP-310, 120 VAC [volts alternating current] Instrument and Control System, requires the operator to check the status of the DC BUS CHARGED light *after* ten seconds, not *within* 10 seconds as the choices stipulate.

Additionally, this question is inappropriate, as it tests detailed knowledge of Subsection III.N., which is well into the body of SOP-310. Because SOP-310 is a Continuous Use Procedure, the operator would be expected to follow it verbatim when placing an inverter in service.

Therefore, this question should be deleted because there are no choices that reflect the expected outcome (i.e., a "lit" indicating light <u>after</u> 10 seconds) and because this detail is not one that operators are expected to know in order to safely implement the procedures.

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO._____________

SYSTEM OPERATING PROCEDURE

SOP-310

120 VAC INSTRUMENT AND CONTROL SYSTEM

REVISION 10

SAFETY RELATED

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing.

N. RETURN OF INVERTER XIT5901 TO SERVICE

1.0 INITIAL CONDITIONS

☐ 1.1 A **Pre-Job Brief** has been conducted per OAP-100.3.

☐ 1.2 APN1FA-EM, 120 VOLT AC INST MAIN DISTR PANEL 1FA is providing power to APN5901, 120V VITAL AC DISTR PNL 1 NSSS through the ALT. AC SOURCE Breaker on the inverter front.

☐ 1.3 The TEST TRANSFER Switch is in the ALT position.

☐ 1.4 The MAN. BYPASS Switch is in the BYP TO ALT position.

☐ 1.5 The RETURN MODE Switch is in the AUTO position.

1.6 The following breakers are closed:

☐ a. XMC1DA2Y 06ABL, INVERTER 1 NSSS FEED XIT5901-EV.

☐ b. DPN1HA 22, INVERTER XIT5901.

2.0 INSTRUCTIONS

☐ 2.1 Close the NORMAL AC SOURCE Breaker, on the inverter front.

2.2 After ten seconds, verify the following:

☐ a. DC BUS voltage rises to between 120 volts and 141 volts.

☐ b. The DC BUS CHARGED Light illuminates.

☐ c. The NORMAL AC ON Light illuminates.

> NOTE 2.3
>
> A five minute inverter logic warm-up period should be allowed before continuing.

☐ 2.3 After the five minute inverter logic warm-up period, place the SOURCE SELECTOR Switch, to the INV. position.

SYSTEM INFORMATION

1. 120VAC Vital Instrument supply:

   a. Four Inverters (XIT5901, 5902, 5903, 5904).

   b. Normal AC supply - 480V Vital Buses.

   c. Backup DC supply - 125V Vital DC Buses.

   d. If the inverter is out of service, the alternate AC supply is from APN1FA(1FB) through the static switch. The static switch will automatically connect the alternate supply to its associated APN if the inverter fails.

2. 120VAC Balance of Plant Instrument supply:

   a. Inverter XIT5905:

      1) Normal AC supply - 480V BOP Bus.

      2) Backup DC supply - 125V Non-Vital DC Bus.

      3) Alternate AC supply - 120V BOP Bus.

   b. Inverter XIT5906:

      1) Normal AC supply - 120V APN1FX.

      2) Backup DC supply - 125V Non Vital DC Bus.

   c. Inverters XIT5909 and XIT5910:

      1) Normal AC supply - (2)480V Vital Buses through transfer switch XET4006.

      2) Backup DC supply - 125V Non Vital DC Bus.

Question 55

55. 076A1.02 002

The crew is performing actions per EOP-17.0, *Response to High Reactor Building Pressure,* following a LOCA inside containment when the following occurs:

- Annunciator SWBP/A DISCH FLO LO actuates.
- Annunciator SWBP A TRIP actuates.
- Concurrent with the above annunciators, the 1DB1 bus lost power due to an electrical fault.

Which ONE of the following describes how RBCU cooler service water return temperatures will respond given the above information with no operator action?
Train 'A' RBCU service water return temperature will _____.Train 'B' RBCU service water return temperature will _____.

A. decrease
Decrease

B. remain relatively unchanged
Increase

C. increase
remain relatively unchanged

D. remain relatively unchanged
remain relatively unchanged

Feedback

DISTRACTORS:

A INCORRECT See distractor analysis for answer D.
B INCORRECT See distractor analysis for answer D.
C INCORRECT See distractor analysis for answer D.
D CORRECT With the loss of 1DB1, power to the train 'B' service water booster pump (SWBP) is lost. The annunciators provide clear indication that the running 'A' train SWBP is lost which would result in its associated valves shutting automatically. These conditions would result in both SWBP trains being isolated. With no flow through either train, temperatures would remain relatively unchanged.

REFERENCES:

1 AB-17, "Reactor Building Ventillation System," pages 14 & 15.
2 IB-1, Service Water System, pages 23 - 25, 34, & 40, figures IB-1.1 - IB-1.3.
3 ARP-001-XCP-606, Annunciator Point 1-4, page 6.
4 EOP-17.0, "Response to High Reactor Building Pressure," Steps 3 & 4, page 4.

K/A CATALOGUE QUESTION DESCRIPTION:
- Service Water System (SWS); Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including reactor and turbine building closed cooling water temperatures.

Question #55 (076A1.02 002)

Recommendation:

Delete the question.

Justification:

There is no correct answer available from the selection.

The examinee is expected to know that the A Service Water Booster Pump (SWBP) discharge isolation valve closes automatically on pump trip, isolating all A Train Service Water (SW) return from the Reactor Building. The loss of bus 1DB1 results in the loss of all B Train SW return from the Reactor Building but does not isolate the return header. The temperature elements for both A and B Train SW return headers are located inside the Reactor Building. Upon loss of SW cooling flow return from both headers, it is expected that both A and B Train SW return temperature elements would indicate an ***increase*** in temperature as they reach equilibrium with the Reactor Building environment. There is no answer alternative for return temperature to ***increase*** for both A and B SW.

It should also be noted that Step 4.b of EOP 17 requires verification of SW to RBCUs by ensuring that greater than 2000 gallons per minute flow from at least one train of SWBP. For this question, the lack of flow would result in the alternate action to consult the Technical Support Center for evaluation of heat removal capabilities of RBCUs. By procedure, the operator would not be required to use available indications and sequence of events to determine conditions within the SW return headers.

Also, the question does not match the K/A in that it does not provide for operation of SWS controls. Instead, it tests system parameter response to multiple malfunctions in an operationally insignificant manner.

The question should be deleted because there is no correct answer and it does not match the K/A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

EMERGENCY OPERATING PROCEDURE

EOP-17.0

RESPONSE TO HIGH REACTOR BUILDING PRESSURE

REVISION 9

SAFETY RELATED

DISCIPLINE SUPERVISOR 10/29/97
DATE

R M [illegible] 10-29-97
APPROVAL AUTHORITY DATE

RESPONSE TO HIGH REACTOR BUILDING PRESSURE

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
(Step 2 continued)	(Step 2 continued)
d. Verify RB Spray flow is GREATER THAN 2500 gpm on at least <u>one</u> operating train as indicated on: • FI-7368, SPR PP A DISCH FLOW GPM. ☐ <u>OR</u> • FI-7378, SPR PP B DISCH FLOW GPM. ☐	d. Consult TSC personnel for methods to reduce RB pressure. ☐
e. Verify Phase B Isolation by ensuring RB SPRAY/PHASE B ISOL monitor lights are bright on XCP-6105. ☐	e. Close valves as necessary for Phase B Isolation. ☐
f. Stop <u>all</u> RCPs. ☐	
3 Ensure <u>two</u> RBCU Fans are running in slow speed (<u>one</u> per train). ☐	
4 Verify Service Water flow to RBCUs:	
a. Ensure <u>both</u> Service Water Booster Pumps are running. ☐	a. Ensure XCP-6104 3-1(7-1) is dim for the non-running pump (SW TO RB OUT A(B) ISOL 3107A(B)). ☐
b. Verify GREATER THAN 2000 gpm flow for at least <u>one</u> train on: • FI-4466, SWBP A DISCH FLOW GPM. ☐ <u>OR</u> • FI-4496, SWBP B DISCH FLOW GPM. ☐	b. Consult TSC personnel for evaluation of heat removal capabilities of RBCUs. ☐

PANEL XCP-604
ANNUNCIATOR POINT 2-3

SWBP A DISCH FLOW LO	SETPOINT: 2000 gpm (Enabled five seconds after breaker closure)	ORIGIN: IFB04466 52 contact on XSW1DA1 07A

PROBABLE CAUSE:

1. Service Water Booster Pump A malfunction.
2. Motor operated valve failure.
3. Valve misalignment.
4. Reactor Building Cooling Unit flow blockage.
5. IFT04466-SW, SW BOOSTER PUMP A DISCHARGE FLOW XMTR (AB-436), failure.

AUTOMATIC ACTIONS:

1. None.

CORRECTIVE ACTIONS:

1. Verify Service Water Booster Pump flow on FI-4466, SWBP A DISCH FLOW GPM.
2. Verify Service Water Booster Pump B is operating.
3. Monitor Reactor Building pressure and, if required, initiate Reactor Building Spray.

SUPPLEMENTAL ACTIONS:

1. Using status lights verify the following motor operated valve alignment:
   Open: MVB-3106A, MVG-3108A, MVG-3108B, MVG-3103A, MVG-3107A and either MVG-3109A or MVG-3109B.
   Closed: MVB-3110A, MVG-3111A, MVG-3112A.
2. If Service Water Booster Pump A failed, determine and correct the cause.
3. Verify proper valve alignment per SOP-117.
4. Refer to Technical Specification 3.7.4.

REFERENCES:

1. B-208-101, SW-71.
2. B-804-604, Sh. 2.
3. D-302-222.
4. SOP-117.
5. V.C. Summer Technical Specifications.

PANEL XCP-604
ANNUNCIATOR POINT 2-4

	SETPOINT:	ORIGIN:
SWBP A SUCT/DISCH PRESS LO	Suction-44 psig	IPS04521
	Discharge-144 psig (Enabled five seconds after breaker closure)	IPB04523 with 52 contact on XSW1DA1 07A

PROBABLE CAUSE:

1. If Service Water Loop A is inactive, this annunciator will normally be activated on low pressure.
2. Loop A Service Water Pump malfunction or degradation.
3. Service Water Booster Pump A malfunction or degradation.
4. Valve misalignment or failure.
5. Excessive Service Water flow to other loads or pipe rupture.
6. IPS04521-SW, SW BOOSTER PUMP A SUCTION PRESS SWITCH (IB-412), failure.
7. IPT04523-SW, SW BOOSTER PUMP A DISCH PRESS XMTR (IB-412), failure.

AUTOMATIC ACTIONS:

1. None.

CORRECTIVE ACTIONS:

1. Monitor the operating Loop A Service Water Pump discharge pressure.
2. Monitor PI-4523, SWBP A DISCH PRESS PSIG.
3. Verify Service Water Booster Pump B is operating.
4. Monitor Service Water Loop A indications for excessive flow, valve failure, or misalignment.
5. Monitor Reactor Building pressure and, if required, initiate Reactor Building Spray.

SUPPLEMENTAL ACTIONS:

1. If Service Water Booster Pump A failed, determine and correct the cause.
2. Verify proper valve alignment per SOP-117.
3. If excessive flow to any component is observed, perform a visual inspection for piping integrity.
4. Refer to Technical Specification 3.7.4.

NUCLEAR OPERATIONS
COPY NO. ______

NUCLEAR OPERATIONS TRAINING

AUXILIARY BUILDING SYSTEM

AB-17

REACTOR BUILDING VENTILATION SYSTEM

REVISION 7

Recommended ______ **Date** 2/16/98

Approved J. Callicott **Date** 2/16/99

Senior Instructor, Development

energized to reach the design flow rate. The total "water-side" switchover from normal to emergency operation is completed approximately 75 seconds after the reference time.

In the event of a main steam line break outside of containment accompanied by a Safety Injection, the Service Water Booster Pumps may not be available to supply cooling flow due to exposure to the harsh environment. The cooling units could be supplied from the Industrial Cooling System if they are available by either removing power from the Reactor Building Cooling Unit supply and discharge valves and manually repositioning them or by defeating the associated interlocks.

Reactor Building Cooling Units Instrumentation And Control

The Reactor Building Cooling Unit fans are safety related devices which can be actuated by control switches on the main control board. Two control switches are provided for each fan, one controlling a slow speed motor and the other controlling a fast speed motor. During normal operation, three of the four fans run continuously at fast speed. If excessive fan vibration occurs during normal operation (fast speed), that running fan automatically trips. The RBCU fans will not trip on excessive vibration during emergency operation (slow speed) due to their vital function during accident conditions. Fan status is indicated by lights on the main control board (MCB) and at the switchgear.

If three fans are normally operating at fast speed and a Safety Injection or loss of offsite power signal is received, the fans are automatically tripped by the Engineered Safety Features Loading Sequencer (ESFLS) and then two fans (one Channel A and one Channel B) are automatically started in slow speed by the ESFLS. MCB switches select which one of the two Channel A and which one of the two Channel B fans are started in response to the ESFLS.

Booster Pumps (Figure IB1.3)

Each service water loop supplies water to the suction of a booster pump. Water flows through service water booster pump suction isolation valves (XVB-3134A/XVB-3134B) to the suction of the respective booster pump. Status lights on the MCB indicate the position of these butterfly valves.

The service water booster pumps provide the necessary driving head to supply flow to the Reactor Building cooling units. Orifices in the discharge lines from containment ensure the pressure in the Reactor Building cooling units is at a greater pressure than Reactor Building post accident design pressure (53.5 psig). Therefore, no back leakage of contaminated water can enter the Service Water System and be carried to the outside environment. The pumps (XPP-45A and B) are single stage, horizontal, centrifugal types. Each pump has a design flow capacity of 4000 gpm and a shutoff head of 121 psid. Both service water booster pumps are located in the northwest corner of the Intermediate Building on the 412'-0" level. The pumps are near the component cooling booster pumps.

The short recirc. pipe for the booster pumps will heat up quickly, causing pump damage as NPSH is reduced by high suction temperature. Operation on recirc. should be minimized (a few minutes) and pump operation closely monitored at the pump.

Dashpots on the pump discharge check valves help minimize water hammer during pump auto re-start on restoration of electric power.

Booster Pump Function During Safety Injection

The service water booster pumps receive their power from 480 V buses 1DA1 and 1DB1. The pump control switches on the MCB have a STOP, START, and an unlabeled midposition. The booster pump loads (i.e., the Reactor Building cooling units) are normally supplied by industrial cooling water. Thus, the booster pumps are

normally idle. During a loss of coolant accident (LOCA), these pumps are started automatically by the ESFLS (Step 7) to supply the Reactor Building cooling units. Auto start occurs upon receipt of a safety injection signal, or under voltage condition on the respective diesel generator bus.

Pump discharge pressure indication (PI-4523 and PI-4543) is available on MCB. Annunciator "SWBP A(B) SUCT DISCH PRESS LO" actuates on the MCB if the suction pressure of the pump is less than 44 psig or the discharge pressure of the pump is less than 144 psig. Pump discharge flowrate indication (FI-4466 and FI-4496) is also on the MCB . If discharge flowrate drops below 2000 gpm, annunciator "SWBP/A(B) DISCH FLO LO" actuates on the MCB. Red pens on two flow recorders (FR-4466 and FR-4496) on the MCB record the flowrate from each pump to the four Reactor Building cooling units.

A possible cause for a low discharge pressure or flowrate from a booster pump is a pump trip, for example on motor electrical fault. Pump trips actuate the "SWBP A(B) TRIP" annunciator on the MCB. An ammeter on the MCB indicates the pump motor status. The discharge of the service water booster pumps flows through discharge isolation valves (MVB-3106A and B) in the respective loop. These motor operated isolation valves are controlled by CLOSE, AUTO, OPEN switches on the MCB. These valves open automatically on booster pump start and close on pump stop. ESF monitor lights "SW TO RB IN A(B) ISOL 3106A(B) CLSD" indicate the status of these valves.

Reactor Building Cooling Units

The normal water supply to the Reactor Building coolers is industrial cooling water. Industrial cooling water taps into the Reactor Building cooler supply just downstream of the booster pump discharge isolation valves. The isolation valves for the industrial cooling supply lines (MVB-3110A and B) are operated from the MCB. The control switches for these valves have CLOSE, AUTO, OPEN positions. In AUTO, the valves close upon receipt of an ESFLS signal. When these valves close, the normal supply of

NUCLEAR OPERATIONS TRAINING

INTERMEDIATE BUILDING SYSTEM

IB-1

SERVICE WATER SYSTEM

REVISION 15

Recommended *Charles J. Dickey Jr* Date *10/25/04*

Approved *Douglas O. Watson* Date *10/25/04*
Senior Instructor, Development

SERVICE WATER SYSTEM

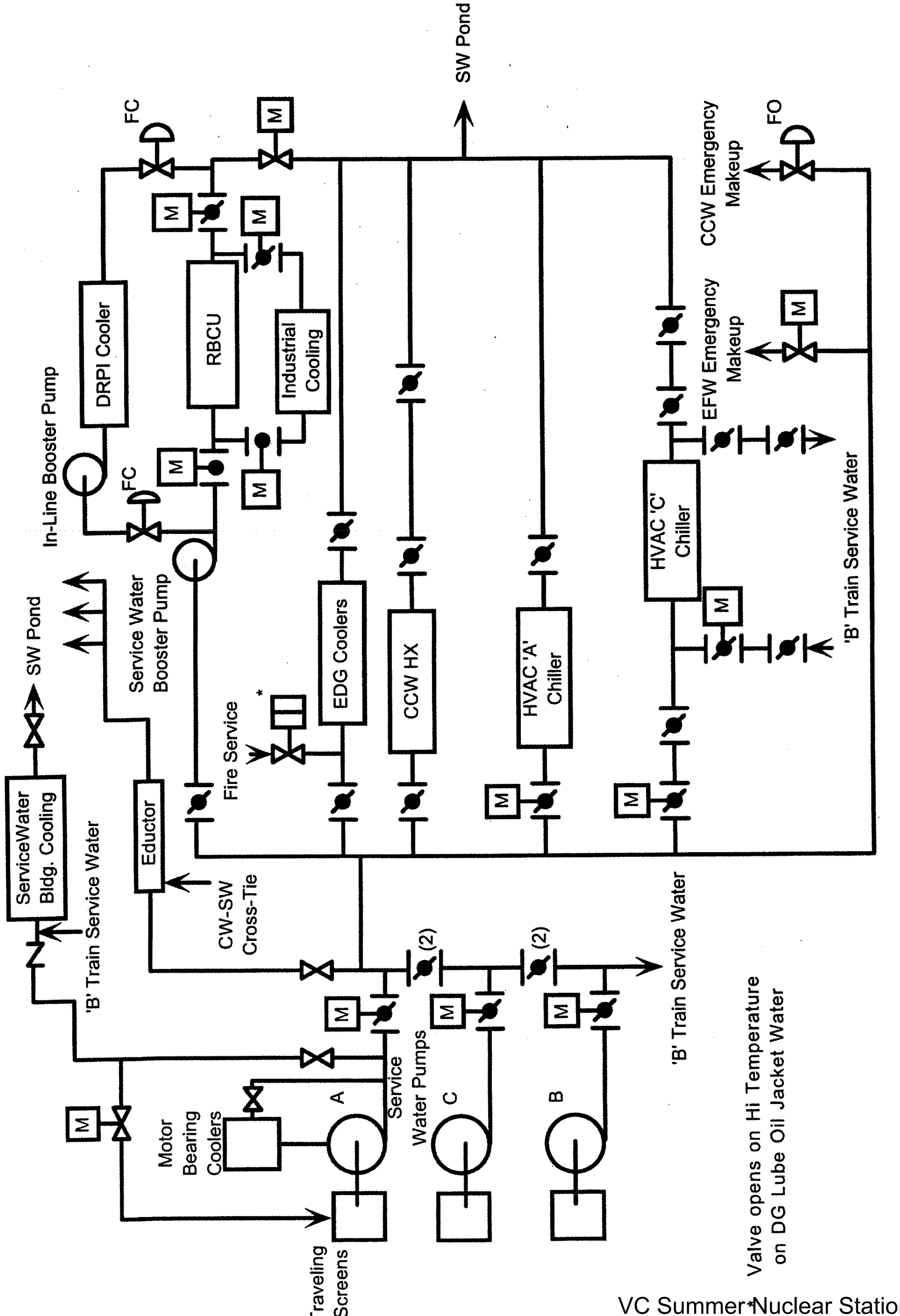


Figure IB1.1

IB1.1*1

SERVICE WATER SYSTEM

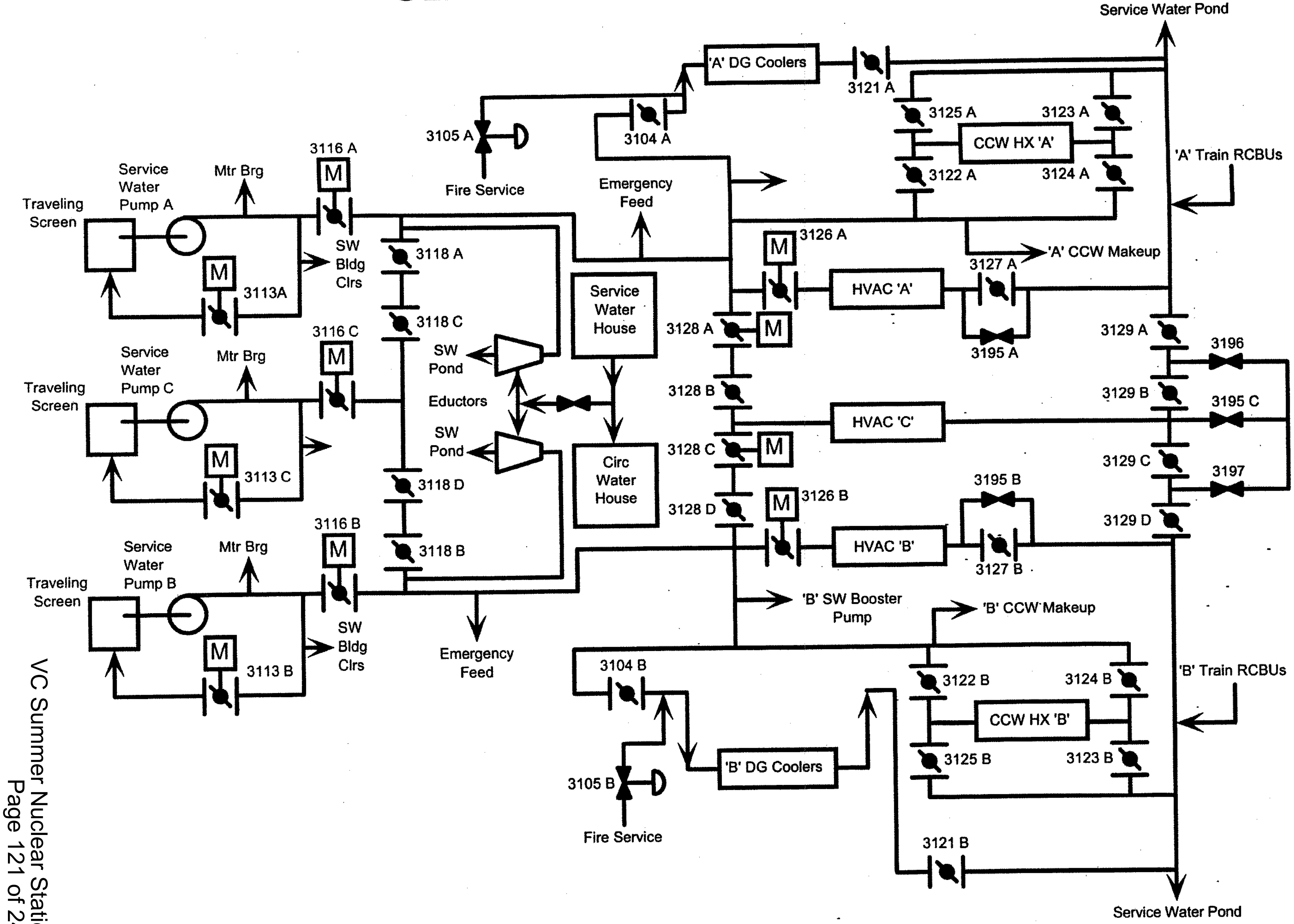


Figure IB1.2

Question 69

69. W/E04EA2.1 004

The following conditions exist:

- A Reactor Trip and Safety Injection has occurred.
- EOP-1.0 "Reactor Trip/Safety Injection Actuation has been entered.
- Pressurizer Pressure is 1850 psig and decreasing.
- Steam Generator pressures are all 850 psig and stable.
- Steam Generator Levels are all approximately 20% and rising.
- Containment Pressure indicates 0.25 psig.
- RM-A3, MAIN PLANT VENT EXH ATMOS, is in alarm.
- XCP-631-6-1, AB SUMP LVL HI, is lit.

Which ONE of the following describes the correct procedure that should be entered next?

A. EOP- 2.0 "Loss of Reactor or Secondary Coolant."
B. EOP-3.0 "Faulted Steam Generator Isolation."
C. EOP-2.5 "LOCA Outside Containment."
D. EOP-2.1 "Post-LOCA Cooldown and Depressurization."

Feedback

DISTRACTORS:

A. INCORRECT A loss of RCS inventory is occurring, however conditions in containment do not indicate that the leak is in Containment, and S/G conditions are normal for this condition.
B. INCORRECT None of the indications provided in the stem indicate a Faulted S/G.
C. CORRECT In accordance with Step 23 of EOP-1.0, the two alarms, RM-A3 and XCP-631-6-1, provide indication that an RCS leak exists outside containment and support this transition. None of the other indications trigger a transition out of EOP-1.0 either before or after step 23.
D. INCORRECT If a small break LOCA was believed to be in progress this would be the procedure to enter after EOP2.0.

REFERENCES:
1 Lesson Plan EOP-2.5 "LOCA Outside Containment". Objective 3
2 EOP-2.5 "LOCA Outside Containment," Step 23, page 14.

K/A CATALOGUE QUESTION DESCRIPTION:
- LOCA Outside Containment; Ability to determine and interpret the following as they apply to the (LOCA Outside Containment): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Question #69 W/E04EA2.1 004

Recommendation:

Delete the question.

Justification:

This question – which involves transitions to EOPs – is inappropriate for an RO examination because it samples SRO knowledge requirements as reflected in 10 CFR 55.43(b)(5), assessment of facility conditions and **selection of appropriate procedures** during normal, abnormal, and emergency situations.

Selecting transitions to EOPs is not addressed in the RO knowledge requirements of 10 CFR 55.41(b)(10), Administrative, normal, abnormal, and emergency operating procedures for the facility.

This question requires the candidate to have detailed knowledge of Step 23 of EOP-1.0, then to apply the Alternative Action, which directs a transition to EOP-2.5. This is a level of knowledge not required of an RO candidate.*

In addition, the stem of this question has psychometric flaws, as it introduced an unexpected thought process on the part of some examinees. While conditions in the stem support a loss-of-coolant accident (LOCA) outside of containment, the information in the stem is not definitive enough to allow the examinee to determine if the transition should occur during the diagnostic step (Step 14) for transition to EOP-2.0 or during Step 23. Lack of information regarding Reactor Building (RB) radiation levels, RB sump levels, Reactor Building Cooling Unit (RBCU) drain flow alarms made it impossible for the examinee to determine the time context in which to apply what *was* given in the body of the question. Given these psychometric flaws, the examinee may have felt that the conditions that existed at the time Step 14 was implemented *should* have dictated a transition to EOP-2.0. In that event, EOP-2.5 would eventually be implemented while in EOP-2.0. Even if the examinee recalled that EOP-2.5 would apply to the given conditions, and since the question stem asked which procedure would be implemented *next*, Choice A should be construed as correct.

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* For this reason, and because the question does not examine the appropriate 10 CFR 55.41 item, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ___________

EMERGENCY OPERATING PROCEDURE

EOP-1.0

REACTOR TRIP/SAFETY INJECTION ACTUATION

REVISION 19

SAFETY RELATED

Original signed by Baker — 11/13/03
DISCIPLINE SUPERVISOR — DATE

Original signed by Lippard — 11/14/03
APPROVAL AUTHORITY — DATE

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
14 **Check if the RCS is INTACT:**	14 **GO TO EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.** ☐
a. RB radiation levels are normal on:	
• RM-G7, CNTMT HI RNG GAMMA. ☐	
• RM-G18, CNTMT HI RNG GAMMA. ☐	
b. RB Sump levels are normal. ☐	
c. RB pressure is LESS THAN 1.5 psig. ☐	
d. The following annunciators are <u>NOT</u> lit:	
• XCP-606 2-2 (RBCU 1A/2A DRN FLO HI). ☐	
• XCP-607 2-2 (RBCU 1B/2B DRN FLO HI). ☐	
15 **Reset <u>both</u> SI RESET TRAIN A(B) Switches.** ☐	
16 **Reset Containment Isolation:**	
• RESET PHASE A - TRAIN A(B) CNTMT ISOL. ☐	
• RESET PHASE B - TRAIN A(B) CNTMT ISOL. ☐	
17 **Place <u>both</u> ESF LOADING SEQ A(B) RESETS to:**	
a. NON-ESF LCKOUTS. ☐	
b. AUTO-START BLOCKS. ☐	

REACTOR TRIP/SAFETY INJECTION ACTUATION

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
***21 Check SG levels:**	
a. Verify Narrow Range level in <u>all</u> SGs is GREATER THAN 30%. ☐	a. Maintain total EFW flow GREATER THAN 450 gpm until Narrow Range level is GREATER THAN 30% in at least <u>one</u> SG. ☐
b. Control EFW flow to maintain Narrow Range SG levels between 40% and 60%. ☐	b. <u>IF</u> Narrow Range level in <u>any</u> SG continues to increase in an uncontrolled manner, <u>THEN</u> **GO TO EOP-4.0, STEAM GENERATOR TUBE RUPTURE, Step 1.** ☐
22 **Check if Secondary activity is normal:**	22 **GO TO EOP-4.0, STEAM GENERATOR TUBE RUPTURE, Step 1.** ☐
a. Place SVX-9398A(B)(C), SG A(B)(C) SMPL ISOL, in AUTO. ☐	
b. Notify Chemistry to sample <u>all</u> SG secondary sides for abnormal activity. ☐	
23 **Check for loss of Reactor Coolant outside Containment:**	23 Evaluate the cause of abnormal AB conditions. <u>IF</u> the cause is a loss of RCS inventory outside Containment, <u>THEN</u> **GO TO EOP-2.5, LOCA OUTSIDE CONTAINMENT, Step 1.** ☐
a. Verify AB radiation levels are normal on:	
• RM-A3, MAIN PLANT VENT EXH ATMOS MONITOR: PARTICULATE, IODINE, GAS. ☐	
• RM-A13, PLANT VENT HI RANGE. ☐	
• RM-A11, AB VENT GAS ATMOS MONITOR. ☐	
• Local area monitors. ☐	
b. Verify annunciator XCP-631 6-1 is <u>NOT</u> lit (AB SMP LVL HI). ☐	
c. Verify annunciators XCP-606 3-4 and XCP-607 3-4 are <u>NOT</u> lit (LD TRBL AB SMP/FLDRN LVL HI). ☐	

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ___________

EMERGENCY OPERATING PROCEDURE

EOP-2.0

LOSS OF REACTOR OR SECONDARY COOLANT

REVISION 13

SAFETY RELATED

Original signed by Baker — 11/14/03
DISCIPLINE SUPERVISOR — DATE

Original signed by Lippard — 11/15/03
APPROVAL AUTHORITY — DATE

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
NOTE - Step 17 Presence of abnormally high levels of radioactivity in the AB indicates that a Containment breach may be in progress. Conditions for upgrading the Emergency status should be evaluated using EPP-001, ACTIVATION AND IMPLEMENTATION OF EMERGENCY PLAN.	
17 **Check the AB for evidence of ECCS leakage:** a. Verify AB radiation levels are normal on: • RM-A3, MAIN PLANT VENT EXH ATMOS MONITOR: PARTICULATE, IODINE, GAS ☐ • RM-A13, PLANT VENT HI RANGE. ☐ • RM-A11, AB VENT GAS ATMOS MONITOR. ☐ • Local area monitors. ☐ b. Verify annunciator XCP-631 6-1 is <u>NOT</u> lit (AB SMP LVL HI). ☐ c. Verify annunciators XCP-606 3-4 and XCP-607 3-4 are <u>NOT</u> lit (LD TRBL AB SMP/FLDRN LVL HI). ☐	17 Try to identify and isolate the leakage. <u>IF</u> the cause is a loss of RCS inventory outside Containment, <u>THEN</u> **GO TO EOP-2.5, LOCA OUTSIDE CONTAINMENT, Step 1.** ☐

Question 75

75. W/E15EK3.3 001

- Operators are responding to a Large Break LOCA
- RCS pressure blew down to Reactor Building pressure about 15 minutes ago.
- The crew has completed transferring both the Safety Injection System and RB Spray System to Cold Leg Recirculation mode per EOP-2.2, Transfer to Cold Leg Recirculation.
- The STA reports that reactor building sump level is 423 feet and increasing
- Annunciator XCP-604 point 3-1 **"SW FR RBCU 1A/2A FLO LO"** is illuminated
- Annunciator XCP-604 point 3-2 **"SW FR RBCU 1A/2A PRESS LO"** is illuminated

Which ONE of the following describes the event that has occurred, the action that is required to be performed, and the reason for performing the action?

A. The "A" SW Booster pump has tripped. Determine and correct the cause and start the "A" SW Booster pump. To ensure that reactor building integrity is maintained.
B. The "A" RBCU has ruptured. Secure and isolate the "A" SW Booster pump. To prevent flooding of vital equipment.
C. The "A" SW Booster pump has tripped. Ensure the "B" SW Booster pump is operating. To ensure that reactor building temperature is maintained within limits.
D. The "A" RBCU has ruptured. Secure and isolate the "A" SW Booster pump. To maintain reactor building integrity.

Feedback

DISTRACTORS:

A Incorrect, the RBCU has ruptured, the pump should be secured and isolated to reduce flooding.
B Correct, the RBCU has ruptured, and this is the correct action and reason.
C Incorrect, the RBCU has ruptured, the pump should be secured and isolated to reduce flooding. Sprays along with the other train of RBCU will maintain reactor building temperature.
D Incorrect, the "A" RBCU has ruptured, but the reason for isolation is to stop reactor building flooding and protect vital equipment.

REFERENCES:
1 EOP-17.1 "Response to Reactor Building Flooding".
2 APP-001 XCP-604 3-1
3 APP-001 XCP-604 3-2

K/A CATALOGUE QUESTION DESCRIPTION:
WE15EK3.3 Knowledge of the reasons for the following responses as they apply to the (Containment Flooding) Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations. (2.9/2.9)

<u>Question #75 (W/E15EK3.3 001)</u>

Recommendation:

Accept Choices B and D as the correct answer.

Justification:

As stated in the "Distractors" section: "B. Correct, the RBCU has ruptured." The question stem statement that the Reactor Building (RB) sump level is 423 feet would result in an Orange Path to EOP 17.1, "Response to Reactor Building Flooding." EOP-17.1 provides directions to identify unexpected sources of water by verifying a balance between the Service Water Booster Pump (SWBP) discharge flow and Service Water (SW) return flow to the pond for both trains. If there is not a balance in either train, directions are provided to stop the respective SWBP and isolate the SW train by closing both return isolation valves MVB-3106A (B) and MVB-3107A (B).

However, there also is an alternate answer. A rupture in the RBCU SW tubes would result in a containment barrier breach if not isolated. Training handout IB-1, Service Water System (Page 34), states: "The operation of the service water booster pumps during a [loss-of-coolant accident] LOCA must be verified to avoid the creation of a possible release pathway. Without the discharge pressure of the booster pumps, service water pressure inside the upper RCBU tubes would be less than the (post-accident) containment pressure. If any RCBU tubes leaked, fission products in the containment atmosphere could enter the tubes and be flushed out to the service water pond (and the environment)". When the SWBP breaker opens, valve MVG-3107A(B), which is the SW return from the RB, will close automatically. Choice D is also correct since securing the SWBP will result in isolating the RBCU to maintain containment integrity.

In conclusion, Choices B and D should be accepted as correct answers.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

EMERGENCY OPERATING PROCEDURE

EOP-17.1

RESPONSE TO REACTOR BUILDING FLOODING

REVISION 5

SAFETY RELATED

M E Eichhorst — 11/14/94

DISCIPLINE SUPERVISOR — DATE

Gary E. Williams — 11/17/94

APPROVAL AUTHORITY — DATE

RESPONSE TO REACTOR BUILDING FLOODING

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION

OPERATOR ACTIONS

NOTE

Conditions for implementing Emergency Plan Procedures should be evaluated using EPP-001, ACTIVATION AND IMPLEMENTATION OF EMERGENCY PLAN.

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
1 **Check the following to identify the unexpected source of water:**	
a. Verify RBCU supply flow equals return flow for <u>each</u> train: • Compare FI-4466, SWBP A DISCH FLOW, to FI-4468, FR LOOP A TO POND FLOW. ☐ • Compare FI-4496, SWBP B DISCH FLOW, to FI-4498, FR LOOP B TO POND FLOW. ☐	a. <u>IF</u> <u>either</u> flow is <u>NOT</u> balanced, <u>THEN</u>: 1) Stop the associated Service Water Booster Pump. ☐ 2) Isolate the associated Service Water train: • Ensure MVB-3106A, SWBP A DISCH, <u>AND</u> MVG-3107A, RBCU 64A/65A RTN TO SW POND, close. ☐ <u>OR</u> • Ensure MVB-3106B, SWBP B DISCH, <u>AND</u> MVG-3107B, RBCU 64B/65B RTN TO SW POND, close. ☐
b. Verify indications for CCW to RB loads are normal: • <u>All</u> indications on Moduflash M2/CC. ☐ • Annunciators on XCP-601. ☐ • Annunciators on XCP-602. ☐ • Annunciators on XCP-603. ☐	b. Use indications to determine the source of leakage. **REFER TO the appropriate ARP.** ☐
(Step 1 continued on next page)	(Step 1 continued on next page)

PANEL XCP-604
ANNUNCIATOR POINT 3-1

SW FR RBCU 1A/2A FLO LO	SETPOINT: 2000 gpm with MVG-3107A open	ORIGIN: IFB04468 33x contact on MVG-3107A

PROBABLE CAUSE:

1. Service Water Booster Pump A failure.
2. Motor operated valve failure.
3. Valve misalignment.
4. Reactor Building Cooling Unit tube leak.
5. Service Water piping leak or rupture.

AUTOMATIC ACTIONS:

1. None.

CORRECTIVE ACTIONS:

1. Verify Service Water Booster Pump operation is required.
2. Verify Service Water Booster Pump B is operating.
3. Monitor Reactor Building pressure and, if necessary, initiate Reactor Building Spray.
4. Monitor Reactor Building Sump level for evidence of Service Water leakage.

SUPPLEMENTAL ACTIONS:

1. Verify the following valve alignment using the status lights:
   Open: MVB-3106A, MVG-3108A, MVG-3108B, MVG-3103A, MVG-3107A, and either MVG-3109A or MVG-3109B.
   Closed: MVB-3110A, MVG-3111A, MVG-3112A.
2. If Service Water Booster Pump A failed, determine the cause and correct.
3. Verify proper valve alignment per SOP-117.
4. Refer to Technical Specification 3.7.4.

REFERENCES:

1. B-804-604, Sh. 2.
2. B-208-101, SW-70.
3. D-302-222.
4. SOP-117.
5. V.C. Summer Technical Specifications.

SW FR RBCU 1A/2A PRESS LO	<u>SETPOINT</u>: 90 psig decreasing with MVG-3107A open	<u>ORIGIN</u>: IPB04528 33x contact on MVG-3107A

PROBABLE CAUSE:

1. Service Water Booster Pump A failure.
2. Motor operated valve failure.
3. Valve misalignment.
4. Reactor Building Cooling Unit flow blockage.

AUTOMATIC ACTIONS:

1. None.

CORRECTIVE ACTIONS:

1. Verify Service Water Booster Pump operation is required.
2. Verify Service Water Booster Pump B is operating.
3. Monitor Reactor Building pressure and, if necessary, initiate Reactor Building Spray.
4. Monitor Reactor Building Sump level for evidence of Service Water leakage.

SUPPLEMENTAL ACTIONS:

1. Verify the following valve alignment using the status lights:
   Open: MVB-3106A, MVG-3108A, MVG-3108B, MVG-3103A, MVG-3107A and either MVG-3109A or MVG-3109B.
   Closed: MVB-3110A, MVG-3111A, MVG-3112A.
2. If Service Water Booster Pump A failed, determine the cause and correct.
3. Verify proper valve alignment per SOP-117.
4. Refer to Technical Specification 3.7.4.

REFERENCES:

1. B-804-604, Sh. 2.
2. B-208-101, SW-70.
3. D-302-222.
4. SOP-117.
5. V.C. Summer Technical Specifications.

NUCLEAR OPERATIONS TRAINING

INTERMEDIATE BUILDING SYSTEM

IB-1

SERVICE WATER SYSTEM

REVISION 16

Recommended	Original Signed by Riley R. Johnson	**Date**	01/10/06
Approved	Original Signed by Douglas O. Watson **Senior Instructor, Development**	**Date**	01/11/06

- An ESF bus blackout requires the normal ESF bus power to be established with the diesel generator breaker open. This condition allows the reset of auto start blocks (switches on the MCB) thus allowing auto start of the service water pumps when required.

A safety injection or blackout condition also bypasses the normal traveling screen controls and causes the screens to run continuously.

The ESFLS starts both service water booster pumps (step 7) and switches Reactor Building cooling units cooling supply from the Industrial Cooling Water System to the Service Water System. The Solid State Protection System aids in repositioning the valves necessary to complete this alignment, including the closure of the idle RBCU discharge valves.

The operation of the service water booster pumps during a LOCA must be verified to avoid the creation of a possible release pathway. Without the discharge pressure of the booster pumps, service water pressure inside the upper RCBU tubes would be less than the (post-accident) containment pressure. If any RCBU tubes leaked, fission products in the containment atmosphere could enter the tubes and be flushed out to the service water pond (and the environment). If the booster pump cannot be started, the SW return valves (3107A/B) must be shut to isolate the pathway.

Another special consideration for post accident operation is the high temperatures (~214°F) of the SW leaving the RBCUs. This temperature could result in flashing across the orifices in the RBCU discharge line. Flow should be checked to ensure that flow oscillations from flashing do not limit RBCU heat removal. One way to reduce flashing would be to initiate flow through the parallel (idle) RBCU to reduce the bulk temperature at the orifice.

Question 77

Question #77 (003A2.03 002)

The following conditions exist:

- Reactor Power is 9%.
- A Total Loss of All Service Water has occurred.
- AOP-117.1, "Total Loss of Service Water," has been entered.
- RCP temperatures are beginning to rise.
- Service Water can not be restored.

Which ONE of the following describes the action(s) the operators must take **and** the sequence of those actions (in accordance with AOP-117.1)?

A. Initiate a reactor plant shutdown per GOP-4B, POWER OPERATION (MODE 1 - DESCENDING). Stop up to TWO RCPs. Isolate unnecessary CCW loads, and ensure FS is aligned to the D/Gs. When an RCP motor bearing temperatures or lower seal water bearing temperature exceeds the specified limit, stop the affected RCP.

B. Initiate a reactor plant shutdown per GOP-4B, POWER OPERATION (MODE 1 - DESCENDING). Isolate unnecessary CCW loads, and ensure FS is aligned to the D/Gs. Secure an RCP only if motor bearing temperatures or lower seal water bearing temperature exceeds the specified limit.

C. Initiate a reactor plant shutdown per GOP-4B, POWER OPERATION (MODE 1 - DESCENDING). Stop at least TWO RCPs. Isolate unnecessary CCW loads, and ensure FS is aligned to the D/Gs. When the running RCP motor bearing temperatures or lower seal water bearing temperature exceeds the specified limit, increase monitoring and continue pump operation until the unit is shutdown then stop the affected pump.

D. Stop ONE RCP. Initiate a reactor plant shutdown per GOP-4B, POWER OPERATION (MODE 1 - DESCENDING). When an RCP motor bearing temperatures or lower seal water bearing temperature exceeds the specified limit, trip the reactor and stop the affected RCP.

Distractor Analysis:

A. Correct: IAW AOP-117.1, the reactor should be shutdown (not tripped). Secure up to TWO RCPs (Step 12). The affected RCP should be shutdown if RCP motor bearing temperatures exceeds **195 ^{o}F** or lower seal water bearing temperature exceeds **225^{o}F** (Step 13).

B. Incorrect: Do not wait until temperature are exceeded to secure RCPs

C. Incorrect: Step 12 allows two RCPs to be stopped if plant conditions permit. Prudent action is to shutdown with 2 RCPs running and secure one if necessary for temperature.

D. Incorrect: Shutdown is initiated in step 3 and Step 12 secures the RCP. Reactor is not tripped unless above P-7.

References:
GOP-4B
AOP-117.1, page 8
AOP-118.1, page 5

<u>K/A CATALOGUE QUESTION DESCRIPTION</u>:
- Reactor Coolant Pump System (RCPS); Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems.

Question #77 (003A2.03 002)

Recommendation:

Delete the question.

Justification:

Choice A is the designated correct answer; however, it is incorrect due the sequence of the actions not being correct. The correct order of actions is as follows in accordance with AOP-117.1:

1) Initiate a reactor shutdown per GOP-4B, POWER OPERATION (MODE 1-DESCENDING.
2) Isolate unnecessary CCW loads
3) Ensure FS is aligned to the D/G's
4) Stop up to TWO RCPs
5) When an RCP motor bearing temperatures or lower seal water bearing temperature exceeds the specified limit, stop the affected RCP

Choices B, C, and D are incorrect as stated in the Distractor Analysis.

The information needed to answer this question is initiated as Alternate Step 3 in AOP 117.1 which is designated as a Continuous Use Procedure.* Only Step 1 of the AOP is designated as an immediate action where the operator would be expected to take some compensatory action. This Step has the operator determine the cause for loss of Service Water (SW) Pumps, no active actions are taken outside of referencing ARPs.

This question does not match the K/A in that it is related to a loss of SW, not a Reactor Coolant Pump (RCP) motor bearing, system level, failure. The actual K/A tested is APE 062 G2.4.7, for knowledge of the Loss of Service Water AOP. To meet the K/A, the item should have tested a component failure that would be covered by an RCP failure AOP.

In conclusion, this question should be deleted for the following reasons:

1. It has no correct answer.
2. This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.
3. The question does not match the K/A.

*See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ____________

ABNORMAL OPERATING PROCEDURE

AOP-117.1

TOTAL LOSS OF SERVICE WATER

REVISION 3

SAFETY RELATED

Original signed by D. A. Baker — 8/20/02
DISCIPLINE SUPERVISOR — DATE

Original signed by Dan Gatlin — 8/21/02
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLED DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLED DATE

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing.

TOTAL LOSS OF SERVICE WATER

TABLE OF CONTENTS

TOTAL LOSS OF SERVICE WATER

REFERENCES

1. Tech Spec 3.7.4.
2. FSAR 9.2.1.
3. DBD, Service Water System.
4. SOP-117, Service Water System.
5. D-302-221 and 222.

COMMITMENTS

1. C01 - NRC Security Order (February 25, 2002): Entire procedure.

REVISION SUMMARY

Incorporated Change A of Revision 2. Added SCOPE Section to procedure and added Commitment C01 for NRC Security Order (February 25, 2002).

TOTAL LOSS OF SERVICE WATER

PURPOSE

This procedure provides instructions for responding to Service Water System malfunctions including pump failure, loss of pressure, Traveling Screen obstruction, and loss of inventory.

SCOPE

The regulatory processes 10CFR50 Appendix B, 10CFR50.59, and SAP-630 apply to this procedure.

SYMPTOMS/ENTRY CONDITIONS

1. Trip of running Service Water Pumps.
2. Service Water Pump discharge pressure less than 20 psig as indicated on the following:
   - PI-4402, PP A DISCH PRESS PSIG.
   - PI-4442, PP C DISCH PRESS PSIG.
   - PI-4443, PP C DISCH PRESS PSIG.
   - PI-4422, PP B DISCH PRESS PSIG.
3. Low Service Water Pond level.
4. Increased temperature of Service Water System loads.
5. Various Service Water System low flow and/or pressure alarms with corresponding Moduflash points in alarm.
6. Visual indication of a Service Water System leak.
7. Any of the following Main Control Board annunciators in alarm:
   - SWP A(B)/C DISCH PRESS LO (XCP-604(605) 1-4).
   - SWP A(B)/C TRIP (XCP-604(605) 1-2).
   - SWP A(B)/C AUTOSTART FAIL (XCP-604(605) 1-1).

TOTAL LOSS OF SERVICE WATER

ACTION/EXPECTED RESPONSE		ALTERNATIVE ACTION	
OPERATOR ACTIONS			
1 **Determine the cause for the loss of Service Water Pumps:**			
a. Check for annunciators on XCP-603, 604, and 605.	☐		
b. **REFER TO the appropriate ARPs.**	☐		
c. Attempt to correct the cause for loss of Service Water.	☐		
2 **Start <u>both</u> Service Water loops. REFER TO SOP-117, SERVICE WATER SYSTEM.**	☐	2 <u>IF</u> <u>either</u> Service Water Pump can <u>NOT</u> be started, <u>THEN</u> start the spare Service Water Pump. **REFER TO SOP-117, SERVICE WATER SYSTEM.**	☐
3 **Check if at least <u>one</u> Service Water loop is restored.**	☐	3 <u>IF</u> <u>no</u> Service Water loop can be restored, <u>THEN</u> perform the following:	
		a) **REFER TO** the following while continuing with this procedure:	
		• **AOP-118.1, TOTAL LOSS OF COMPONENT COOLING WATER.**	☐
		• **AOP-501.2, TOTAL LOSS OF CHILLED WATER.**	☐
		b) Initiate plant shutdown. **REFER TO GOP-4, POWER OPERATION (MODE 1).**	☐
		c) Isolate unnecessary CCW System heat loads.	☐
		d) **GO TO Step 9.**	☐

TOTAL LOSS OF SERVICE WATER

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
4 **Verify the operating Service Water Pump discharge pressure is GREATER THAN 50 psig.** ☐	4 Perform the following: a) Dispatch an operator to the Service Water Pump House to perform the following: 1) Check the Traveling Screens for heavy debris. ☐ 2) Check the Service Water Screen Wash System for proper operation. ☐ 3) <u>IF</u> the Screen Wash System is <u>NOT</u> operating, <u>THEN</u> manually wash the Traveling Screens. ☐ b) Check for Service Water System pipe ruptures: 1) Identify and isolate <u>any</u> Service Water System leakage paths. ☐ 2) Consult with the Rad Waste operator for symptoms of excessive drainage or leakage. ☐ c) Check the Service Water Pond level: • LI-4418, LEVEL FEET. ☐ • LI-4458, LEVEL FEET. ☐ d) **GO TO Step 7.** ☐
5 **Start the HVAC Chilled Water System on the operating Service Water loop. REFER TO SOP-501, HVAC CHILLED WATER SYSTEM.** ☐	
6 **Transfer the CCW Active Loop to the operating Service Water loop. REFER TO SOP-118, COMPONENT COOLING WATER.** ☐	

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
7 **Check if the Service Water System operation is normal:**	7 **GO TO Step 9.** ☐
a. Check if at least <u>one</u> Service Water loop is restored. ☐	
b. Verify operating Service Water Pump discharge pressure is GREATER THAN 50 psig. ☐	
8 **RETURN TO the Procedure and Step in effect.** ☐	
* 9 **Continue efforts to restore a Service Water loop to service.** ☐	
10 **Align cooling water to <u>each</u> operating Diesel Generator:**	10 <u>IF</u> <u>no</u> DG is operating, <u>THEN</u> **GO TO Step 11.** ☐
a. Open PVG-3105A-SW, FS TO DG A. ☐	
b. Open PVG-3105B-SW, FS TO DG B. ☐	
c. Verify cooling flow through the Diesel Generator coolers:	c. Stop <u>each</u> operating Diesel Generator with <u>no</u> cooling flow indicated. ☐
• FM-4462, DG A CLG (Moduflash M3/SW Pt #16). ☐	
• FM-4492, DG B CLG (Moduflash M4/SW Pt #16). ☐	
d. Locally monitor temperatures on <u>each</u> operating Diesel Generator. ☐	
11 **Check if <u>any</u> RCPs are running.** ☐	11 **GO TO Step 14.** ☐
12 **When plant conditions allow, stop up to <u>two</u> RCPs.** ☐	

TOTAL LOSS OF SERVICE WATER

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
NOTE - Step 13 "AFFECTED" refers to <u>any</u> RCP with bearing temperatures exceeding the limits.	
***13 Check Reactor Coolant Pump temperatures:**	
a. Check if <u>any</u> Reactor Coolant Pumps are running. ☐	a. **GO TO Step 14.** ☐
b. Display Dedicated Display ZZRCPBRG on the IPCS to monitor RCP bearing temperatures: • RCP motor bearing temperatures are LESS THAN 195°F. ☐ • Lower Seal Water Bearing temperature is LESS THAN 225°F. ☐	b. <u>IF</u> Reactor power is GREATER THAN P-7, <u>THEN</u> perform the following: 1) Trip the Reactor. ☐ 2) Stop the AFFECTED RCP(s). ☐ 3) **GO TO EOP-1.0, REACTOR TRIP/SAFETY INJECTION ACTUATION.** ☐ <u>IF</u> Reactor power is LESS THAN P-7, <u>THEN</u> perform the following: 1) Stop the AFFECTED RCP(s). ☐ 2) Initiate plant shutdown. **REFER TO the appropriate GOP.** ☐
14 Alternate operation of Component Cooling Water loops. ☐	
15 Check if the RHR System is operating. ☐	15 **GO TO Step 17.** ☐

TOTAL LOSS OF SERVICE WATER

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
16 **Check if RCS temperatures are stable.** ☐	16 IF RCS temperatures are increasing, THEN **GO TO the applicable AOP** for present plant conditions: • **AOP-115.3, LOSS OF RHR WITH THE RCS INTACT.** ☐ • **AOP-115.4, LOSS OF RHR WHILE REFUELING.** ☐ • **AOP-115.5, LOSS OF RHR WITH THE RCS NOT INTACT (MODE 5).** ☐
17 **Verify at least one Service Water loop is restored.** ☐	17 **RETURN TO Step 9.** ☐
18 **RETURN TO Step 4.** ☐	

End of AOP-117.1

Question 80

80. 007EA2.01 002

At 50% power, the plant experienced a loss of BOTH running Main Feedwater Pumps with a concurrent failure of the Reactor trip breaker A to open. The crew is performing the immediate actions of EOP-1.0, "Reactor Trip/Safety Injection Actuation."

Current plant conditions are as follows:

- The Integrated Plant Computer System has failed.
- SG LO-LO Level annunciators are lit.
- Reactor Power is 7% and slowly decreasing.
- All EFW Pumps failed to start.

Which ONE of the following describes the procedure path based on the above information?

A. Remain in EOP-1.0, until directed to monitor Critical Safety Functions then transition to EOP-15.0, "Response To Loss of Secondary Heat Sink."
B. Directly enter EOP-15.0, "Response To Loss of Secondary Heat Sink."
C. Remain in EOP-1.0, until directed to monitor Critical Safety Functions then transition to EOP-13.0, "Response To Abnormal Nuclear Power Generation."
D. Transition from EOP-1.0 to EOP-13.0, "Response To Abnormal Nuclear Power Generation."

Feedback

DISTRACTORS:
A. INCORRECT Should transition directly to EOP-13.0.

B. INCORRECT Should transition directly to EOP-13.0.

C. INCORRECT Should transition directly to EOP-13.0.

D. CORRECT Should transition directly to EOP-13.0. Since there are no given conditions that would warrant an SI, the CRS should follow the Alternative Action for Step 5 of EOP-1.0 and transition to EOP-1.1. Upon transition from EOP-1.0, the STA begins monitoring of CSFs, and should infrom CRS of Red path to EOP-13.0 based on power >5%.

REFERENCES:
1.

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Trip; Ability to determine or interpret the following as they apply to a reactor trip: Decreasing power level, from available indications.

Question #80 (007EA2.01 002)

Recommendation:

Accept Choices C and D as correct answers.

Justification:

Assuming an examinee recalls the procedure in sufficient detail to analyze this question, with the given conditions, the examinee cannot determine if the crew is at an Action/Expected Response (left-hand column) for Steps 2 - 5 or the Alternative Action for Step 5 in EOP-1.0.

If the examine assumes that the crew has reached the Alternative Action for Step 5, then Choice D is correct for the reasons already stipulated in the Reference section. Specifically, the Feedback section states: "Should transition directly to EOP-13.0. Since there are no given conditions that would warrant an SI, the CRS should follow the Alternative Action for Step 5 of EOP-1.0 and transition to EOP-1.1. Upon transition from EOP-1.0, the STA begins monitoring of CSFs, and should inform CRS of Red path to EOP-13.0 based on power >5%."

Also note that Step 1 of EOP-1.0 does not include a check of actual power <5%, only a decreasing power. An immediate transition to EOP-13.0 would only occur after monitoring of CSFs had begun.

If the examinee determines that the wording in the question stem "is performing the immediate actions" means the crew is somewhere between Steps 2 and 5, then the examinee may select Choice C. For example, if the crew were verifying the Turbine Trip; they would remain in EOP-1.0, perform Steps 2-5, then transition to EOP-1.1. Per EOP usage rules, the crew would then be "directed" to monitor CSFs and then transition to EOP-13.0 based on the Red path.

Additionally, there is a psychometric flaw in this question that could lead to the examinee choosing Choice A. Although we do not believe that this is enough justification to accept Choice A as a correct answer, the examinee may assume that, if the crew has just started Step 2, then power may (or should) be below 5% by the time they reached Step 5. In that case, the crew should transition to EOP-15.0 based on the given conditions of LO-LO SG [steam generator] levels and emergency feedwater pump failures. Lack of a given rate of power decrease, and lack of additional information (such as indications of other events that could cause reactor power to be above 5%), both of which would allow the examinee to predict if power *could* decrease below 5% by the time the Immediate Operator Actions were complete, contributed to the ambiguity.

This question does not match the K/A as it does not require the applicant to determine power decrease in relation to a reactor trip. As written, this question appears only to evaluate whether 7% power indication is a reason to transition to the Abnormal Transient Without Scram (ATWS) procedure. The K/A for this test item is EPE 029 G2.4.4 for ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. Reactor power is given in the stem; therefore, no determination is required. Additionally, a reactor trip has <u>not</u> occurred; rather, an ATWS has occurred.

In conclusion, Choices C or D should be accepted as the correct answer.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ____________

EMERGENCY OPERATING PROCEDURE

EOP-1.0

REACTOR TRIP/SAFETY INJECTION ACTUATION

REVISION 19

SAFETY RELATED

Original signed by Baker	11/13/03
DISCIPLINE SUPERVISOR	DATE
Original signed by Lippard	11/14/03
APPROVAL AUTHORITY	DATE

TABLE OF CONTENTS

PURPOSE

This procedure provides instructions to:

- Verify proper response of the automatic protection systems following manual or automatic actuation of a Reactor Trip or Safety Injection.
- Assess plant conditions.
- Identify the appropriate recovery procedure.

SYMPTOMS/ENTRY CONDITIONS

1. Attachment 1 provides a list of symptoms which require a Reactor Trip, if a trip has NOT occurred.
2. The following are symptoms of a Reactor Trip:
    - Any red first-out Reactor Trip annunciator lit.
    - Rapid decrease in neutron flux level as indicated by the NI System.
    - Shutdown and Control Rods fully inserted. Rod Bottom Lights lit.
3. Attachment 2 provides a list of symptoms which require a Reactor Trip AND Safety Injection, if a trip AND SI have NOT occurred.
4. The following are symptoms of a Reactor Trip AND Safety Injection:
    - Any red first-out Safety Injection annunciator lit.
    - Both Charging AND RHR Pumps running.
    - Actuation of the SI ACT status light.
    - Actuation of the ESF LOADING SEQUENCER status lights.
    - Actuation of SAFETY INJECTION/PHASE A ISOL status lights.

1 **RCP TRIP CRITERIA**

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1400 psig.

2 **REDUCING CONTROL ROOM EMERGENCY VENTILATION**

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

REACTOR TRIP/SAFETY INJECTION ACTUATION

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION

OPERATOR ACTIONS

NOTE

- **Steps 1 through 5** are Immediate Operator Actions.
- The EOP REFERENCE PAGE should be monitored throughout the use of this procedure.
- Conditions for implementing Emergency Plan Procedures should be evaluated using EPP-001, ACTIVATION AND IMPLEMENTATION OF EMERGENCY PLAN.

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
1 **Verify Reactor Trip:**	1 IF the Reactor will NOT trip using both Reactor Trip Switches, OR is NOT subcritical, THEN **GO TO EOP-13.0, RESPONSE TO ABNORMAL NUCLEAR POWER GENERATION, Step 1.** ☐
• Trip the Reactor using either Reactor Trip Switch. ☐	
• Verify all Reactor Trip and Bypass Breakers are open. ☐	
• Verify all Rod Bottom Lights are lit. ☐	
• Verify Reactor Power level is decreasing. ☐	

1 RCP TRIP CRITERIA

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1400 psig.

2 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
2 Verify Turbine/Generator Trip:	
a. Verify all Turbine STM STOP VLVs are closed. ☐	a. Trip the Turbine. ☐ IF the Turbine will NOT trip, THEN: • Set LOAD LMT SET fully Counterclockwise. ☐ • Place EHC Pumps A and B in PULL TO LK NON-A. ☐ • Locally trip the Main Turbine from the Turbine Front Standard (TB-463). ☐
b. Ensure Generator Trip (after 30 second delay): 1) Ensure the GEN BKR is open. ☐ 2) Ensure the GEN FIELD BKR is open. ☐ 3) Ensure the EXC FIELD CNTRL is tripped. ☐	

1 RCP TRIP CRITERIA

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1400 psig.

2 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
3 **Verify both ESF buses are energized.** ☐	3 Perform the following: a) Verify at least one ESF bus is energized: • 7.2 KV BUS 1DA is energized. ☐ OR • 7.2 KV BUS 1DB is energized. ☐ IF no ESF bus is energized, THEN try to restore power to at least one ESF bus using the DG. ☐ IF power can NOT be restored to at least one ESF bus, THEN **GO TO EOP-6.0, LOSS OF ALL ESF AC POWER, Step 1.** ☐ b) Try to restore power to the deenergized bus while continuing with this procedure. **REFER TO AOP-304.1, LOSS OF BUS 1DA(1DB) WITH THE DIESEL NOT AVAILABLE.** ☐
4 **Check if SI is actuated:**	
a. Check if either: • SI ACT status light is bright on XCP-6107 1-1. ☐ OR • Any red first-out SI annunciator is lit on XCP-626 top row. ☐	a. **GO TO Step 5.** ☐
b. Actuate SI using either SI ACTUATION Switch. ☐	
c. **GO TO Step 6.** ☐	

1 **RCP TRIP CRITERIA**

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1400 psig.

2 **REDUCING CONTROL ROOM EMERGENCY VENTILATION**

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

REACTOR TRIP/SAFETY INJECTION ACTUATION

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
5 **Check if SI is required:**	
a. Check if <u>any</u> of the following conditions exist:	a. **GO TO EOP-1.1, REACTOR TRIP RECOVERY, Step 1.** ☐
• PZR pressure LESS THAN 1850 psig. ☐	
<u>OR</u>	
• RB pressure GREATER THAN 3.6 psig. ☐	
<u>OR</u>	
• Steamline pressure LESS THAN 675 psig. ☐	
<u>OR</u>	
• Steamline differential pressure GREATER THAN 97 psid. ☐	
b. Actuate SI using <u>either</u> SI ACTUATION Switch. ☐	
6 **Initiate ATTACHMENT 3, SI EQUIPMENT VERIFICATION.** ☐	
7 **Announce plant conditions over the page system.** ☐	

Note to NRC Exam Reviewer:

Per the Justification for Question #80, this step (EOP-1.0 Step 5 Alternative Action) delineates where the candidate could have chosen to transition to EOP-13.0 or EOP-15.0 based on his/her interpretation of current plant conditions.

1 **RCP TRIP CRITERIA**

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1400 psig.

2 **REDUCING CONTROL ROOM EMERGENCY VENTILATION**

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ____________

EMERGENCY OPERATING PROCEDURE

EOP-12.0

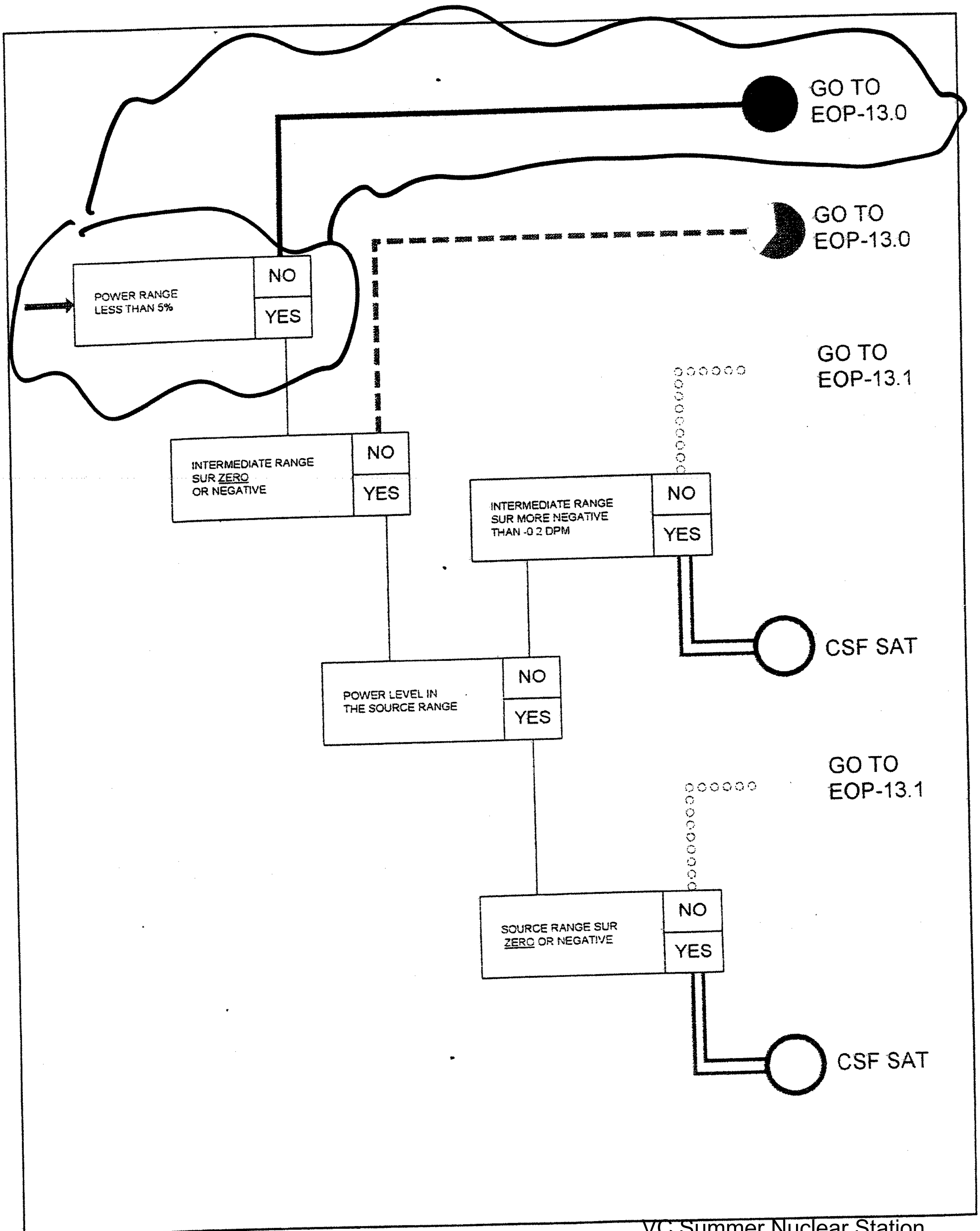
MONITORING OF CRITICAL SAFETY FUNCTIONS

REVISION 12

SAFETY RELATED

Original signed by Crider	10/21/00
DISCIPLINE SUPERVISOR	DATE
Original signed by Goldston	12/27/00
APPROVAL AUTHORITY	DATE

SUBCRITICALITY



SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.________

OPERATIONS ADMINISTRATIVE PROCEDURE

OAP-103.4

EOP/AOP USER'S GUIDE

REVISION 0

SAFETY RELATED

ORIGINAL SIGNED BY: D. A. BAKER — **04/17/03**
DISCIPLINE SUPERVISOR — DATE

ORIGINAL SIGNED BY: G. A. LIPPARD — **04/17/03**
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE

INFORMATION USE

Procedure May Be Performed From Memory.
User Retains Accountability for Proper Performance.

k. The Shift Engineer may evaluate any procedure referenced by a yellow path, to assist the CRS in determining the advisability of its use.

l. Procedures referenced by a yellow path are entered at the discretion of the CRS. While using a procedure based on a yellow path, all reference page actions and continuous actions that were in effect should be monitored and performed as applicable. While these FRGs should normally be followed to their completion, if the CRS determines that other actions would be more effective in protecting the plant, the procedure and step in effect may be returned to at any time.

m. The Shift Engineer should begin monitoring EOP-12.0 as directed by the EOP or when a transition is made from EOP-1.0 to another EOP. Monitoring should continue until normal plant conditions are restored or the plant is in Cold Shutdown.

n. The Status Trees should be monitored continuously if any red or orange path conditions are found to exist. If all parameters indicate a yellow or green path condition, then monitoring frequency may be reduced to every 10 to 20 minutes, unless some significant change in plant status occurs.

o. EOP-12.0 is monitored for information only while in EOP-6.0 due to a loss of ESF electrical power. Procedures referenced by EOP-12.0 are implemented after transitioning out of EOP-6.0 when directed by EOP-6.1 or EOP-6.2. This is further reinforced via notes in all EOP-6 series procedures.

7.0 RECORDS

7.1 Records required due to EOP implementation will be per SAP-116.

8.0 REVISION SUMMARY

8.1 Revision 0 is a new procedure derived from OAG-103.4, Revision 3 Change A.

8.2 Added Scope statement to procedure.

8.3 Clarified Steps 6.11.a and 6.11.b for RCP Trip Criteria Applicability per procedure feedbacks.

Question 84

84. 032AA2.08 003

Refueling operations are in progress, with SR monitor N33 out of service, when power is suddenly lost to source range neutron flux monitor N31 and subsequently regained 30 minutes later.

Which ONE of the following describes the action to be taken for this situation when power is lost?

A. Suspend all core alterations and perform an analog channel operational test of source range neutron flux monitor N31 within 8 hours prior to the initial start of core alterations.
B. Suspend all core alterations and perform a neutron flux response time test AND operational test of source range neutron flux detector N31 within 8 hours prior to the initial start of core alterations.
C. Determine boron concentration and perform a channel check of source range neutron flux monitor N31 within 12 hours.
D. Determine boron concentration and perform a neutron flux response time test of source range neutron flux detector N31 within 12 hours.

Feedback

DISTRACTORS:

A. CORRECT T.S. 3.9.2 requires immediate suspension of CORE ALTERATIONS when one of the two SR monitors are lost
B. INCORRECT Per T.S. Table 3.3-2 (* and Note 1), neutron detectors (not the channel) are exempt from response time testing.
C. INCORRECT Boron concentration measurements are only required when both monitors are down.
D. INCORRECT Boron concentration measurements are only required when both monitors are down. Neutron detectors are exempt from response time testing.

REFERENCES:
1 TS 3.9.2, "Instrumentation."
2 TS 3.9.1, "Boron Concentration."
3 TS Table 3.3-2, "Reactor Trip System Instrumentation Response Times."
4 IC-8, "Nuclear Instrumentation," pages 24, 48, & 50.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Source Range Nuclear Instrumentation; Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Testing required if power is lost, then restored.

<u>Question #84 (032AA2.08 003)</u>

Recommendation:

Delete the question.

Justification:

The immediate operator actions of AOP-401.9, Source Range Channel Failure (a Continuous Use Procedure), are to suspend all core alterations and stop all positive reactivity additions. Actions beyond the immediate actions are not required to be committed to memory unless they are needed to stabilize the plant; however, all of the choices reflect actions beyond those needed to stabilize the plant.*

In order to make a distinction between Choices A and C, the examinee is required to recall that Technical Specifications Table 3.3-2, Note 1, exempts the neutron *detectors,* not the *monitors*, from time response testing. As such, this question inappropriately examines minutia (see Form ES-401-9). It also inappropriately examines actions related to retest (surveillances) activities, which should not be required of the candidate.

This question should be deleted in that it inappropriately examines actions that are beyond the level of knowledge expected of operators.

* See Section II.A, above.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. ____________

ABNORMAL OPERATING PROCEDURE

AOP-401.9

SOURCE RANGE CHANNEL FAILURE

REVISION 4

SAFETY RELATED

Original Signed by Dennis Baker — 01/13/05
DISCIPLINE SUPERVISOR — DATE

Original Signed by Robert Ray — 01/18/05
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLED DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLED DATE

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing.

SOURCE RANGE CHANNEL FAILURE

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
OPERATOR ACTIONS	
1 Stop all core alterations. ☐	
2 Stop all positive reactivity additions. ☐	
3 Verify NI-31 OR NI-32 is operable. ☐	3 Initiate GTP-702, Attachment VI.L-1, No Operable Source Range Neutron Flux Channel. ☐ Initiate GTP-702, Attachment VI.L-2, Inoperable High Flux at Shutdown Alarms - Modes 3,4,5,6. ☐
4 Check if the Reactor Building evacuation alarm has actuated: • SR HI FLUX AT SHUTDN (XCP-620 4-2 or 4-3), annunciator is lit. ☐ OR • Report from the RB Coordinator. ☐	4 GO TO Step 6. ☐
5 Inform plant personnel that the Reactor Building evacuation alarm was spurious, and evacuation of the RB is NOT required. ☐	
6 Bypass the failed Source Range channel: a. Place LEVEL TRIP Switch for the AFFECTED channel in BYPASS. ☐ b. Verify IR&SR TRIP BYP (XCP-620 4-5), annunciator is lit. ☐	

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	≤ 0.5 seconds(1)
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Deleted	
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	≤ 0.5 seconds(1)
7.	Overtemperature ΔT	≤ 8.5 seconds(1)(2)
8.	Overpower ΔT	≤ 8.5 seconds(1)(2)
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

(1) Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

(2) The 8.5 second response time includes a 5.0 second delay for the RTDs mounted in thermowells.

Question 87

87. 068G2.1.20 002

A Liquid Radwaste Release is been in progress:

- XCP-646 2-5, MON TK DISCH RM-L5 HI RAD, has just actuated for the second time.
- RCV00018-WL, Liquid Radioactive Waste Control Valve, indicates shut.
- Within 30 seconds of the alarm , RM-L5's reading returns to below the setpoint.

Which ONE of the following correctly states the next procedure steps to be taken.

A. The tank must be sampled and activity levels verified, then open RCV00018-WL and resume the release per SOP-108.
B. Verify that the RM-L5's reading is below the setpoint, then open RCV00018-WL and resume the release per SOP-108.
C. Verify that the RM-L5's reading is below the setpoint, then open RCV00018-WL and resume the release per SOP-108. Direct Heath Physics to continue to monitor the release and reduce the release rate.
D. Notify Health Physics and request a radiological survey. The release can not be reinitiated under the current release permit.

Feedback

DISTRACTORS:
A. CORRECT As per XCP-646-2-5, this is the first step of the supplemental actions.
B. INCORRECT This is the action if this is the first time the release has been automatically terminated.
C. INCORRECT This is the action if this is the first time the release has been automatically terminated, coupled with the actions for a malfunctioning RM-L5.
D. INCORRECT This would be plausible if it is believed that the release can not be continued.

REFERENCES:
1 XCP-646 2-5 & 2-6, pages 12 & 13.
2 XCP-644 2-5, page 15.
3 XCP-643 4-1, page 22.

K/A CATALOGUE QUESTION DESCRIPTION:
- Liquid Radwaste System; Ability to execute procedure steps.

Question #87 (068G2.1.20 002)

Recommendation:

Delete the question.

Justification:

HPP-710 contains more definitive requirements than SOP-108 for a second termination of a release. Section 5.3.3 of HPP-710 states: "If the release is terminated a second time by a radiation monitor high alarm, notify the Count Room, verify proper valve lineup, return tank to recirculation, begin waste line flush, and terminate the LWRP [liquid radwaste release permit] and the Request for Redundant Analysis. Generate a CER to evaluate the condition."

In addition, SOP-108, Liquid Waste Processing System, attachment VA, which is a Continuous Use Procedure being used by the Control Room crew, states that for a second spike occurrence, contact the Count Room to initiate a Request for Redundant Analysis. Comparing these actions to the possible choices indicates that there is no completely correct answer for this question and it should, therefore, be deleted on this basis alone. The question is better suited as a JPM rather than question material.

In addition, the information needed to answer this question is contained in Supplemental Actions of an Annunciator Response Procedure (ARP). In order to derive the correct answer, and to eliminate Choice B, the examinee not only would be required to recall a Supplemental Action from memory, but also to recall that another Supplemental Action applied to a single termination and recall that there were differences in the subsequent actions between the two steps. The application of procedures at this level is beyond the expectations for operators to recall from memory.*

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

HEALTH PHYSICS PROCEDURE

HPP-710

SAMPLING AND RELEASE OF RADIOACTIVE LIQUID EFFLUENTS

REVISION 11

SAFETY RELATED

Original Signed by M. Coleman 12/13/02
DISCIPLINE SUPERVISOR DATE

Original Signed by L.A. Blue 12/16/02
APPROVAL AUTHORITY DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	09/23/03					

INFORMATION USE

Procedure May Be Performed From Memory.
User Retains Accountability for Proper Performance.

1. If redundant analysis results (i.e., total gamma concentration) are within 20% of the original analysis result, proceed to Section 5.3.1.D.

2. If redundant analysis results (i.e., total gamma concentration) exceeds 20% of the original analysis results, terminate the existing LWRP and initiate a new LWRP in accordance with this procedure.

D. The Count Room should review previously established set points. If the set points were set significantly more conservative than maximum allowed by the LWRP, increase to four (4) times radiation monitor expected response, not to exceed the computer generated set point..

E. If release restart is acceptable, the Count Room should ensure that Parts I and II of the Request for Redundant Analysis are completed and transmitted to Operations. A copy should be retained with the active copy of the LWRP.

5.3.2 Operations may now terminate the flush in progress, as applicable, and restart the release using new set points established on the Request for Redundant Analysis.

A. Routinely observe the monitor(s) during the release and note any abnormal or unusual fluctuations.

B. Record release data on Part III of the Request for Redundant Analysis and transmit to the Count Room along with the LWRP after the release and flush are completed.

5.3.3 If the release is terminated a second time by a radiation monitor high alarm, notify the Count Room, verify proper valve lineup, return tank to recirculation, begin waste line flush, and terminate the LWRP and the Request for Redundant Analysis. Generate a CER to evaluate the condition.

C01→5.4 Continuous Releases Terminated by Radiation Monitor High Alarm

5.4.1 Any time a radiation monitor high alarm terminates a release (RM-L3 or RM-L10 for SGBD, RM-L8 for TBS), the following actions shall be performed.

A. Verify with Operations that the release has been diverted or stopped in accordance with Reference 3.10.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

SYSTEM OPERATING PROCEDURE

SOP-108

LIQUID WASTE PROCESSING SYSTEM

REVISION 22

QUALITY RELATED

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing.

INITIALS

d. If both inline RMLs are inoperable, perform redundant sample per HPP-710. Refer to Offsite Dose Calculation Manual Limiting Condition for Operation 1.1.1.1. ________

e. Ensure the Radiation Monitor setpoints are adjusted per the LWRP. ________

f. Ensure the recorder for the RML to be monitored is operating:

1) R/R #5 for RML-9. ________

2) R/R #10 for RML-5 or RML-7. ________

3. Verify Fairfield Pumped Storage is in the Generate Mode. ________

4. Notify the Dispatcher of release. ________

5. If at anytime a release is terminated by the radiation monitor(s), perform the following as applicable:

a. If the release was terminated due to a spike in the Radiation Monitor, re-establish the release after radiation levels decrease below the setpoint (This is applicable only for a single spike. If a second spike occurs, go to Step 5.c). ________

b. If the termination was not due to a spike, perform the following prior to re-establishing the release:

1) Perform a redundant valve lineup and correct as necessary. ________

2) Contact the Count Room to verify the correct setpoints and reset as necessary. ________

3) If required, flush the discharge line up to RM-L5 and/or RM-L9 per Attachment VB. ________

c. If Steps 5.a and 5.b do not address the cause for termination <u>or a second spike occurs</u>, contact the Count Room to initiate a request for redundant analysis. ________

6. When notified by the local operator that the required volume of water has been released, verify Radiation Monitors return to normal background. ________

7. If Radiation Monitors do not return to normal, flush per SOP-124. ________

8. Notify the Dispatcher of completion of release. ________

PANEL XCP-646
ANNUNCIATOR POINT 2-5

MON TK DISCH RM-L5 HI RAD	SETPOINT: Refer to the LWRP	ORIGIN: RM-L5

PROBABLE CAUSE:

1. If a planned release is in progress, the following may have occurred:
   a. Release limits have been exceeded.
   b. Release flow rate has been exceeded.
2. An unplanned release is in progress.

AUTOMATIC ACTIONS:

1. RCV00018-WL, LIQUID RADIOACTIVE WASTE CONTROL VALVE, trips closed.

CORRECTIVE ACTIONS:

1. Verify the Automatic Action has occurred.
2. Verify the alarm is valid by observing RM-L5 and R/R-10 for increasing radiation.

SUPPLEMENTAL ACTIONS:

1. If the release is isolated on High Radiation and radiation levels immediately return below the setpoint, open RCV00018-WL, LIQUID RADIOACTIVE WASTE CONTROL VALVE, and resume the release per SOP-108. CHG B
2. If the release is isolated a second time or if radiation levels do not return below the setpoint, have the tank sampled before resuming the release.
3. Notify Health Physics and request a radiological survey.
4. If the alarm is invalid, remove RM-L5 from service per SOP-124 and refer to the ODCM.

REFERENCES:

1. D-302-362.
2. D-806-005.
3. 1MS-94B-800-8.
4. FSAR Section 11.4.
5. Radiation Monitoring System DBD.
6. SOP-108.
7. SOP-124.
8. V.C. Summer ODCM.

Question 92

92. G2.2.7 001

A bypass authorization request, prepared per SAP-148, "Temporary Bypass, Jumper, and Lifted Lead Control," requires prior PSRC and NSRC review for which ONE of the following conditions?

A. A review indicates that system operability will be affected.
B. A review indicates that 10 CFR 50 Appendix R fire protection criteria are impacted.
C. A review indicates that Seismic or blowout provisions are being diminished.
D. A review indicates that a full safety evaluation is required per 10 CFR 50.59.

Feedback

DISTRACTORS:

A. INCORRECT

B. INCORRECT

C. INCORRECT

D. CORRECT

REFERENCES:
1. SAP-148, "Temporary Bypass, Jumper, and Lifted Lead Control." Attachment 1, page 14 of 20.

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of the process for conducting tests or experiments not described in the safety analysis report.

Question #92 G2.2.7 001

Recommendation:

Delete the question.

Justification:

There is no correct answer for this question. Answers A, B, and C are incorrect per the Distractor analysis. Choice D states that "a review indicates that a full safety evaluation is required per 10 CFR 50.59." The Bypass Authorization Request form (BAR), Attachment 1 of SAP-148 Section IV, requires that the Shift Engineer perform a 10 CFR 50.59 evaluation and consider any affirmative answers from check list questions in Section III . Note 6.3.6 contained in SAP-148 states that "a YES answer to any question on SAP-107, Attachment III, 50.59 Evaluation, requires a Plant Safety Review Committee (PSRC) and Nuclear Safety Review Committee (NSRC) review prior to implementation." However, the 10 CFR 50.59 evaluation could be performed resulting in all "no" answers. In this case, the BAR is processed through the appropriate approval authorities (not including the PSRC and NSRC) and implemented. Section 6.6 of SAP-148 states that, by the end of the next working day, the PSRC will review the BAR and verify that prior NRC approval is not required. Choice D, which was designated as the correct answer is, therefore, also incorrect.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

STATION ADMINISTRATIVE PROCEDURE

SAP-148

TEMPORARY BYPASS, JUMPER, AND LIFTED LEAD CONTROL

REVISION 9

SAFETY RELATED

Original Signed By: T. D. Gatlin **02/03/03**
DISCIPLINE SUPERVISOR DATE

Original Signed By: G. Halnon **02/03/03**
APPROVAL AUTHORITY DATE

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	09/14/05					

INFORMATION USE

Procedure May Be Performed From Memory
User Retains Accountability for Proper Performance.

6.3 The Shift Engineer performs a technical review of the Bypass Authorization Request and completes the applicable parts of Attachment I as follows:

6.3.1 Uses the Bypass Tracking Log (Attachment II) and the Bypass Tag Log (Attachment III), to assign a unique identifying number consisting of the year and a sequential number (e.g. 02-01, 02-02, etc.) to the request.

6.3.2 Addresses the function of the affected system, structure, or component.

6.3.3 Identifies any effects of the temporary bypass, electrical jumper, or lifted lead on system, structure, or component operability.

6.3.4 Completes the Bypass Authorization Checklist.

6.3.5 Provides details under "Precautions or other requirements" of the proposed temporary modification specifying the following as applicable:

A. Wire size and terminations to accommodate expected voltages and currents.

B. Routing and location criteria to consider channel separation, seismic movement, access, interference, etc.

C. Special precautions, administrative controls, or surveillance testing required to support the temporary modification.

NOTE 6.3.6

A YES answer to any question on SAP-107, Attachment III, 50.59 Evaluation, requires a PSRC and NSRC review prior to implementation.

6.3.6 Performs a 10CFR50.59 evaluation per SAP-107. The determination that prior NRC approval is required precludes the use of this procedure in performing a Bypass Authorization Request.

6.3.7 Generates an MWR for the required activity and records the MWR number on Attachment I.

C01→ 6.3.8 Prepares and appends a set of drawings depicting the installation of electrical jumpers or the location of the lifted lead to each of the following:

A. The Bypass Authorization Request.

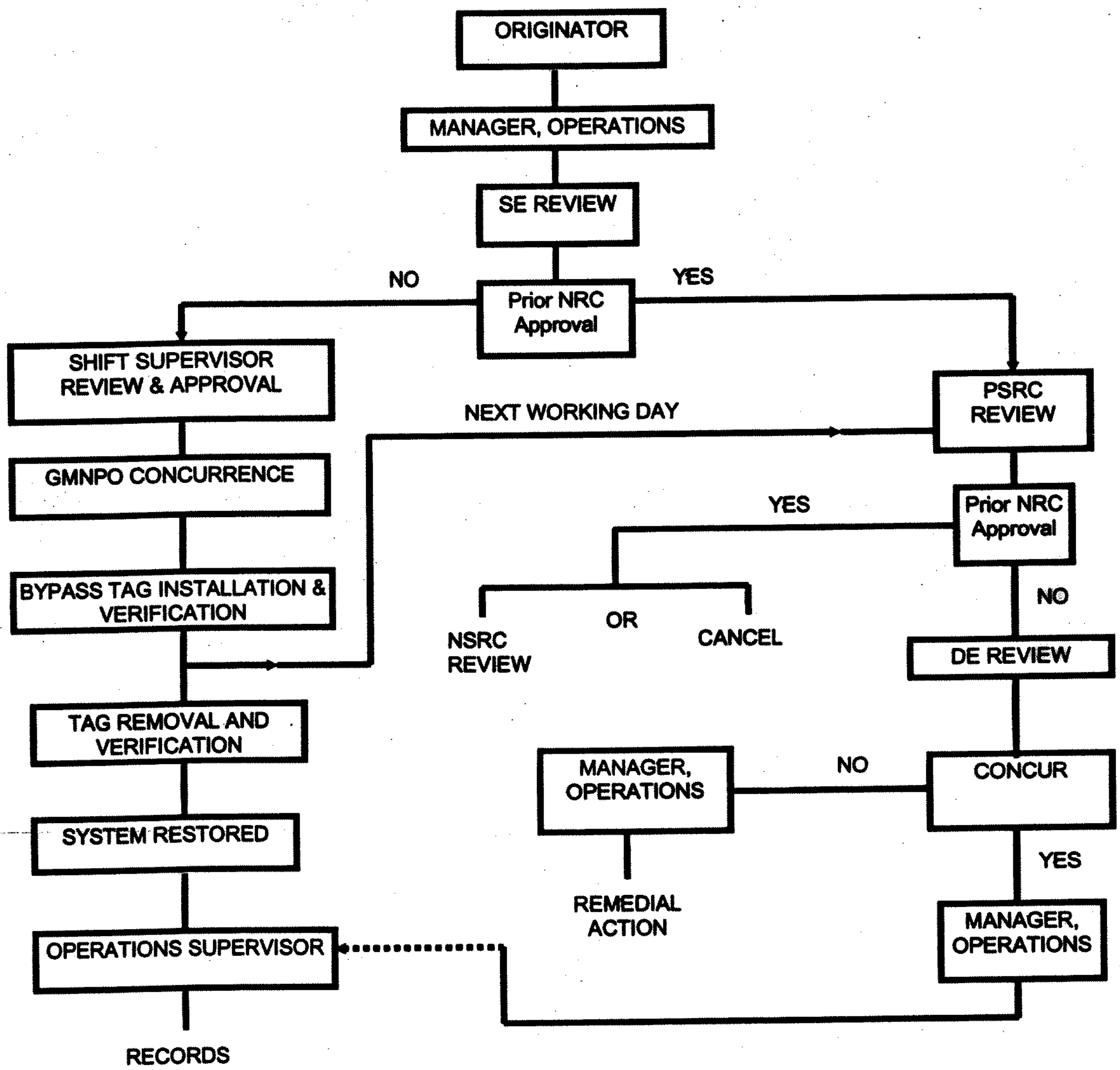
B. The Control Room SAP-148 In-process Drawing Stick.

6.4 The Duty Shift Supervisor provides the final pre-implementation review and approval, using Attachment I, Section V, as follows:

6.4.1 Ensures preparation of the MWR and Bypass Tags (Attachment V).

6.4.2 Denotes as comments any precautions, recommendations, or remedial actions necessary to install the bypass.

7.0 FIGURES


ORIGINATOR
MANAGER, OPERATIONS
SE REVIEW
NO
YES
Prior NRC Approval
SHIFT SUPERVISOR REVIEW & APPROVAL
PSRC REVIEW
NEXT WORKING DAY
GMNPO CONCURRENCE
YES
Prior NRC Approval
BYPASS TAG INSTALLATION & VERIFICATION
NO
OR
NSRC REVIEW
CANCEL
DE REVIEW
TAG REMOVAL AND VERIFICATION
MANAGER, OPERATIONS
NO
CONCUR
SYSTEM RESTORED
YES
REMEDIAL ACTION
MANAGER, OPERATIONS
OPERATIONS SUPERVISOR
RECORDS

BAR # ___________

Bypass Authorization Checklist (each question answered in the affirmative shall be considered in the 10CFR50.59 Evaluation and appropriate compensatory action specified in the Precautions):

a.	Does this adversely affect operability of a system, structure, or component?	☐ Yes	☐ No
b.	Does this alteration adversely affect security criteria?	☐ Yes	☐ No
c.	Does this alteration adversely affect Appendix R fire protection criteria?	☐ Yes	☐ No
d.	Does this alteration adversely affect environmental qualification of the equipment involved?	☐ Yes	☐ No
e.	Does this alteration adversely affect the separation criteria?	☐ Yes	☐ No
f.	Does this alteration adversely affect seismic or blowout criteria?	☐ Yes	☐ No

Precautions or other Requirements:___________________________________

Does this alteration require a retest?___________________________________ CHG A

____
Shift Engineer

IV. 10 CFR 50.59 Review

Perform a 10 CFR 50.59 review per SAP-107.

If the answer to any of the questions on Attachment III of SAP-107 is YES, PSRC and NSRC review shall be required prior to implementation.

If prior NRC approval is required, the modification shall not be completed under this procedure.

____
Shift Engineer

MWR # ______________

CER# ______________

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
NUCLEAR OPERATIONS

NUCLEAR OPERATIONS
COPY NO. ____________

STATION ADMINISTRATIVE PROCEDURE

SAP-107

10CFR50.59 REVIEW PROCESS

REVISION 5

Original signed by Ron Clary
DISCIPLINE SUPERVISOR

07-12-2004
DATE

Original signed by Jeffrey Archie
APPROVAL AUTHORITY

07-15-2004
DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE

INFORMATION USE
Procedure May Be Performed From Memory.
User Retains Accountability For Proper Performance.

50.59 EVALUATION

50.59 Evaluation Report Log Number: ________________ Evaluation Revision Number: ________ Page 1 of ___

Parent Activity/Document Number: ____________________

Title: ____

<u>Summary:</u> (To be used in preparation of the periodic report submitted pursuant to 10CFR50.59(d)(2).)

Activity Description:

Continued see attached ☐

Summary of Evaluation:

Continued see attached ☐

Based upon the results of this evaluation:

☐ Implement the activity per plant procedures without obtaining a License Amendment.

☐ Request and receive a License Amendment prior to implementation.

50.59 Evaluator: Name: ________________ Signature: ________________ Date: ___/___/___

50.59 Independent Reviewer: Name: ________________ Signature: ________________ Date: ___/___/___

50.59 Approver: Name: ________________ Signature: ________________ Date: ___/___/___

PSRC Meeting Number: ________ Date: ___/___/___

(Original to the Parent Activity Document package. Copy sent to Nuclear Licensing m/c 830 ________/________)
Initial Date

50.59 EVALUATION

50.59 Evaluation Report Log Number: ____________________ Evaluation Revision Number: ________ Page __ of __

NOTES: A separate written response providing the basis for the answer to each question below must accompany this form. The 10CFR50.59 Resource Manual (RM) should be used to determine the content of each response (See Section 6.2 for additional guidance). Identify the references used to perform the evaluation, either in a single list or within the written responses.

If the answer to any of the 50.59 Evaluation questions is Yes, then request and receive NRC approval prior to implementation of the activity.

Throughout this evaluation, FSAR/FPER refers to the current FSAR as updated per 10CFR50.71(e), approved changes to the FSAR/FPER which have not yet been submitted to the NRC by amendment, and documents incorporated into the FSAR/FPER by reference.

EFFECT ON ACCIDENTS AND MALFUNCTIONS PRECIOUSLY EVALUATED IN THE FSAR/FPER

1. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR/FPER? (See Section 6.2.1 of the RM)

☐ Yes ☐ No

2. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR/FPER? (See Section 6.2.2 of the RM)

☐ Yes ☐ No

3. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR/FPER? (See Section 6.2.3 of the RM)

☐ Yes ☐ No

4. Does the proposed activity result in more than a minimal increase in the consequences of an malfunction of an SSC important to safety previously evaluated in the FSAR/FPER? (See Section 6.2.4 of the RM)

☐ Yes ☐ No

POTENTIAL FOR CREATION OF A NEW TYPE OF EVENT NOT PRECIOUSLY EVALUATED IN THE FSAR/FPER

5. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the FSAR/FPER? (See Section 6.2.5 of the RM)

☐ Yes ☐ No

6. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR/FPER? (See Section 6.2.6 of the RM)

☐ Yes ☐ No

IMPACT ON FISSION PRODUCT BARRIERS AS DESCRIBED IN THE FSAR/FPER

7. Does the proposed activity result in a design basis limit for a fission product barrier as described in the FSAR/FPER being exceeded or altered? (See Section 6.2.7 of the RM)

☐ Yes ☐ No

IMPACT ON EVALUATION METHODOLOGIES DESCRIBED IN THE FSAR/FPER

8. Does the proposed activity result in a departure from a method of evaluation described in the FSAR/FPER used in establishing the design bases or in the safety analyses? (See Section 6.2.8 of the RM)

☐ Yes ☐ No

Question 93

93. G2.3.2 002

Which ONE of the following is correct per HPP- 709, Sampling and Release of Radioactive Gaseous Effluents:

A. Discharges from the Waste Gas Decay Tank or other high activity gaseous releases should be avoided when the wind is from the East-Southeast. This will prevent the released activity from being drawn into the Auxiliary Building ventilation.
B. Discharges from the Waste Gas Decay Tank or other high activity gaseous releases should be avoided when the wind is from the West-Southwest. This will prevent the released activity from being drawn into the Auxiliary Building ventilation.
C. Discharges from the Waste Gas Decay Tank or other high activity gaseous releases should be avoided when the wind is from the East-Southeast. This will prevent the released activity from being drawn into the Control Building ventilation.
D. Discharges from the Waste Gas Decay Tank or other high activity gaseous releases should be avoided when the wind is from the West-Southwest. This will prevent the released activity from being drawn into the Control Building ventilation.

Feedback

Distractor Analysis:

A. Correct: Discharges from the Waste Gas Decay Tank or other high activity gaseous releases should be avoided when the wind is from the **East-Southeast**. This will prevent the released activity from being drawn into the **Auxiliary** Building ventilation. Per HPP-709 NOTE 5.1.H

B, C, D Incorrect Reference: HPP- 709, Sampling and Release of Radioactive Gaseous Effluents, page 10

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of facility ALARA program.

Question #93 (G2.3.2 002)

Recommendation:

Delete the question.

Justification:

The information needed to answer this question is contained in Note 5.1.1.H on page 10 of 19 of Health Physics Procedure, HPP-709. This is the eighth major step (with numerous substeps) in the Sampling, Analysis, and Calculations section of the HPP. The application of procedures at this level is beyond the expectations for operators to recall from memory. Also, the operations procedure which is used by operations personnel to perform gaseous releases (SOP-119) is a Continuous Use Procedure.*

In addition, the question does not match the K/A, as it does not test knowledge of the facility ALARA program (G2.3.2). Rather, the question matches K/A for knowledge of the process for performing planned Gaseous Radioactive releases (G2.3.8).

This question is also inappropriate as the choices are a series of True/False questions which can be answered without having to refer to the stem of the question. The stem can be removed and replaced with "Which of the following is true?" Use of this type of question is specifically prohibited in NUREG-1021, Section D.2.b of ES-401, and Form ES-401-9.

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

HEALTH PHYSICS PROCEDURE

HPP- 709

SAMPLING AND RELEASE OF RADIOACTIVE GASEOUS EFFLUENTS

REVISION 10

SAFETY RELATED

Original Signed By M. Coleman — 07/02/02
DISCIPLINE SUPERVISOR — DATE

Original Signed By L.A. Blue — 07/08/02
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	05/27/03					
B	P	09/23/03					
C	P	11/11/03					

INFORMATION USE

Procedure May Be Performed From Memory.
User Retains Accountability For Proper Performance.

TABLE OF CONTENTS

3. If Waste Gas Decay Tank G or H radioactive gas concentration is equal to or less than 1.0E-4 uCi/mL then the warning and high radiation setpoints for RM-A10 should be set at 1000 cpm as long as this does not exceed the ODCM calculated setpoint value. If the concentration is greater than 1.0E-4 uCi/mL contact the Supervising Health Physicist, Count Room / designee for RM-A10 setpoint evaluation.

4. Complete Part I of Attachment I of the GWRP and review for compliance with release requirements. Limits are tabulated in Attachment III.

G. If the release cannot be permitted, notify the Shift Supervisor (or his designee) of the problem areas and possible courses of action.

NOTE: 5.1.1 H

Discharges from the Waste Gas Decay Tank or other high activity gaseous releases should be avoided when the wind is from the East-Southeast. This will prevent the released activity from being drawn into the Auxiliary Building ventilation.

H. If the release is acceptable as proposed, sign and date the Count Room section of Attachment I. Store the pre-release data on computer files as applicable.

I. For an approved release:

1. Copy the GWRP and attach the corresponding computer analysis printout or manual calculation sheet(s). Place in the active GWRP file.

2. Forward the approved GWRP to the Control Room, where Operations will make the release.

3. The Waste Gas Decay Tank permit is valid for twenty-four (24) hours (without the release commencing) from approval in Part I of Attachment I. A new WGDT sample is required within the 24-hour period if additions (known or suspected) have been made to the tank to verify that the current permit is valid. If the sample results indicate that the current permit is not valid, a new permit shall be generated. If the release is unexpectedly terminated, refer to step 5.3.

4. After completing the release, update the computer files with the actual release data or if the current authorization expires,

GWRP No.

GASEOUS WASTE RELEASE PERMIT (GWRP)

☐ G TANK ☐ H TANK

I. RELEASE AUTHORIZATION (COUNT ROOM)

Date/time Sampled:____________________ Total Noble Gas, (µCi/cc):______________

Maximum WGDT Release Rate, (cfm):________ Initial Tank Pressure, (psig):______________

	Background (cpm)	Alarm Set point (cpm)
RM-A3 Gas Channel		
RM-A10 Gas Channel		

Additional Requirements:____

____

____

Count Room:________________________________ Date/Time:__________________

II. ACTUAL RELEASE DATA (Operations)

Release Approved, SS/CRS:__________________________ Date/Time:__________________

Meterology Acceptable ☐ or, Unacceptable ☐ Wind Speed, (mph):________ ∆T:________

	RM-A3 (CPM)	RM-A-10 (CPM)	INITIALS
Alarm Set Point (cpm)			
Reading @ Release Start (cpm)			
Reading @ 10 mins into Release (cpm)			
Reading @ End of Release (cpm)			
Reading After Purge (cpm)			
Alarm Set Point returned to 2 x ni			

	Start	Finish	Net
Release Date/Time			hours
Flow, (cfm)			
Pressure, (psig)			psig

COMMENTS ____

____

Release Conducted by:________________________ Date/Time:________________________

Operations Review:__________________________ Date/Time:________________________

Volume Released, (ft3):________________________

Updated by:________________________________ Date/Time:________________________
(Count Room)

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO._________

SYSTEM OPERATING PROCEDURE

SOP-119

WASTE GAS PROCESSING

REVISION 16

QUALITY RELATED

Original Signed by D. A. Baker
DISCIPLINE SUPERVISOR

12/12/02
DATE

Original Signed by D. R. Goldston
APPROVAL AUTHORITY

12/12/02
DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	06/17/03		E	P	07/21/05	
B	P	11/06/03					
C	P	12/03/03					
D	P	06/03/04					

MULTIPLE USAGE LEVELS

This Procedure Contains Multiple Usage Levels.
Refer to the Procedure Contents for Clarification.

CONTINUOUS USE

Continuous Use of Procedure Required.
Read Each Step Prior to Performing.

GASEOUS WASTE RELEASE WORKSHEET - CONTROL ROOM

GWRP #:	WGDT #:	DATE:

CAUTION 1

During the release of gases, the conditions specified in the Gaseous Waste Release Permit (GWRP) must be adhered to (flow rate, radiation monitor setpoints, meteorological conditions, etc.). CHG C

INITIALS

1. Gaseous Waste Release Permit (GWRP) is returned from Health Physics with PART I completed and approved. ______ CHG C

2. Ensure the following:

   a. Gas Decay Tank G(7) or H(8) is not in service. ______

   b. The AB Ventilation System is operating per SOP-502 with at least one AB Charcoal Exhaust Fan verified running (XFN-19A or XFN-19B). ______

   c. Wind direction is not from the East Southeast per HPP-709 to prevent activity from being drawn into the AB Ventilation System. ______

INITIALS

3. Radiation Monitors:

NOTE 3.a

1) The channel check should include a comparison of any local indication.

2) With the number of inoperable meteorological monitoring channels less than that which is required by Technical Specification 3.3.3.4, all gaseous releases must be stopped until the inoperable channel is restored.

a. Perform a channel and source check of RM-A3 Gas Channel and RM-A10: ________

RM-A3 Gas Channel	Channel Check: SAT/UNSAT Source Check: SAT/UNSAT	COMMENTS:
RM-A10	Channel Check: SAT/UNSAT Source Check: SAT/UNSAT	COMMENTS:

NOTE 3.b

If RM-A10 or RM-A3 is not operable, refer to offsite Dose Calculation Manual 1.2.1.1.

b. Verify RM-A10 or RM-A3 is operable and the Interlock Switch(s) on the Radiation Monitoring Panel is (are) in the NORMAL position: ________

RM-A3	NORMAL/BYPASS	COMMENTS:
RM-A10	NORMAL/BYPASS	COMMENTS:

INITIALS

c. Adjust Radiation Monitor setpoints per the Gaseous Waste Release Permit (GWRP). ________ CHG C

4. Verify meteorological instrumentation is operable and meteorological conditions are satisfactory for the release per TABLE A (next page). ________

5. Mark the chart recorders for RM-A3G and RM-A10 in the Control Room with the following:

   a. Tank Name/Number. ________

   b. Date/Time. ________

6. Direct the building operator to commence Attachment VB. ________

7. At least once per hour, monitor Control Room meteorological indicators to verify conditions specified in TABLE A (next page) are acceptable for continued release. ________

8. When notified by the local operator that the required volume of gas and nitrogen has been released, verify Radiation Monitors return to normal background. ________

9. If Radiation Monitors do not return to normal, notify the Count Room. ________

10. Reset the RM-A10 alarm setting as specified in the Gaseous Waste Relase Permit (GWRP), (RM-A3 alarm setting to remain at 300 cpm). ________ CHG C

11. Shift Supervisor review package and attach worksheet and applicable attachments. ________

Release conducted by: ____________________ Date: __________

Shift Supervisor review: ____________________ Date: __________

REMARKS: ____

TABLE A

ACCEPTABLE METEOROLOGY FOR PLANNED WGDT RELEASES

DIFFERENTIAL TEMPERATURE (ΔT) °F (NOTE 1)		STABILITY CLASS	MINIMUM WIND SPEED (mph) (NOTE 2)
61m - 10m	40m - 10m		
$\Delta T \leq -1.74$	$\Delta T \leq -1.03$	A	*
$-1.74 < \Delta T \leq -1.56$	$-1.03 < \Delta T \leq -0.92$	B	*
$-1.56 < \Delta T \leq -1.38$	$-0.92 < \Delta T \leq -0.81$	C	1.6
$-1.38 < \Delta T \leq -0.46$	$-0.81 < \Delta T \leq -0.27$	D	4.1
$-0.46 < \Delta T \leq 1.38$	$-0.27 < \Delta T \leq 0.81$	E	6.6
$1.38 < \Delta T \leq 3.67$	$0.81 < \Delta T \leq 2.16$	F	14.0
$3.67 < \Delta T$	$2.16 < \Delta T$	G	18.9

NOTES:

1. The ΔT values for 61m - 10m are considered as primary indicators for determination of stability class. The 40m - 10m ΔT values are used only when 61m - 10m values are not available. All ΔT values are listed in °F and are based on values in USNRC Regulatory Guide 1.23.

2. The 10m wind speed is considered the primary indication for wind speed. The 61m wind speed indication should only be used if the 10m indicator is not available.

* No wind is required for planned releases.

Question 94

94. G2.4.33 002

Which ONE of the following individual's approval is required to extend the time that an invalid nuisance annunciator is removed from service past 96 hours?

A. Duty Shift Engineeer

B. Duty Shift Supervisor

C. Manager, Operations

D. General Manager, Nuclear Plant Operations

Feedback

DISTRACTORS:

A.
B.
C. Correct per OAP-100.5, Section 14.0
D.

REFERENCES:

1.

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of the process used to track inoperable alarms.

Question #94 (G2.4.33 002)

Recommendation:

Delete the question.

Justification:

According to Operations Administrative Procedure (OAP), OAP-100.5, Note 14.0, the Operations Manager approval is required for troubleshooting between 48 and 96 hours. Following the 96-hour point, the Operations Manager would direct the method of continuous inactivation using SAP-148, "Temporary Bypass, Jumper, and Lifted Lead Control," or SAP-300, "Conduct of Maintenance." Therefore, this OAP controls only until the 96-hour point, and then another program would be implemented.

Per SAP 148, Section 5.2.1, the Operations Manager is responsible for approving the use of any Bypass Authorization Requests (BAR) and assigning a Shift Engineer to develop the proper documentation. Section 5.6.1 defines that the General Manager Nuclear Operations is responsible for providing concurrence with the implementation of all BARs prior to implementation. Section 6.4 states that the Duty Shift Supervisor provides the final pre-implementation review and approval by obtaining concurrence of the General Manager, Nuclear Plant Operations. The nuisance annunciator removed from service by a SAP-148, Temporary Bypass, Jumper, or Lifted Lead, in this question, thus, requires the approval of the Duty Shift Supervisor, the Operations Manager, and the General Manager, Nuclear Plant Operations prior to implementation. Therefore, all three approvals are necessary before the 96-hour period can be extended. Since all three positions must approve the BAR, there is no correct answer because all three positions are not included in a single choice.

The information needed to answer this question is contained well within the body of an OAP and a Station Administrative Procedure (SAP). The application of procedures at this level is beyond the expectations for operators to recall from memory.*

This item also does not meet the requirements of 10 CFR 55.43(b) item 5, nor any of the other 10 CFR 55.43(b) items. To make that K/A topic meet the requirements for an SRO item, it would have to be written in a significantly different manner.

*This question requires an application of procedures which is beyond expectations for operators to recall from memory. As such, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.________

OPERATIONS ADMINISTRATIVE PROCEDURE

OAP-100.5

GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT

REVISION 1

SAFETY RELATED

Original Signed By: D. A. Baker — 09/02/04
DISCIPLINE SUPERVISOR — DATE

Original Signed By: R. F. Ray — 09/23/04
APPROVAL AUTHORITY — DATE

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	02/02/05					
B	P	03/21/05					
C	P	04/14/05					
D	P	04/19/05					

INFORMATION USE

Procedure May Be Performed From Memory.
User Retains Accountability For Proper Performance.

2) Determined that the equipment requires alignment different than specified by procedures.

3) Determined that the alignment is due to conditions that are unusual and therefore, do not require a formal procedure change.

4) Determined that personnel safety or equipment protection is not involved, thus ensuring Danger Tags are not required.

5) Determined that the misalignment does not impact Tech Spec operability requirements.

13.5 The Shift Supervisor or Control Room Supervisor shall determine the required position for component restoration per the applicable SOP lineup.

13.6 An entry shall be made in the Station Log Book showing which component was added to the Misalignment Status Log Book.

NO1 → 13.7 The Equipment Misalignment Status Log will be audited monthly, per Attachment IA, for accuracy. Items exceeding 30 days shall be evaluated for continued misalignment. If continued misalignment is required, a Caution Tagout shall be issued and the item(s) removed from the Equipment Misalignment Status Log. When items are Caution Tagged, generate a CER with an action to the TAG REQUESTER to perform a 10CFR50.59 screening prior to the component being misaligned more than 90 days.

13.8 The Equipment Misalignment Status Log will be updated monthly after the performance of 30 day evaluations.

13.9 Completed Equipment Misalignment Status Logs and 30 day evaluations will be maintained in the Operations satellite retention file for the current refueling cycle.

<u>NOTE 14.0</u>

Components may be removed from service for up to 96 hours for situations where further troubleshooting is required. Operations Management approval is required for the extension from 48 to 96 hours. If an MWR already exists against the annunciator for equipment repair, an additional MWR should be written to disable the annunciator.

14.0 CONTROL OF ANNUNCIATORS

14.1 Removal of Nuisance Annunciators from service:

a. Refer to the applicable Annunciator Response Procedure to ensure no actions are required.

b. Initiate an MWR for I&C to disable and troubleshoot the affected annunciator. If an MWR already exists against the annunciator for equipment repair, an additional MWR must be written to disable the nuisance annunciator. CHG C

c. Initiate an Action R&R against the affected component and assign a 48 hour time limit.

d. Evaluate and describe any alternate monitoring method for the affected component(s), and record results on OAP-106.1, Attachment XXII.

e. Authorize I&C to work the MWR and record actions taken on Attachment V. CHG C

f. Attach Attachment V to the R&R form.

14.2 Invalid annunciator removed for greater than 96 hours:

a. Implement one of the following at the Operations Managers discretion to document the long-term removal:

1) SAP-148, Temporary Bypass, Jumper, and Lifted Lead Control.

2) SAP-300, Conduct Of Maintenance.

b. If no Tech Spec applies to the affected component, clear the associated R&R after recording the relevant SAP-148 (or MWR) documentation in the R&R remarks section.

c. Upon Operations Management approval, direct I&C to jumper or lift terminal leads, as appropriate, to place the affected annunciator window in the dark condition. Place a label stating the controlling document and ALARM DISABLED across the window.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

STATION ADMINISTRATIVE PROCEDURE

SAP-148

TEMPORARY BYPASS, JUMPER, AND LIFTED LEAD CONTROL

REVISION 9

SAFETY RELATED

Original Signed By: T. D. Gatlin — **02/03/03**
DISCIPLINE SUPERVISOR — DATE

Original Signed By: G. Halnon — **02/03/03**
APPROVAL AUTHORITY — DATE

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	09/14/05					

INFORMATION USE

Procedure May Be Performed From Memory
User Retains Accountability for Proper Performance.

4.4 Lifted Lead (LL) - A temporarily disconnected wire previously connected to a terminal. Opening of sliding link connections or placing wires on a standoff also constitutes a lifted lead.

5.0 RESPONSIBILITIES

5.1 The originator of a Bypass Authorization Request must ensure that the information provided in Section I of Attachment I is complete and as detailed as possible.

5.2 Operations Personnel are responsible for the following:

5.2.1 The Manager, Operations receives and approves all Bypass Authorization Requests, determines applicability, and assigns a Shift Engineer to review and complete the request.

5.2.2 The Operations Supervisor or his designee performs 90-day audits of the Bypass Tag Log and assesses continued applicability of the active Bypass Authorization Requests per Attachment IV.

5.2.3 The Duty Shift Supervisor is responsible for the following:

A. Reviewing and approving the implementation and restoration of all Bypass Authorization Requests.

B. Ensuring all precautions and special requirements of the Bypass Authorization Requests are satisfied.

5.2.4 Shift Engineers (SE) are responsible for the following:

A. Performing a technical review and a 10 CFR 50.59 review per SAP-107 prior to implementation of the change.

B. Tracking the status of all active Bypass Authorization Requests (BARs), in accordance with Step 6.14, insuring all applicable time frames are met.

C. The Duty Shift Engineer coordinates installation, removal, and independent verifications of temporary bypasses, electrical jumpers, and lifted leads.

5.2.5 Operations personnel control temporary bypasses, electrical jumpers, lifted leads, and the Bypass Tag Log.

5.3 Site Engineering personnel are responsible for the following:

5.3.1 Design Engineering (DE) personnel concur with or disapprove all implemented Bypass Authorization Requests.

5.3.2 Design Engineering reviews all BARs that will exceed 90 day limit.

5.4 Plant Support Engineering Computer Group personnel inhibit, remove from scan, or otherwise temporary bypass inputs into the plant computer (IPCS) or any process computer where such action is required by this procedure.

5.5 Electrical Maintenance personnel and I & C personnel install and remove electrical jumpers and lifted leads in accordance with this procedure and SAP-300.

5.6 The General Manager, Nuclear Plant Operations is responsible for the following:

5.6.1 Providing concurrence with the implementation of all Bypass Authorization Requests prior to implementation.

5.6.2 Approval of all BARs that will exceed 90 day limit.

5.7 The various committees associated with the Plant are responsible for the following:

5.7.1 The Plant Safety Review Committee (PSRC) is responsible for reviewing all approved Bypass Authorization Requests the next working day.

5.7.2 The Plant Safety Review Committee (PSRC) and Nuclear Safety Review Committee (NSRC) are responsible for reviewing all Bypass Authorization Requests which involve prior NRC approval.

6.0 PROCEDURE

6.1 The originator requiring a temporary bypass, electrical jumper, or lifted lead:

6.1.1 Notifies their immediate supervisor in conjunction with the initiation of Attachment I, Bypass Authorization Request.

6.1.2 Completes Section I of Attachment I.

6.1.3 Forwards Attachment I with Section I completed to the Manager, Operations.

6.2 The Manager, Operations or his designated alternate, performs one of the following:

6.2.1 Approves the request for further consideration and assigns a Shift Engineer to perform a technical review.

6.2.2 Disapproves the request and performs the following:

A. Provides a brief explanation for the disapproval.

B. Forwards the disapproved request to the originator.

6.3 The Shift Engineer performs a technical review of the Bypass Authorization Request and completes the applicable parts of Attachment I as follows:

6.3.1 Uses the Bypass Tracking Log (Attachment II) and the Bypass Tag Log (Attachment III), to assign a unique identifying number consisting of the year and a sequential number (e.g. 02-01, 02-02, etc.) to the request.

6.3.2 Addresses the function of the affected system, structure, or component.

6.3.3 Identifies any effects of the temporary bypass, electrical jumper, or lifted lead on system, structure, or component operability.

6.3.4 Completes the Bypass Authorization Checklist.

6.3.5 Provides details under "Precautions or other requirements" of the proposed temporary modification specifying the following as applicable:

A. Wire size and terminations to accommodate expected voltages and currents.

B. Routing and location criteria to consider channel separation, seismic movement, access, interference, etc.

C. Special precautions, administrative controls, or surveillance testing required to support the temporary modification.

<u>NOTE 6.3.6</u>

A YES answer to any question on SAP-107, Attachment III, 50.59 Evaluation, requires a PSRC and NSRC review prior to implementation.

6.3.6 Performs a 10CFR50.59 evaluation per SAP-107. The determination that prior NRC approval is required precludes the use of this procedure in performing a Bypass Authorization Request.

6.3.7 Generates an MWR for the required activity and records the MWR number on Attachment I.

C01→ 6.3.8 Prepares and appends a set of drawings depicting the installation of electrical jumpers or the location of the lifted lead to each of the following:

A. The Bypass Authorization Request.

B. The Control Room SAP-148 In-process Drawing Stick.

6.4 The Duty Shift Supervisor provides the final pre-implementation review and approval, using Attachment I, Section V, as follows:

6.4.1 Ensures preparation of the MWR and Bypass Tags (Attachment V).

6.4.2 Denotes as comments any precautions, recommendations, or remedial actions necessary to install the bypass.

6.4.3 Notifies, as applicable, any organization responsible for compliance with precautions, recommendations, or remedial actions.

6.4.4 Documents Step 6.4.3 compliance as deemed necessary:

A. Verbal - records the time/date, and the name/title of the person notified.

B. Restricted procedure change - records procedure name and number.

C. Special compensatory actions (logs, computer trends, other alternative actions).

D. R&R entries.

E. MWRs.

6.4.5 Obtains the concurrence of the General Manager, Nuclear Plant Operations or his designated alternate.

6.4.6 If any question on the 10CFR50.59 Evaluation was answered YES, ensures PSRC and NSRC review prior to implementation.

6.4.7 Contacts the duty Shift Engineer to coordinate the Bypass Authorization Request implementation.

6.5 The Duty Shift Engineer:

6.5.1 Conducts a pre-implementation briefing with applicable Operations personnel and plant personnel performing the actions of the request.

NOTE 6.5.2

All jumpers shall be tagged in accordance with this procedure prior to installation. Jumpers which are installed between cabinets or of a length such that both ends are not visible when installed, shall have a bypass tag attached at each end. Bypass Tags shall be placed on each lifted lead or sliding link that is opened. Any bypass affecting plant computer inputs shall also have a bypass tag denoting affected changes attached to the appropriate console.

6.5.2 Provides independent verification that the bypass is properly installed and the bypass tags properly hung.

6.5.3 Initiates a CER and assigns an action for Design Engineering, which will:

A. Track completion of Design Engineering review of the Bypass Authorization Request within 14 days.

B. Track the 90 day duration and ensure timely removal of the bypass.

Question 96

96. W/E02EG2.4.6 001

Plant conditions are as follows:

- A reactor trip and SI have occurred due to a steam break.
- ALL Main Steam Isolation Valves initially failed to close.
- EOP-3.1, Uncontrolled Depressurization of All Steam Generators, is in progress at Step 17, Establish Normal Charging.
- PZR level is 58%.
- EFW flowrate is 50 gpm to each Steam Generator due to required operator action.
- All Steam Generator Narrow Range levels are 4%.
- Reactor Building pressure has remained below 1 psig.
- RCS pressure is 1750 psig and going UP.
- Core Exit TCs are 435°F and going DOWN.

The "C" Main Steam Isolation Valve closed 30 seconds ago and "C" Steam Generator pressure has changed from 80 to 130 psig.

Which ONE of the following correctly describes the actions the crew should take?

A. Must remain in EOP-3.1 until the Critical Safety Function Status Trees direct entering an orange or red path Emergency Operating Procedure.
B. IMMEDIATELY transition to EOP-3.0, Faulted Steam Generator Isolation, Step 1.
C. Complete EOP-3.1 through Step 20, verify SI Flow is NOT required, and then transition to EOP-3.0, Faulted Steam Generator Isolation, Step 1.
D. Complete ALL steps of EOP-3.1 and then transition to EOP-1.2, Safety Injection Termination, Step 1.

Feedback

DISTRACTORS:
A. Incorrect. The C SG pressure has increased. Per EOP-3.1 Reference Page item 2, Secondary Integrity Transition Criteria, the crew should go to EOP-3.0, Faulted Steam Generator Isolation, Step 1, after completing EOP-3.0 SI Termination steps 15 through 20. Plausible if applicant does not recognize secondary integrity transition criteria.
B. Incorrect. Per EOP-3.1, Reference Page item 2, the crew should go to EOP-3.0 if any SG pressure increases at any time EXCEPT while performing SI Termination in steps 15 through 20. Plausilbe if applicant does not recognize step number or step description as an SI Termination step or does not remember an exception to Secondary Integrigy Transition Critierion.
C. Correct. Per EOP-3.1, Reference Page, item 2, Secondary Integrity Transition Criterion.
D. Incorrect. Per EOP-3.1 Reference Page, item 2, the crew should transition to EOP-3.0 after completing SI Termination in Steps 15 through 20. Plausible because the last step of EOP-3.0, Faulted Steam Generator Isolation, directs a transition to EOP-1.2.

REFERENCES:
1 EOP-3.1, Uncontrolled Depressurization of All Steam Generators

2 EOP-3.1LP, Uncontrolled Depressurization of All Steam Generators Lesson Plan

K/A CATALOGUE QUESTION DESCRIPTION:
- W/E02 SI Termination
- Knowledge symptom based EOP mitigation strategies (3.1/4.0).

Question #96 (W/E02EG2.4.6 001)

Recommendation:

Delete the question.

Justification:

The information needed to answer this question must be recalled from the Reference Page of EOP-3.1, "Uncontrolled Depressurization of All Steam Generators." To answer the question, the examinee must determine that the given information of Step 17, Establish Normal Charging, is part of the Safety Injection (SI) termination sequence and apply the Reference Page item. The Reference Page item, Secondary Integrity Transition Criterion, directs transition out of EOP-3.1 when one SG is isolated from the other two, except when performing Steps 15 through 20. The operationally valid point of this question would be recognizing that we have recovered one steam generator (due to its main steam isolation valve being closed) and can now use it for plant recovery per EOP-3.0. It is not operationally valid to recall that Step 17 is part of the SI termination sequence and relate Step 17 to the Reference Page item. Additionally, note that there are no Immediate Operator Actions in EOP-3.1.

This question does not match the K/A item (EO2 EPE) in that it does not test the mitigation strategies of SI termination (EOP-1.2). Rather, it tests the knowledge of transition criteria (selection of procedures) while mitigating an uncontrolled depressurization of all three SGs (EOP-3.1). The concept that makes the applicant determine that Step 17 happens to be in the SI Termination sequence of that procedure is an inappropriate application of the K/A match. The actual K/A being tested is E12 EA2.1

This question requires an application of procedures which is beyond expectations for operators to recall from memory.* As such, this question should be deleted.

* See discussion in Section II.A.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO. __________

EMERGENCY OPERATING PROCEDURE

EOP-3.1

UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS

REVISION 13

SAFETY RELATED

Original signed by Baker — 11/14/03
DISCIPLINE SUPERVISOR — DATE

Original signed by Gatlin — 11/15/03
APPROVAL AUTHORITY — DATE

1 SI REINITIATION CRITERIA

IF either of the following conditions occurs, THEN start Charging Pumps and operate valves as necessary:

- RCS subcooling on TI-499A(B), A(B) TEMP °F, is LESS THAN 30°F.

OR

- PZR level can NOT be maintained GREATER THAN 18% [38%].

2 SECONDARY INTEGRITY TRANSITION CRITERION

IF any SG pressure increases at any time, except while performing SI Termination in **Steps** 15 through 20, THEN **GO TO EOP-3.0, FAULTED STEAM GENERATOR ISOLATION, Step 1.**

3 TUBE RUPTURE TRANSITION CRITERIA

IF any SG level increases in an uncontrolled manner OR if any SG has abnormal radiation, THEN start Charging Pumps and operate valves as necessary, and **GO TO EOP-4.0, STEAM GENERATOR TUBE RUPTURE, Step 1.**

4 COLD LEG RECIRCULATION TRANSITION CRITERION

IF RWST level decreases to LESS THAN 18%, THEN **GO TO EOP-2.2, TRANSFER TO COLD LEG RECIRCULATION, Step 1.**

5 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
*14 **Check if SI flow should be reduced:**	
a. Verify RCS subcooling on TI-499A(B), A(B) TEMP °F, is GREATER THAN 30°F. ☐	a. **RETURN TO Step 2.** Observe the **CAUTION** and **NOTE** prior to **Step 2.** ☐
b. Verify RCS pressure is stable OR increasing. ☐	b. **RETURN TO Step 2.** Observe the **CAUTION** and **NOTE** prior to **Step 2.** ☐

NOTE - Step 14.c

If PZR level is LESS THAN 18% [38%], the PZR should refill from SI flow after pressure is stabilized.

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
c. Verify PZR level is GREATER THAN 18% [38%]. ☐	c. Try to stabilize RCS pressure with Normal PZR Spray. ☐ **RETURN TO Step 14.a.** ☐
15 **Stop all but one Charging Pump and place in Standby.** ☐	
16 **Verify RCS pressure is stable OR increasing.** ☐	16 **RETURN TO Step 2.** Observe the **CAUTION** and **NOTE** prior to **Step 2.** ☐
17 **Establish Normal Charging:**	
a. Close FCV-122, CHG FLOW. ☐	
b. Open both MVG-8107 and MVG-8108, CHG LINE ISOL. ☐	
c. Adjust FCV-122, CHG FLOW, to obtain 60 gpm Charging flow. ☐	
d. Close both MVG-8801A(B), HI HEAD TO COLD LEG INJ. ☐	

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
18 **Control FCV-122, CHG FLOW, to maintain PZR level between 30% [50%] and 68% [54%].** ☐	18 <u>IF</u> PZR level continues to decrease, <u>THEN</u>: a) Close FCV-122, CHG FLOW. ☐ b) Open MVG-8801A(B), HI HEAD TO COLD LEG INJ. ☐ c) **RETURN TO Step 2.** Observe the **CAUTION** and **NOTE** prior to **Step 2.** ☐
19 **Stop <u>both</u> RHR Pumps and place in Standby.** ☐	
20 **Verify SI flow is <u>NOT</u> required:**	
a. RCS subcooling on TI-499A(B), A(B) TEMP °F, is GREATER THAN 30°F. ☐	a. Perform the following: 1) Start Charging Pumps. ☐ 2) Close FCV-122, CHG FLOW. ☐ 3) Open MVG-8801A(B), HI HEAD TO COLD LEG INJ. ☐ 4) **RETURN TO Step 2.** Observe the **CAUTION** and **NOTE** prior to **Step 2.** ☐
b. PZR level is GREATER THAN 18% [38%]. ☐	b. Adjust Charging flow to increase PZR level. ☐ <u>IF</u> PZR level can <u>NOT</u> be maintained, <u>THEN</u>: 1) Start Charging Pumps. ☐ 2) Close FCV-122, CHG FLOW. ☐ 3) Open MVG-8801A(B), HI HEAD TO COLD LEG INJ. ☐ 4) **RETURN TO Step 2.** Observe the **CAUTION** and **NOTE** prior to **Step 2.** ☐
21 **Verify RCS T_{hot} is stable <u>OR</u> decreasing.** ☐	21 Adjust EFW flow or operate Steam Dumps as necessary to stabilize RCS T_{hot}. ☐

Facility: Date of Exam:

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	1	4		N/A		3	2	N/A		1	18	4	5	6
	2	1	1	3				1	1			3	9	2	0	4
	Tier Totals	4	2	7				4	3			4	27	6	5	10
2. Plant Systems	1	5	1	4	8	1	1	0	3	2	3	5	28	2	2	5
	2	2	1	0	1	0	0	0	1	0	1	3	10	1	2	3
	Tier Totals	7	2	4	9	1	1	0	4	2	4	8	38	3	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
					3		2		2		2			2 2	1 2	

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by± 1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2. Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401 PWR Examination Outline Form ES-401-2

Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 (BW/E02&E10; CE/E02) Reactor Trip – Stabilization – Recovery / 1	X						EK1.05		10
000007 (BW/E02&E10; CE/E02) Reactor Trip – Stabilization – Recovery / 1			X				EK3.01 intended for 054 ak3.01		37
000008 Pressurizer Vapor Space Accident / 3									
000009 Small Break LOCA / 3						***X***	***G2.4.30***		***82***
000011 Large Break LOCA / 3									
000015/17 RCP Malfunctions / 4	X						015AK1.02		19
000022 Loss of Rx Coolant Makeup / 2				X			022AA1.08		22
000022 Loss of Rx Coolant Makeup / 2						***X***	***G2.4.11 intended for G2.4.49***		***83***
000025 Loss of RHR System / 4									
000026 Loss of Component Cooling Water / 8									
000027 Pressurizer Pressure Control System Malfunction / 3									
000029 ATWS / 1			X				EK3.12 intended for APE 024 G2.4.6		24
000029 ATWS / 1						***X***	***G2.4.4 intended for 007EA2.01***		***80***
000038 Steam Gen. Tube Rupture / 3			X				EK3.08		32
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture – Excessive Heat Transfer / 4		X					AK2.02		34
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture – Excessive Heat Transfer / 4				X			EA1.3 intended for EK2.1		73
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture – Excessive Heat Transfer / 4					***X***		***EA2.1 intended for E02 G2.4.6***		***96***
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture – Excessive Heat Transfer / 4					***X***		***AA2.01 intended for E12 G2.4.4***		***99***
000054 (CE/E06) Loss of Main Feedwater / 4					***X***		***AA2.03***		***86***
000055 Station Blackout / 6					X		EA2.05		38
000056 Loss of Off-site Power / 6									
000057 Loss of Vital AC Inst. Bus / 6					X		AA2.19 intended for system 012 k2.01		17
000057 Loss of Vital AC Inst. Bus / 6						X	G2.4.4		40
000058 Loss of DC Power / 6									
000062 Loss of Nuclear Svc Water / 4						***X***	***G2.4.7 intended for sys 003A2.03***		***77***
000065 Loss of Instrument Air / 8			X				AK3.04		50
WE04 LOCA Outside Containment / 3					X		EA2.1		69
W/E11 Loss of Emergency Coolant Recirc / 4									
BW/E04; W/E05 Inadequate Heat Transfer – Loss of Secondary Heat Sink / 4						***X***	***G2.4.20***		***85***
BW/E04; W/E05 Inadequate Heat Transfer – Loss of Secondary Heat Sink / 4					***X***		***EA2.1***		***97***
192006 K1.06 or 193004 K1.15	X						Intended for 027AK1.02		29
K/A Category Totals:	3	1	4	2	3/4	1/5	Group Point Total:		18/6

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 2 (RO / SRO)	Form ES-401-2	
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1									
000003 Dropped Control Rod / 1									
000005 Inoperable / Stuck Control Rod / 1					X		AA2.03		6
000024 Emergency Boration / 1									
000028 Pressurizer Level Malfunction / 2									
000032 Loss of Source Range NI / 7					*X*		***AA2.08***		***84***
000033 Loss of Intermediate Range NI / 7									
000036 (BW/A08) Fuel Handling Accident / 8		X					AK2.02 intended for g2.2.28		64
000037 Steam Generator Tube Leak / 3									
000051 Loss of Condenser Vacuum / 4				X			AA1.04		X
000059 Accidental Liquid RadWaste Rel. / 9									
000060 Accidental Gaseous Radwaste Rel. / 9									
000061 ARM System Alarms / 7									
000067 Plant Fire On-site / 8									
000068 (BW/A06) Control Room Evac. / 8									
000069 (W/E14) Loss of CTMT Integrity / 5									
000074 (W/E06&07) Inad. Core Cooling / 4			X				EK3.11 intended for APE 025 G2.1.32		25
000074 (W/E06&07) Inad. Core Cooling / 4			X				E07 EK3.2		70
000076 High Reactor Coolant Activity / 9						X	G2.3.10 intended for 073K5.03		53
W/EO1 & E02 Rediagnosis & SI Termination / 3									
W/E13 Steam Generator Over-pressure / 4						X	G2.4.6 intended for EK2.2		74
W/E15 Containment Flooding / 5			X				EK3.3		75
W/E16 High Containment Radiation / 9									
BW/A01 Plant Runback / 1									
BW/A02&A03 Loss of NNI-X/Y / 7									
BW/A04 Turbine Trip / 4									
BW/A05 Emergency Diesel Actuation / 6									
BW/A07 Flooding / 8									
BW/E03 Inadequate Subcooling Margin / 4									
BW/E08; W/E03 LOCA Cooldown – Depress. / 4									
BW/E09; CE/A13; W/E09&10 Natural Circ. / 4						X	G2.4.49		71
BW/E09; CE/A13; W/E09&10 Natural Circ. / 4						X	EK3.1 intended for EK2.2		72
BW/E09; CE/A13; W/E09&10 Natural Circ. / 4					*X*		***EA2.2***		***98***
BW/E13&14 EOP Rules and Enclosures									
CE/A11; W/E08 RCS Overcooling – PTS / 4									
CE/A16 Excess RCS Leakage / 2									
CE/E09 Functional Recovery									
192004	X						K1.04 intended for APE 003		4
K/A Category Point Totals:	1	1	3	1	1/2	3	Group Point Total:		9/4

ES-401 PWR Examination Outline Form ES-401-2

Plant Systems – Tier 2/Group 1 (RO / SRO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump										X		A4.01		3
003 Reactor Coolant Pump				X								K4.04 intended for EPE 011Ek2.02		15
004 Chemical and Volume Control											X	2.4.10 or 2.4.31 intended for 2.4.49		5
005 Residual Heat Removal	X											K1.01 intended for K6.03		7
005 Residual Heat Removal											***X***	***G2.1.27***		***78***
006 Emergency Core Cooling				X								K4.11		8
006 Emergency Core Cooling			X									K3.02 intended for 013K1.06		18
007 Pressurizer Relief/Quench Tank								X				A2.05		9
007 Pressurizer Relief/Quench Tank	X											K1.01		11
007 Pressurizer Relief/Quench Tank								***X***				***A2.03***		***79***
008 Component Cooling Water				X								K4.01 intended for A3.06		12
008 Component Cooling Water								***X***				***A2.04***		***81***
010 Pressurizer Pressure Control										X		A4.01 intended for APE 008 G2.4.4		13
010 Pressurizer Pressure Control											X	G2.4.4		14
012 Reactor Protection								X				A2.05 intended for 002 A4.08		2
013 Engineered Safety Features Actuation				X								K4.03 intended for 012 A4.03		16
022 Containment Cooling											X	G2.1.28 intended for 022A1.04		21
022 Containment Cooling			X									K3.02		23
025 Ice Condenser														
026 Containment Spray		X										K2.02		27
026 Containment Spray									X			A3.01 OR 022A3.01. intended for 026K3.01		28
039 Main and Reheat Steam														
059 Main Feedwater											X	G2.1.23		42
061 Auxiliary/Emergency Feedwater					X							K5.03		43
062 AC Electrical Distribution								X				A2.06		44
062 AC Electrical Distribution									X			A3.04		45
063 DC Electrical Distribution	X											K1.02 intended for APE 058AA2.01		41
063 DC Electrical Distribution				X								K4.01		47
064 Emergency Diesel Generator						X						K6.07		48
064 Emergency Diesel Generator	X											K1.03 intended for K6.08		49
073 Process Radiation Monitoring	X											K1.01 intended for 039A1.10		33
073 Process Radiation Monitoring			X									K3.01 intended for071K3.05		51
076 Service Water				X								K4.02 intended for APE 062AA1.07		46
076 Service Water			X									K3.03 intended for A1.02		55
076 Service Water				X								K4.02 intended for 076A3.02		56
078 Instrument Air				X								K4.01		57
103 Containment											X	G2.1.33 Could also be APE 068 G2.1.33 intended for 103K3.02		58
103 Containment											***X***	***G2.1.33 intended for G2.1.30***		***88***
K/A Category Point Totals:	5	1	4	8	1	1	0	3/2	2	3	5/2	Group Point Total:		28/5

ES-401 PWR Examination Outline Form ES-401-2

Plant Systems – Tier 2/Group 2 (RO / SRO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
001 Control Rod Drive								X				A2.03		1
002 Reactor Coolant								X				***A2.04***		***76***
011 Pressurizer Level Control														
014 Rod Position														
015 Nuclear Instrumentation														
016 Non-nuclear Instrumentation														
017 In-core Temperature Monitor	X											K1.01		20
027 Containment Iodine Removal														
028 Hydrogen Recombiner and Purge Control														
029 Containment Purge														
033 Spent Fuel Pool Cooling											X	G2.1.32 intended for APE 026 AA1.01		26
034 Fuel Handling Equipment											X	G2.1.33 intended for 034A1.02		30
035 Steam Generator											X	***G2.2.25 intended for EPE E13 G2.2.25***		***100***
041 Steam Dump/Turbine Bypass Control										X		A4.06, also fits K6.05 but below Imp. 2.5. Intended for 035 K6.02		31
045 Main Turbine Generator														
055 Condenser Air Removal														
056 Condensate											X	G2.1.28		39
068 Liquid Radwaste											X	***G2.1.20***		***87***
071 Waste Gas Disposal														
072 Area Radiation Monitoring				X								K4.03		5
075 Circulating Water		X										K2.03		54
079 Station Air														
086 Fire Protection														
192004 K1.03	X											Intended for 045K5.17		35
K/A Category Point Totals:	2	1	0	1	0	0	0	1/1	0	1	3/2	Group Point Total:		10/3

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 1 - SRO Examination

SRO 2 (Question 77)

K/A Mismatch. This is a loss of Service Water question, not an RCP motor bearing failure system level question. The actual K/A tested is APE 062 G2.4.7, for knowledge of the Loss of Service Water AOP. To meet the K/A, the item should have tested a component failure that would be covered by an RCP failure AOP.

SRO 5 (Question 80)

K/A Mismatch. Determine or interpret power related to a reactor trip, not related to an ATWS, as the item evaluates. As written, the question appears only to evaluate whether 7% power indication is a reason to transition to the ATWS procedure. The K/A for this test item is EPE 029 G2.4.4 for ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. The K/A topic requires determination or interpretation of reactor power as related to a reactor trip. Power is given in the stem. No determination is required. A reactor trip has not occurred, and an ATWS has occurred.

SRO 8 (Question 83)

Selected K/A topic relates to APE 022, Loss of Reactor Coolant Makeup, and requires an ability to perform, without reference to procedures, those actions that require immediate operation of system components and controls.

The test item requires knowledge of AOP actions in an alternate action of a procedure step that is not an immediate action (NOT required from memory). Additionally, conditions in the test item do NOT indicate failure of an automatic function that would require immediate manual action in accordance with Conduct of Operations requirements. This is considered an unacceptable flaw in accordance with NUREG 1021, ES-401, Section E.2.D.

Additionally, the SRO requirement for the selected topic is to evaluate 10 CFR 55.43(b), item 2, Facility operating limitations in Technical Specifications and their bases. Although the test item does not evaluate to this criteria, there is an opportunity to evaluate the criteria of 10 CFR 55.43(b), item 5. (Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency conditions).

The requirements of 10 CFR 55.43(b) are not met because the SRO applicant is not required to select the appropriate procedure for the conditions given. Additionally, the appropriate 10 CFR 55.41(b), item 10, that relates to this test item is not evaluating knowledge specific to the SRO job position.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

Therefore, the requirements of NUREG 1021, ES-401, Section D.2.d, are not met.

SRO 10 (Question 85)

Selected K/A topic relates to system 035 (S/GS), generic knowledge of operational implications of EOP warnings, cautions, and notes.

The selected K/A topic has no relation to 10 CFR 55.43(b). The question does not evaluate the criteria required by 10 CFR 55.43(b), item 5, for selection of procedures.

The related item for the question is 10 CFR 55.41(b), item 10. The question evaluates whether the applicant recognizes parameters required to initiate Feed and Bleed of the RCS, and the action taken to initiate Feed and Bleed during a loss of secondary heat sink, which is knowledge required by both RO and SRO job positions. Therefore, the requirements of NUREG 1021, ES-401, Section D.2.d, are not met. Additionally, there is no link to 10 CFR 55.43(b), item 5, or an SRO-specific link to 10 CFR 55.41(b), item 10.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

SRO 11 (Question 86)

Selected K/A Topic requires the candidate to determine conditions and reasons for AFW pump startup for a Loss of Feedwater event. For an SRO, this question is required to be tied to 10 CFR 55.43(b), item 5, to assess facility conditions and select the appropriate procedure.

The question evaluates system knowledge; specifically, plant conditions are provided that require the determination of which event has occurred and which AFW pumps are operating. There is no procedure selection required. This item would be more closely linked to 10 CFR 55.41(b), item 7, which is RO knowledge of design, components, and functions of control and safety systems, including instrumentation, interlocks, failure modes, and automatic and manual features.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

Therefore, the requirements of NUREG 1021, ES-401, Section D.2.d, are not met.

SRO 13 (Question 88)

K/A Mismatch. This K/A should be rejected for SRO. The test item does not meet the K/A intent (locate and operate components, including local controls). This was only made SRO by asking operability status, which is actually closer to K/A 103, Generic 2.1.33, for ability to recognize system operating parameters which are entry level conditions for technical specifications. This knowledge was already tested once on the exam, on RO Question 58.

SRO 14 (Question 89)

Selected K/A topic required generic knowledge of facility requirements for controlling vital/controlled access. The link to 10 CFR 55.43(b), item 5, would require the SRO to assess plant conditions (perhaps a violation of vital area access has occurred) and select appropriate procedures under normal, abnormal, or emergency conditions. (Correct answer may be to contact the Shift Supervisor in accordance with SAP-200.)

The question does not meet 10 CFR 55.43(b), item 5, criteria, because it asks only who has authority for a condition. These types of questions may be acceptable for items pertaining to 10 CFR 55.43(b), item 3, but in this case the test item is only evaluating knowledge represented by 10 CFR 55.41(b), item 10, at a level of knowledge required by everyone on the operations staff.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

SRO 16 (Question 91)

Selected K/A topic requires generic knowledge of the process for managing troubleshooting activities. The link to 10 CFR 55.43(b), item 5, would require the SRO to assess plant conditions (perhaps a set of conditions that provided for an alternative method of completing work) and select appropriate procedures under normal, abnormal, or emergency conditions. (Correct answer may be to approve and document a Temporary Inoperable Status Change in accordance with SAP-205.)

The question does not meet 10 CFR 55.43(b), item 5, criteria. It is either a reverse logic or 4 True/False statements type of question. (The stem could also be written as: "Which ONE (1) of the following statements does NOT fit?") The test item is only evaluating knowledge represented by 10 CFR 55.41(b), item 10, at a level of knowledge that would

only be required during performance of the job as required by the administrative procedure.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

Therefore, the requirements of NUREG 1021, ES-401, Section D.2.d, are not met.

SRO 18 (Question 93)

K/A Mismatch. Item does not test knowledge of the facility ALARA program. Actual K/A tested was Generic 2.3.8 for knowledge of the process for performing planned Gaseous Radioactive releases.

This is a classic 4 True/False statement test item. The stem can be removed and replaced with: "Which of the following is true?"

SRO 19 (Question 95)

Selected K/A topic required generic knowledge of the process used to track inoperable alarms. The link to 10 CFR 55.43(b), item 5, would require the SRO to assess plant conditions (perhaps an annunciator is alarming due to routine maintenance and must be removed from service) and select appropriate procedures under normal, abnormal, or emergency conditions. (Correct answer may be to disable the annunciator and initiate tracking in accordance with OAP-100.5.)

The question does not meet 10 CFR 55.43(b), item 5, criteria, because it asks only who must approve a condition that is tracked and documented in accordance with an administrative procedure. These types of questions may be acceptable for items pertaining to 10 CFR 55.43(b), item 3, but in this case the test item is only evaluating knowledge represented by 10 CFR 55.41(b), item 10, at a level of knowledge that would be considered outside the scope of required memorization for either an RO or an SRO.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

SRO 21 (Question 96)

K/A Mismatch. Selected topic was related to the E02 EPE for SI Termination, which is covered by the SI Termination procedure in the EOP network. The test item is evaluating

action in the E12 category for Uncontrolled Depressurization of all SGs. The concept that makes the applicant determine that step 17 happens to be in the SI Termination sequence of that procedure is an inappropriate application of the K/A match. The actual K/A being tested is E12 EA2.1.

SRO 23 (Question 98)

Selected topic requires the ability to determine or interpret adherence to procedures and operation within the facility's license and amendments for a Natural Circulation cooldown. The link to 10 CFR 55.43(b), item 5, would require the SRO to assess plant conditions (perhaps a loss of offsite power has occurred and one CRDM fan is running, or one stub bus has been reenergized) and select appropriate procedures under normal, abnormal, or emergency conditions. (Correct answer may be to enter EOP 1.3 and initiate a cooldown at a maximum rate of 50 degrees F per hour.) No other links to 10 CFR 55.43(b) are appropriate or apparent in relation to this topic.

The selected K/A topic has no relation to 10 CFR 55.43(b). The question does not evaluate the criteria required by 10 CFR 55.43(b) item 5 for selection of procedures.

The closest related written exam reference item for the question is 10 CFR 55.41(b), item 10. However, the ability to control these parameter limits is not an SRO-specific function. The item should be appropriately linked directly to 10 CFR 55.45(b), item 7, for operating examination applicability. The question evaluates whether the applicant has memorized parameter limits with one CRDM fan inoperable, which is a SKILL required by both RO and SRO job positions, ONLY during performance of the Continuous Use procedure. Essentially, the task is to perform the cooldown, maintaining parameter limits as required by the procedure.

The lesson material for this topic contains NO SRO-specific objectives or information that would support an SRO level test item outside the scope of 10 CFR 55.43(b) as allowed by NUREG 1021, ES-401, Section D.2.d.

Therefore, the requirements of NUREG 1021, ES-401, Section D.2.d, are not met.

SRO 24 (Question 99)

K/A Mismatch. Selected topic was related to E12, Uncontrolled Depressurization of all SGs, which is covered by an EOP Contingency procedure. The test item is a SGTR question with faulted SG. (may choose between EPE 037 and APE 040) The K/A that would best fit an SRO exam is K/A APE 040 AA2.01.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 1 - SRO Examination

SRO 25 (Question 100)

K/A Mismatch. Selected topic was related to E13, for SG Overpressure, which is an EPE corresponding to an EOP Functional Recovery Procedure. The test item contained no relation to the EPE, but was a Technical Specifications Bases question for the SG System (035) or the Main Steam System (039) The K/A examined is 035 G2.2.25.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 2

K/A mismatch. The question tests RPS control function for runback, not the SPDS relationship to RCS, as the topic requires. There is no stated or implied relationship to the RCS. The operation is turbine control, not SPDS. What is power, how fast is it being raised, and how likely is it to get an OTDT runback at 50% power?

The K/A tested was 012 A2.05, for RPS faulty operation of detectors and function generators.

RO 4

K/A mismatch. The question is not testing an operational implication of the effect on FTC. The operational implication would be an effect on Axial Flux Difference, or action required because of the implication. The question has a Difficulty of 1-5, i.e., the examinee either knows the answer or not. This is a GFES item.

The K/A is 192004 K1.04 (Imp <2.5).

RO 5

K/A mismatch. ARPs do NOT contain these actions as immediate actions, as required by the K/A statement.

The closest K/As for this item are 2.4.10 and 2.4.31.

RO 7

K/A mismatch. Why would anyone close the RHR HX CCW outlet valves? The question is not written in a way that demands job knowledge. K/A not met because the stem says 'design function.' The topic should provide higher cognitive evaluation of effect of loss of RHR HX on the RHR system.

The K/A tested was 005 K1.01, Physical connection or Cause – Effect with CCW system.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 12

K/A mismatch. This is not monitoring auto operation of motor current just by including it in the stem. The focus of this question is on Thermal Barrier cooling. It looks like there is a large leak (with only some of the indications – amps increase and discharge pressure decrease), but the correct answer says that the standby pump will fix it. No trend given on CCW surge tank level. At a minimum, it includes implausible conditions.

The K/A tested was 008 K4.01, for design features that provide for automatic start of CCW pumps.

RO 14

K/A mismatch. The item requires recognition of parameters requiring entry to AOP/EOPs. Each distractor tells the examinee that an AOP is entered, so the premise of question does not meet K/A. Also, this item is tied to 10 CFR 55.43(b), item 5, which is an SRO item and is not suitable for an RO level question.

RO 15

K/A mismatch. RCS pressure at 1500 psi is a SBLOCA or Steam Break, and stem indicates Steam Break. K/A requires LBLOCA.

The K/A tested was 003 K4.04, design features and interlocks that provide for adequate cooling of RCP motor and seals.

RO 16

The item requires knowledge of blocks and bypasses on RPS. This is an ESFAS question, not RPS. There are significant differences between the systems.

The K/A tested was 013 K4.03, knowledge of design features that provide for Main Steam Isolation.

RO 18

K/A mismatch. Plant design basis is NOT what this K/A is supposed to evaluate. ESFAS operation in relation to ECCS is a more operational topic than provided in this question. This should be a SYSTEM question related to sequencer operation. The actual link for this item is SRO 10 CFR 55.43(b), item 1. The item requires memorization of

fact, and is lower cognitive. Just saying that a malfunction of the sequencer causes a delay in energizing ESF components does not examine the physical connection or cause – effect relationship between ESF and ECCS.

The K/A tested was 006 K3.02, loss or malfunction of ECCS effect on fuel.

RO 21

K/A mismatch. The item does not evaluate changes in parameters associated with operating CCS controls. This is a question examining knowledge, not ability or skill, as required by the A1 K/A Topic. There is window dressing here. The item is evaluating RCS leakage causing humidity that will show up in condensation on CCS cooling coils.

The K/A tested was 022 K5.01 (Imp <2.5) or possibly 022 G2.1.28 if a K/A greater than 2.5 is necessary. Purpose and function of components and controls (K/A stretch).

RO 24

K/A mismatch. Highlight **CANNOT** in the question stem. Implausible conditions are given in the stem, and it is doubtful that a control circuit malfunction would affect all PORVs. This is an ATWS T1G1 EPE item, not an Emergency Boration APE (T1G2) as selected for topic.

The K/A tested was 029 EK3.12, knowledge of reasons for action contained in EOPs.

RO 25

K/A mismatch. Required topic is for loss of RHR APE, which is covered by Shutdown AOPs. The question is testing a caution in FR-C.1 (EOP-14.0) that has nothing to do with the APE. An argument can be made that this is an RHR system question.

The K/A tested was 074 EK3.11, knowledge of reasons for guidance contained in EOP for inadequate core cooling.

RO 26

K/A mismatch. The topic requires evaluation of CCW temperatures, not component temperatures serviced by CCW. A lot of information in the question stem is unnecessary. The question could be worded as:

> Which ONE (1) of the following is the high temperature limit for the Spent Fuel Pool?

This wording elicits the same information as the actual question. This has nothing to do with a Loss of CCW, particularly the APE that is covered by an AOP.

Although it is a stretch, the K/A tested was 033 G2.1.32,Ability to explain and apply all system limits and precautions. In this case, the applicant is only required to STATE the limit.

RO 28

K/A mismatch. The question does not test effect of CSS failure on CCS. It only asks the effect on both systems due to loss of electrical bus, plus additional failures.

The K/A tested was either 022 A3.01 or 026 A3.01 or both, ability to monitor automatic operation of CCS or CSS including; initiation of safeguards mode of operation.

RO 29

K/A mismatch. This is a GFES item, made correct by putting a power change in the stem that had nothing to do with the APE. Additionally, stating the change in temperature is not an operational implication. The result of, or actions required because of the temperature change, are operational implications.

The K/A tested was either 192006 K1.06 or 193004 K1.15.

RO 30

K/A mismatch. This item includes nothing about operating the FHS controls in the question. There is no tie to 10 CFR 55.41(b). This question should be considered an SRO level item in accordance with 10 CFR 55.43(b), item 6 or 7. The first sentence is window dressing. The second parameter limit is minutia for a closed reference examination. The item ONLY requires knowledge of parameter limit, and no ability, as required by the topic.

Although a stretch (but likely acceptable), the K/A tested was 034 G2.1.33, and the question tested only an administrative limit.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 31

K/A mismatch. This question is really asking transmitter failure effect on the SG PORV, not effect of PORV malfunction on the SG.

The K/A tested was 041 K6.05 (Imp <2.5), or possibly 041 A4.06, which is the ability to manually operate and monitor atmospheric relief controllers.

RO 33

K/A mismatch. The item does not test change in Air Ejector PRM based on operating MRSS controls, (ability), but it just asks what happens during an event. (Knowledge.) Noun names of radiation monitors should be included. 'A few' gallons per minute is subjective wording. It is also unlikely a trip would be required. This is a lower cognitive level question, and no comprehension is required.

The K/A tested was 073 K1.01.

RO 35

K/A mismatch. The stem should not require statement of negative MTC at 95% power. The item does not ask for the operational implication, but is a GFES question. It only asks boron and what happens to MTC. Assuming no change in rods and no effect on xenon removes the operational validity from this item. What is the operational implication (what the operator will see, interpret, or perform) due to MTC changing?

The K/A tested was 192004 K1.03.

RO 37

K/A mismatch. The bullets are unnecessary (Window Dressing). The reason does not change for any event requiring or causing a reactor trip. To meet the K/A, an examinee could just give conditions requiring trip due to loss of feed, and ask if and why trip is required. (For example, an appropriate answer could be that a Lo-Lo SG level could lead to exceeding the RCS pressure safety limit, which IS the reason for tripping the unit on a loss of feedwater.) This is memorization of fact, and no comprehension is required. Possibly, both B and D are correct answers.

The K/A tested was 007 EK3.01, knowledge of reasons for actions contained in EOP for reactor trip.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 46

K/A mismatch. The question does not evaluate loss of service water APE as required by K/A. Tests start interlocks for SWS pumps at the system level. It requires memorization of fact.

The K/A tested was 076 K4.02, knowledge of design features and interlocks that provide for auto start of SWS pumps.

RO 49

K/A mismatch. Stem wording 'unable to be' should be worded as 'cannot be.' Loss of Day Tank or Loss of fuel oil pump is a significantly different K/A than loss of a fuel oil storage tank.

The K/A tested was 064 K1.03, physical connection or cause effect between DG and fuel oil supply system.

RO 51

K/A mismatch. The item does not evaluate the effect of loss of WGS on ARM/PRM. It asks what effect loss of the PRM has on the system. A noun name for RM-A10 should be provided.

The K/A tested was 073 K3.01, effect that a loss or malfunction of PRM will have on radioactive effluent releases.

RO 53

K/A mismatch. This evaluates the reason for AOP action, not the relationship between exposure and radiation intensity as it applies to the process radiation monitoring system. This K/A is typically thrown out and considered trivia for written exams.

The K/A tested was APE 076 G2.3.10 High RCS Activity generic item for ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 55

K/A mismatch. The item is evaluating KNOWLEDGE of a 'loss or malfunction', not ABILITY to 'operate system controls.' Operational importance seems marginal, at best.

The K/A tested was 076 K3.03, knowledge of the effect that a loss or malfunction of the service water system will have on reactor and turbine building closed cooling water.

RO 56

K/A mismatch. The item does not include operation or monitoring of the DG (Emergency heat loads). Rather, it tests SW Pump start circuitry knowledge under a set of seriously implausible conditions. (Procedure Violation.)

The K/A tested was either 076 A4.01 or 076 K4.02. Overlap with item 46.

RO 58

K/A mismatch. The question does not evaluate the effect of a loss or malfunction. Item provides 4 mini-scenarios (T/F statements). Without reading the stem, an examinee can look at the distractors and say: "Which one of these is a loss of containment integrity?"

The K/A tested was 103 G2.1.33 or possibly APE 069 G2.1.33, same basic knowledge as SRO 13 (Question 88) Ability to recognize system indications that are entry level conditions for technical specifications.

RO 62

K/A mismatch. It is questionable whether this is related to expected RO knowledge. The item evaluates rules for PROCEDURE changes, different K/A than required topic for temporary facility changes. (T-mods.)

The K/A tested was G2.2.6, knowledge of the process for making changes in procedures as described in the safety analysis report.

RO 63

Not an RO question. The question fits SRO level 10 CFR 55.43(b), item 2. It requires memorization of facts and no comprehension is required to answer it.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 67

K/A mismatch. The premise of the question does not evaluate the use of communications procedures, but, rather, simply requires the examinee to know that communications have to be established for local operation of a valve. This is too specific to a situation, rather than generic to all EOPs as required. It requires memorization of fact, and no comprehension is required.

The K/A tested was closest to EPE 055 G2.4.12, knowledge of general operating crew responsibilities during EOP flowchart use (but this is a stretch).

RO 69

The questions is not an RO question. SRO knowledge is appropriate per 10 CFR 55.43(b), item 5.

RO 72

K/A mismatch. The question uses confusing language. There is **no** relationship to facility heat removal systems as required by the K/A.

The closest K/A tested was E10 EK3.1, reasons facility operating characteristics during transients, including coolant chemistry and the effect of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

RO 73

K/A mismatch. The item is EK 2.1, but the question is written and the words are placed in an item for EK 2.2. This does not match the EK 2.1 K/A (Components, function and control of safety systems, interlocks, auto and manual features). Distractor C does not fit. Either an examinee knows this or not. There is no higher cognitive thought required for the way this item is written.

The K/A tested was E12 EA1.3, ability to operate and/or monitor desired operating results during abnormal or emergency situations.

VC Summer
Initial License Examination 2005-301
Test Item Evaluation
Part 2 - RO Examination

RO 74

K/A mismatch. This question requires an RO examinee to know the sequence of action of yellow path procedures from memory. This is an EOP strategy generic K/A question. It does not evaluate the relationship between the operation of any system and the operation of the facility. It is also memorization of fact, and no higher cognitive thinking is required.

The K/A tested was E13 G2.4.6, generic knowledge of symptom based EOP mitigation strategies