Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

10 CFR 55

LIC-14-0076
June 6, 2014

Mr. Steve Garchow
Chief Examiner, Operations Branch, Region IV
U. S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, TX 76011-4511

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Reference: NUREG 1021 Revision 9, "Operator Licensing Examination Standards for Power Reactors"

SUBJECT: NRC Licensed Operator Written Exam Reviews

In accordance with NUREG 1021, the Fort Calhoun Station Training Department has completed a review of the NRC licensed operator written exam that was conducted on May 27, 2014, at the Fort Calhoun Station.

During the exam review, the applicants identified issues with five of the exam questions. In three cases it was determined that the answer in the key was incorrect, in one case a question had two right answers, and it was determined that one question had no right answer. Therefore, Fort Calhoun Station is submitting recommended changes to the original written exam key based on the applicant feedback.

Twelve (12) questions on the reactor operator (RO) section of the exam were missed by 50% or more of the candidates. Eight (8) questions on the senior reactor operator (SRO) exam were missed by 50% or more of the SRO candidates. These questions were reviewed for quality issues. All of the questions that were missed by 50% or more of the exam candidates were discussed with the candidates on May 28, 2014. For each question that was missed by 50% or more of the candidates, the training material and training provided were reviewed.

If you require additional information, please contact Terrence W. Simpkin, Manager, Site Regulatory Assurance, at (402) 533 6263. No commitments are made in this letter.

The following items were provided to you electronically on May 27, 2014:

1. Copies of ES-401-7 and ES-401-8 cover sheets and each applicant's answer (bubble) sheets in pdf format
2. A copy of the exam analysis spreadsheet
3. Seating chart

The following enclosures are included with this transmittal:

1. Grading quality checklist (ES-403-1).
2. The applicant questions and proctor answers provided during the written exam administered on May 27, 2014.
3. The original ES-401-7 and ES-401-8 cover sheets and each applicant's answer (bubble) sheets.
4. Copies of the original applicants answer sheets graded to the original answer key.
5. Copy of the SRO and RO answer keys.
6. Requested changes to FCS exam answer key.
7. Two copies of the exam analysis spreadsheet, one with applicants' names and one with the applicants' names redacted.
8. Original seating chart.
9. ES-201-3 Examination Security Agreement.

Respectfully,

Louis P. Cortopassi
Site Vice President and CNO

LPC/epm

Enclosures

c: M. L. Dapas, NRC Regional Administrator, Region IV, w/o enclosures
J. M. Sebrosky, NRC Senior Project Manager, w/o enclosures
J. C. Kirkland, NRC Senior Resident Inspector, w/o enclosures
Document Control Desk, w/o enclosures

Requested Changes to FCS Exam Answer Key Summary.

The following issues were identified during the 2014 NRC Exam review with the applicants. Each of the following requested changes have been reviewed and approved for submittal as changes to the exam by the FCS Training Director, Shift Operations Superintendent, and Site Vice President. In addition, a peer review of each change requested to the exam was conducted by the Exelon Corporate Training - Operations Training Program Specialist.

Question 46 - Missed by all Applicants

Plant References provide contradictory information on how long the batteries will last after minimizing DC loads. The USAR provides information to support both 4 and 8 hours. However, the 4 hours stated for a station blackout (SBO) in the USAR refers to the commitment to meet a four hour duration during a SBO and does not mean the batteries will only last that long. The bounding "maximum" time is actually for a DBA in accordance with calculation FC-05960. A note in MVA-24, Minimizing DC Loads supports 8 hours for both a DBA and SBO. FCS respectfully requests that choice D be accepted as the only correct answer. Condition Report 2014-0664 has been written to document this condition. (See attachment One)

Question 47 – Missed by 5 of 6 Applicants

FCS requests choice D is accepted as the only correct answer. Choice D is supported by surveillance test EM-ST-EE-0006, Battery No. 2 (EE-8B) Capacity Discharge Test data, provided as attachment Two.

Question 58 - Missed by all Applicants

Based on the CET temperature stated in question 58, this question has only one correct answer. If the temperature indicated by the failed CET is significantly different from the temperatures of the other CETs, the failed CET will be rejected and will not be used in the calculation of "Representative CET Temperature." This is based on the Chauvenet Criterion using standard deviation of valid samples. If the applicant assumed normal operating conditions, which is a standard assumption if conditions are not stated, then a CET temperature of 123.4 °F would be outside one standard deviation and not be used in calculations by the QSPDS. If the temperature indicated by the failed CET is not significantly different from the temperature of the other CETs, the failed CET will be used in the calculation of "Representative CET Temperature." Therefore, choice C is correct based on the value provided for the CET indication in alarm. FCS respectively requests that choice C be accepted as the only correct answer. (See Attachment 3)

Requested Changes to FCS Exam Answer Key Summary.

Question 83 – Missed by both SRO Applicants

FCS believes this question has two correct answers. Choices A and B both describe actions that are taken in AOP-08 (Attachment 4). Although the event occurred in the containment and not the Auxiliary Building, step 8 would eventually be performed for either location. The stem states radiation monitors are not in alarm; therefore VIAS will not have occurred. If VIAS has not occurred, the contingency action for step 8 requires manual VIAS actuation making choice A correct. Choice B is also an action taken in AOP-8 as asked by the question "what is an action taken." FCS respectfully requests that choices A and B both be accepted as correct.

Question 98 – Missed by 1 of 2 SRO Applicants

Question 98 asks "What is the **maximum dose** that an individual at the site boundary could experience in the first two hours following the accident?" The question's choices are based on the regulatory limits and not the actual USAR Chapter 14.14 Steam Generator Tube Rupture Accident analysis. The table in USAR section 14.14.4 indicates the actual dose would be 1.0 REM for either a Pre-Accident Spike or a Concurrent Spike condition. The limits are listed as 25 REM and 2.5 REM for each condition respectively. (Attachment 5) Therefore, FCS believes this question has no correct answer and requests it be deleted.

Attachment 1

Question 46: Station Battery Capacity Following Minimizing DC Loads.

(46)QUESTION NUMBER: 046

The plant has experienced a station blackout (SBO).

What is the maximum time the station batteries are designed to maintain design voltage to DC Bus 1 and 2 in this condition?

A. 4 hours with no operator action.

B.✓ 4 hours if non-essential DC loads are removed.

C. 8 hours with no operator action.

D. 8 hours if non-essential DC loads are removed.

Plant References provide contradictory information on how long the batteries will last after minimizing DC loads. The USAR provides information to support both 4 and 8 hours. However, the 4 hours stated for a station blackout (SBO) in the USAR refers to the commitment to meet a four hour duration during a SBO and does not mean the batteries will only last that long. The bounding "maximum" time is actually for a DBA in accordance with calculation FC-05960. A note in MVA-24, Minimizing DC Loads supports 8 hours for both a DBA and SBO. FCS respectfully requests that choice D be accepted as the only correct answer. Condition Report 2014-0664 has been written to document this condition.

EPG Step: Safety Function 1
Deviation: 1) EOP criteria does not discuss the option of boration to compensate for CEAs that are not fully inserted.

10CFR50.63 and REG GUIDE 1.155 require only that Station Blackout procedures address Station Blackout events with the only further complication being RCS inventory loss (rates of approximately 100 gpm at hot standby RCS pressures, 25 gpm leak rate from each RCP, and normal limits of on-line leakage 10 gpm).

Hence, existing minimum NRC requirements do not require station blackouts with failures of passive reactivity control systems (CEA failures) to be addressed.

In the extremely unlikely case that all regulating and shutdown CEAs are not fully inserted, as long as electrical power is not available to charging and HPSI pumps boration success paths are not available anyway. If more than one CEA was not fully inserted during Station Blackout the following sequence would occur:

The fact that all CEAs had not been inserted would be identified during performance of Standard Post Trip Actions. If all rods were not inserted or cold shutdown boron concentration achieved by the time that an initial event diagnosis was made operators would be directed to the Functional Recovery Procedure EOP-20. As soon as power was restored to bus 1A3 or 1A4, EOP-20 would require that boration to the RCS be implemented using charging or safety injection systems. This justifies deviation 1.

For all emergency events, the reactor must be shutdown. Reactor power lowering in conjunction with negative startup rate is a positive indication that reactivity control has been established. The criterion that no more than one regulating or shutdown CEA is not inserted is in compliance with Technical Specification requirements. 10CFR50.63 and REG GUIDE 1.155 require only that Station Blackout procedures address Station Blackout events. Hence SBO procedures are not required to address Station Blackout events that are complicated by failures of multiple CEAs to fall into the core, causing an event where a reactor trip is required [(R1)].

2. Maintenance of Vital Auxiliaries

ACCEPTANCE CRITERIA

a. 125V DC Bus 1 and 2 energized. ___ ___ ___ ___ ___ ___

b. DC loads have been minimized per Attachment MVA-24, Minimizing DC Loads. ___ ___ ___ ___ ___ ___

EPG Step: Safety Function 2

Deviation:
1) The EOP requires both DC buses to be energized, the EPG only requires one train.
2) The EOP confirms that DC loads have been minimized.
3) The EPG requires one train of 120 VAC instrument bus power be energized.

The 125V DC system is designed as one of the basic sources of power for plant control and instrumentation. During a Station Blackout, both DC buses should remain energized. One DC bus is required as a minimum to provide monitoring and limited control of other safety functions. The loss of one DC bus limits the ability to take an optimal approach to plant recovery. For this reason, both DC buses must be energized to remain in the Station Blackout procedure. The DC buses provide power for DC control power and the four (4) vital AC instrument bus inverters. Two vital AC inverters receive power from DC Bus one (1), the other two from DC bus two (2). By requiring both DC buses to be energized, power will be available for all four AC instrument bus inverters and DC control power. This justifies deviation 1.

Studies at FCS indicate that USAR requirements for 8 hours of battery life, DC loads must be minimized. This condition is applicable during any DBA, not just Station Blackout. This safety function status check item implements the requirements of FC05960, [R4] and helps ensure that battery life is extended. This justifies deviation 2.

Directions to recover a deenergized DC bus during a Station Blackout are in the Functional Recovery Procedures. If a DC bus is deenergized the Safety Function Status Check for Station Blackout vital auxiliaries cannot be met and the Functional Recovery Procedures will be implemented.

The EOP does not address 120 VAC instrument power. The 120 VAC instrument buses are powered via the vital inverters fed from the DC buses. Thus, if a vital bus is energized along with a DC bus, the 120 VAC bus will remain powered. This justifies deviation 3.

Loading of each diesel-generator unit is functionally diagramed in Figure 8.4-2; the control circuits and automatic load sequencer operation are discussed in Section 7.3. The arrangement of the motor control centers supplying Engineered Safeguards and essential loads and the load distribution are shown in USAR Figure 8.1-2.

Periodical maintenance and inspection of the Emergency Diesel Generators is performed and controlled by the plant Preventive Maintenance Program.

8.4.1.3 Design Analysis

The capacity of each Diesel-Generator is adequate to support the operation of required Engineered Safeguards under the most restrictive design basis accident from initiation through long term post accident cooling.

8.4.2 Station Batteries

8.4.2.1 Design Bases

Station batteries are an emergency source of d-c and a-c power for instrumentation and control, and are elements of the d-c systems generally described in Section 8.3.4.

The battery installation is designed to survive without interruption of output or impairment of function of the environmental design bases cited in Section 8.1.1 for Class 1 seismic.

The capacity of the storage batteries in the two, separate d-c systems is adequate for up to 8 hours operation of control and instrumentation devices required in the event of a DBA, or for reactor shutdown and standby, without battery charger operation. To ensure 8 hours of battery capacity manual actions must be taken to minimize DC system loads. The battery worst case loading schedule assumed during these emergency conditions accounts for operation of the following:

DC Bus #1

1. Emergency Bearing Oil Pumps (including starting transient and subsequent shutdown)

2. Emergency Lighting (a portion is load shed per EOPs)

3. Breaker transient loads and DG start attempts

4. Continuous instrument and control loads.

5. Shutdown of nonessential loads.

DC Bus #2

1. Emergency Turbine Seal Oil Pump (including starting transient and subsequent shutdown)

2. Emergency Lighting (a portion is load shed in minutes)

3. Breaker transient loads and DG start attempts

4. Continuous instrument and control loads.

5. Shutdown of nonessential loads.

Analysis has demonstrated that the installed station batteries have adequate capacity to meet the present eight hour load demand. The capacity of the batteries therefore meets the required load demand criteria.

The battery capacity calculation, FC05690, Battery Load Profile and Voltage Drop Calculation, verifies that the class 1E batteries have sufficient capacity to meet station blackout (SBO) loads for four hours assuming certain loads not needed to cope with an SBO are removed. These loads are identified in the battery capacity calculation FC05690 and are incorporated into the applicable emergency/abnormal operating procedure for minimizing DC loads. (Reference 8.7.2)

8.4.2.2 Description and Operation

Two storage batteries are provided. Each battery has sufficient capacity to meet the power demands as described in Section 8.4.2.1. Each battery is installed in a separate room for physical segregation and protection; the rooms are separately ventilated. Battery racks are designed to hold the battery cells in position in the event of the maximum hypothetical earthquake. The arrangement of the battery rooms and their ventilation is as shown in Figure 8.4-3.

During normal operation, the batteries share in meeting control power demand only during peaks when the battery charger rating is exceeded; otherwise the battery chargers meet system demand and simultaneously float-charge the batteries to maintain full charge.

Attachment 2

Question 47: Battery Discharge Capacity Verses Time

(47)QUESTION NUMBER: 047

Battery Charger 1, EE-8C, is faulted and has been removed from service. Battery 1, EE-8A, is supplying Battery Bus AI-41A.

Assuming the DC loads remain constant, the battery voltage will ____________.

A. Lower at a linear rate until fully discharged.

B.✓ Remain stable, and drops rapidly at end.

C. Lower exponentially until fully discharged.

D. Initial short rapid drop off and then lower at a linear rate.

FCS requests choice D is accepted as the only correct answer. Choice D is supported by surveillance test EM-ST-EE-0006, Battery No. 2 (EE-8B) Capacity Discharge Test data.

Voltage Changes During Discharge. At the end of a charge, and before opening the charging circuit, the voltage of each cell is about 2.5 to 2.7 volts. As soon as the charging circuit is opened, the cell voltage drops rapidly to about 2.1 volts, within three or four minutes. This is due to the formation of a thin layer of lead sulphate on the surface of the negative plate and between the lead peroxide and the metal of the positive plate. Fig. 21 shows how the voltage changes during the last eight minutes of charge, and how it drops rapidly as soon as the charging circuit is opened. The final value of the voltage after the charging circuit is opened is about 2.15-2.18 volts. This is more fully explained in Chapter 6. If a current is drawn from the battery at the instant the charge is stopped, this drop is more rapid. At the beginning of the discharge the voltage has already had a rapid drop from the final voltage on charge, due to the formation of sulphate as explained above. When a current is being drawn from the battery, the sudden drop is due to the internal resistance of the cell, the formation of more sulphate, and the abstracting of the acid from the electrolyte which fills the pores of the plate. The density of this acid is high just before the discharge is begun. It is diluted rapidly at first, but a balanced condition is reached between the density of the acid in the plates and in the main body of the electrolyte, the acid supply in the plates being maintained at a lowered density by fresh acid flowing into them from the main body of electrolyte. After the initial drop, the voltage decreases more slowly, the rate of decrease depending on the amount of current drawn from the battery. The entire process is shown in Fig. 22.

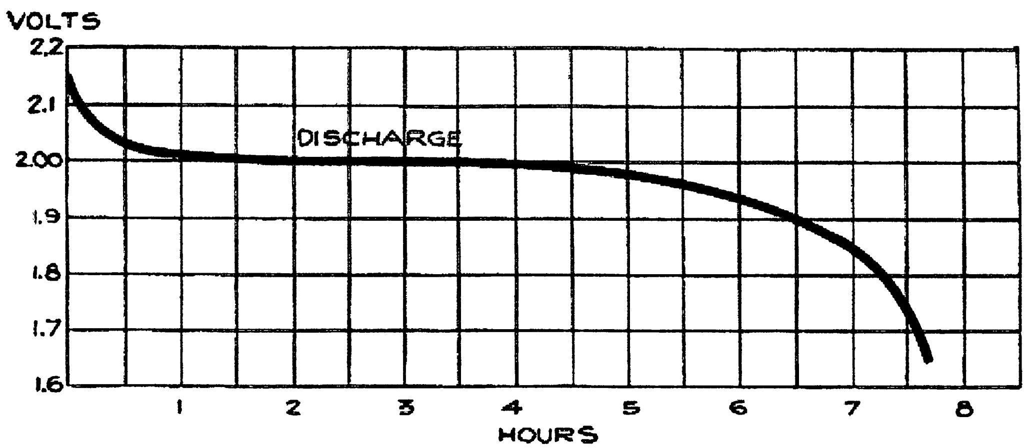


Fig. 22. Voltage Changes During Discharge

Lead sulphate is being formed on the surfaces, and in the body of the plates. This sulphate has a higher resistance than the lead or lead peroxide, and the internal resistance of the cell rises, and contributes to the drop in voltage. As this sulphate forms in the body of the plates, the acid is used up. At first this acid is easily replaced from the main body of the electrolyte by diffusion. The acid in the main body of the electrolyte is at first comparatively strong, or concentrated, causing a fresh supply of acid to flow into the plates as fast as it is used up in the plates. This results in the acid in the electrolyte growing weaker, and this, in turn, leads to a constant decrease in the rate at which the fresh acid flows, or diffuses into the plates. Furthermore, the sulphate, which is more bulky than the lead or lead peroxide fills the pores in the plate, making it more and more difficult for acid to reach the interior of the plate. This increases the rate at which the voltage drops.

The sulphate has another effect. It forms a cover over the active material which has not been acted upon, and makes it practically useless, since the acid is almost unable to penetrate the coating of sulphate. We thus have quantities of active material which are entirely enclosed in sulphate, thereby cutting down the amount of energy which can be taken from the battery. Thus the formation of

10 #GRAPHS

(101)

FORT CALHOUN STATION
FORM

FC-1115A
R6

SURVEILLANCE TEST COVER SHEET

Surveillance Test Number: EM-ST-EE-0006		Work Order Package No: 343985 01	
Title: BATTERY NO. 2 (EE-8B) CAPACITY DISCHARGE TEST			
Due Date: Nov 1, 2009	Drop Date:	Frequency: 103	PMID: 00000051
EEQ: N	Scaffold: N	Trending: N	
Special Conditions: **2009 RFO**			
1. Reason for Test (if other than scheduled):			
2. Postponement explanation (if required):			
Expected Completion:	Supervisor or System Engineer:		Date:
3. Test results acceptable?		[X] Yes	[ ] No
Tech Specs satisified? (Not required for acceptable test)		[ ] Yes	[ ] No
Supervisor or System Engineer: David Koran		Date: 12-2-09	Time: 0739
Additional Code analysis required within 72 hours?		[ ] Yes | [ ] No	[ ] N/A
		STA Initials:	Date:
Responsible System Engineer: ~~JOHNSON, MATTHEW R~~ KALRA			
4. ST Coordinator initial test review		Initials: [illegible]	Date 12-3-09
WO required?		[ ] Yes	[X] No
Frequency Increased?		[ ] Yes	[Y] No
CR required?		[ ] Yes	[X] No
Retest required?		[ ] Yes	[V] No
System Engineer test review: Y		Initials: [illegible]	Date: 12/13/09
IST Coordinator test review: N		Initials:	Date:
ST Coordinator final test closeout review:		Initials: [illegible]	Date: 12-14-09

(SO-G-23)

Fort Calhoun Station
Unit No. 1

EM-ST-EE-0006

SURVEILLANCE TEST

CAPACITY DISCHARGE TEST FOR STATION BATTERY NO. 2 (EE-8B)

Change No.	EC 47417, EC 45495
Reason for Change	Added figures for load bank. Added criteria for termination of test. Changed Independent Verification to Concurrent Verification. Changed multimeter model from 8060A to 189. Identified Tech Spec numbers with a TS.
Requestor	Mike Swan, Vern Smolinski
Preparer	Ron Shirley
Issue Date	10-29-09 3:00 pm

CAPACITY DISCHARGE TEST FOR STATION BATTERY NO. 2 (EE-8B)

SAFETY RELATED

1. PURPOSE

1.1 This procedure provides instructions for performing a discharge on the battery and calculating the battery capacity.

1.2 This test is performed at a frequency of the first refueling shutdown after installation, and every third refueling shutdown thereafter.

1.3 This test satisfies, in part, the requirements of Technical Specifications, TS 3.7(2)b.

2. REFERENCES/COMMITMENT DOCUMENTS

2.1 Technical Specifications, TS 2.7(1)g. and TS 3.7(2)b.

2.2 SO-G-23, Surveillance Test Program

2.3 Ongoing Commitments

- AR 12818, LIC-92-0192

2.4 TM C173.0010, Technical Manual for C&D (Charter Power Systems) Batteries

2.5 Drawings

Drawings	File	Description
P&ID Fig 8.1-1	12234	One Line Diagram Plant Electrical
11405-E-8	12244	125 VDC Misc. Power Diagram
2C6289	44129	DC Dist. Panel EE-8G Schematic

3. DEFINITIONS

- CICV - Corrected individual cell voltage
- ELC - Electrolyte level correction
- ETC - Electrolyte temperature correction
- ICV - Individual cell voltage
- MCM - Thousands of circular mils
- SG - Specific gravity
- VCT - Voltage correction for temperature

Attachment 9.8 - Data Sheet 2A

Step 7.2 Battery No. 2 Isolated Date 11/23/07 Time 1645

Step 7.7 Electrolyte temperature correction factor K_1 1.011

Step 7.13.3 Battery Terminal Voltage Prior to Test 119.86 VDC

Test Time	Date/Time	Discharge Current		Battery Terminal Voltage	
0	11/23/07 2115	272.4	274.2	115.6	117.55
15	2130	272.6	275.4	115.4	115.06
30	2145	272.2	275.	115.3	114.92
45	2200	272.7	275.6	115.2	114.81
100	2215	272.2	275.2	115.1	114.67
115	2230	272.2	275.8	114.9	114.50
130	2245	272.3	275.38	114.7	114.31
145	2300	272.7	275.72	114.5	114.13
200	2315	272.4	275.28	114.3	113.95
215	2330	272.6	275.4	114.1	113.75
230	2345	272.4	275.6	113.9	113.53
245	11/24/07 0000	272.5	275.2	113.8	113.36
300	0015	272.3	275.4	113.5	113.10
315	0030	272.8	275.6	113.3	112.89
330	0045	272.8	275.8	113.0	112.65
345	0100	272.3	275.2	112.8	112.45
400	0115	272.4	275.4	112.5	112.16
415	0130	272.5	275.6	112.3	111.87
430	0145	272.5	275.6	112.0	111.61
445	0200	272.5	275.6	111.7	111.34
500	0215	272.2	275.0	111.5	111.12
515	0230	272.8	275.6	111.2	110.80
		Com	DVM	Com	DVM

Attachment 9.8 - Data Sheet 2A

Test Time	Date/Time	Discharge Current		Battery Terminal Voltage	
530	11/24/09 0245	272.7	275.4	110.8	110.46
545	11/24/09 0300	272.7	275.6	110.5	110.09
* 600	11/24/09 0315	272.6	275.6	110.2	109.81
615	11/24/09 0330	272.4	275.2	109.8	109.40
630	11/24/09 0345	272.2	275.2	109.5	109.06
645	11/24/09 0400	272.8	275.6	109.1	108.73
* 700	11/24/09 0415	272.2	277.6	108.6	108.23
715	11/24/09 0430	272.8	275.8	108.1	107.74
730	11/24/09 0445	272.4	275.2	107.6	107.21
745	11/24/09 0500	272.7	275.6	107.0	106.64
* 800	11/24/09 0515	272.8	275.8	106.5	106.12
815	11/24/09 0530	272.5	275.4	105.7	105.35
830	11/24/09 0545	272.7	275.4	104.8	104.40
845	11/24/09 0600	272.8	275.6	103.0	102.62
* 900	11/24/09 0615	TEST	STOPPED AT	8 HOURS 46	MIN 52 SEC
915					
930					
945					
1000					
		COM	DVM	COM	DVM

Step 7.13.14A Final Battery Terminal Voltage 102.47 VDC

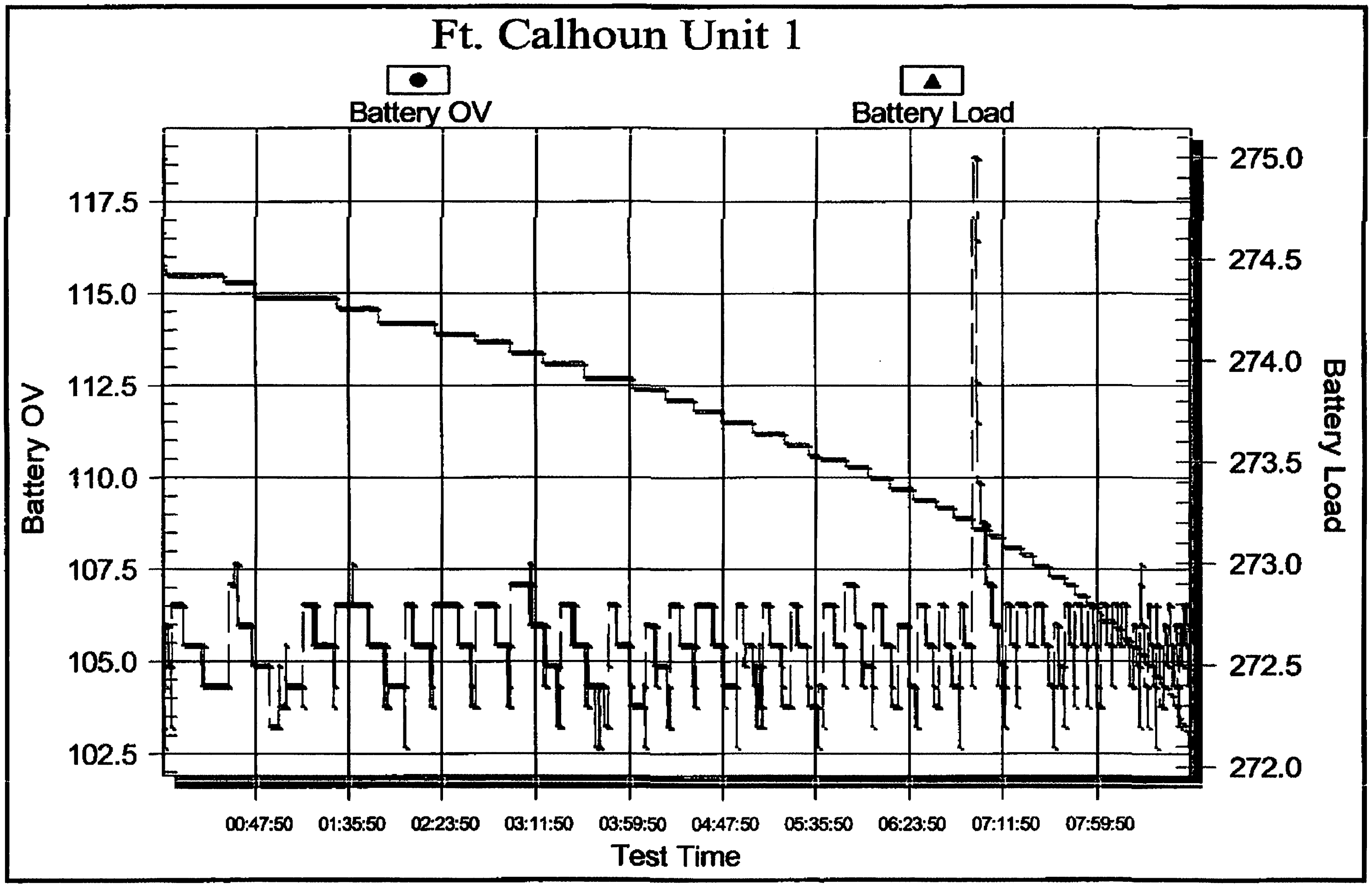
Step 7.13.14C Final Discharge Current 13.76MV Amps 275.2

Step 7.13.14D End of Discharge Date 11/24/09 Time 06:03

8:46:52

8.78 HOURS

Battery OV & Load Graph



Attachment 3

Question 58: Suspect CET Input to QSPDS Calculations

Original Question

(58)QUESTION NUMBER: 058

The "A" QSPDS is displaying a suspect alarm for one of the Core Exit Thermocouples (CET). The alarm has NOT been acknowledged.

1) How is the UN-ACKNOWLEDGED suspect CET alarm displayed on the QSPDS plasma screen, and
2) Will this suspect CET input be used in QSPDS calculations:

	1) How is the suspect CET displayed	2) used in calculations
A.	Normal Mode with a question mark in front of the value ?123.4	Not used
B.	Normal Mode with a question mark in front of the value ?123.4	Still used
C.	Inverse Mode with a question mark in front of the value ?123.4	Not used
D.	Inverse Mode with a question mark in front of the value ?123.4	Still used

Based on the CET temperature stated in question 58, this question has only one correct answer. If the temperature indicated by the failed CET is significantly different from the temperatures of the other CETs, the failed CET will be rejected and will not be used in the calculation of "Representative CET Temperature." This is based on the Chauvenet Criterion using standard deviation of valid samples. If the applicant assumed normal operating conditions, which is a standard assumption if conditions are not stated, then a CET temperature of 123.4 °F would be outside one standard deviation and not be used in calculations by the QSPDS. If the temperature indicated by the failed CET is not significantly different from the temperature of the other CETs, the failed CET will be used in the calculation of "Representative CET Temperature." Therefore, choice C is correct based on the value provided for the CET indication in alarm. FCS respectively requests that choice C be accepted as the only correct answer.

Fort Calhoun Station
Unit No. 1

OI-QSP-1

OPERATING INSTRUCTION

QUALIFIED SAFETY PARAMETER DISPLAY SYSTEM OPERATION

Change No.	EC 49234
Reason for Change	Converted from WordPerfect to Word.
Requestor	N/A
Preparer	J. Briggs
Issue Date	06-17-10 3:00 pm

QUALIFIED SAFETY PARAMETER DISPLAY SYSTEM OPERATION

ATT PURPOSE PAGE

PRECAUTIONS

1. When energizing QSPDS the associated inverter power supply may transfer to the bypass transformer. The Shift Manager shall specify whether the inverter should be manually placed in bypass per OI-EE-4 (requires inverter inoperability), when energizing the QSPDS and/or HJTC heaters.

2. Dampers HCV-724A/B and HCV-725A/B have been permanently failed in their accident positions. Indication for these dampers is for information only. Abnormal indication needs to be verified by ARP-ERFCS before action is taken.

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specification:
   - 2.21, Post-Accident Monitoring Instrumentation
2. USAR:
   - Sections 7.5.5 and 7.5.6

Attachment 2 - Parameter Alarms

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision Number_______________ Date:________ ________

2. At least one Qualified Safety Parameter Display System channel is operational. ________

PROCEDURE

NOTES

1. A summary of display behavior for Parameter Alarms is shown in Figure 2.

2. There are three types of parameter alarms:

   - Setpoint alarms occur when a parameter value exceeds its high or low setpoint (not all parameters have a setpoint).
   - Out-of-Range/Bad Data alarms occur for an out-of-range sensor, a failed sensor, an electrical failure, or a software error.
   - **Suspect alarms occur when an input to a multi-input calculated value is out-of-range/bad data but the value can still be calculated with the remaining inputs, or if the core exit temperatures are not within a valid range.**

3. In the INVERSE MODE, the NORMAL MODE character and background colors are reversed.

1. WHEN a Parameter Alarm occurs for either a Setpoint or an Out-of-Range/Bad Data Alarm,
   THEN the relevant Top-Level Page enters the alarm state AND the corresponding System Alarm Indicator on the current Display Page is shown in BLINK MODE.

   a. IF the parameter is shown on a lower level sector page,
      THEN higher level pages will show the relevant sector numbers in BLINK MODE.

   b. In the case of a Setpoint Alarm, the parameter value is shown in INVERSE MODE with an adjacent asterisk.

Attachment 2 - Parameter Alarms

PROCEDURES (continued) (✓) INITIALS

1

c. In the case of an Out-of-Range/Bad Data Alarm, the parameter value field is filled with question marks in INVERSE MODE.

d. In the case of a Suspect Alarm, a single question mark is displayed by the affected calculated value OR core exit temperature in INVERSE MODE.

2. WHEN the Operator acknowledges an alarm,
THEN:

a. In the case of a Parameter alarm, the relevant System Alarm Indicators and sector numbers remain in INVERSE MODE.

b. In the case of a Setpoint Alarm, a second asterisk is shown next to the parameter value field.

c. In the case of an Out-of-Range/Bad Data Alarm when the parameter has a high or low setpoint, the system and sector alarms either stay in the INVERSE or NORMAL MODE but change the question marks to the NORMAL MODE. IF the parameter does not have any setpoints, then the system and sector alarms along with the question marks return to the NORMAL MODE.

d. In the case of a Suspect Alarm, the question mark and all alarms return to the NORMAL MODE.

Attachment 7 - Core Exit Thermocouple (CET) Temperature Calculation

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision Number_______________ Date:________ ________

2. At least one Qualified Safety Parameter Display System channel is operational. ________

PROCEDURE

1. The QSPDS displays individual Core Exit Temperatures with a Core Map, the highest and next highest Core Exit Temperature in each quadrant, and a representative Core Exit Temperature.

2. The Representative Core Exit Temperature is calculated based on statistical analysis with practical checks from other inputs.

   a. Using the Representative CET Temperature the Pressure AND Temperature Saturation Margins are calculated and displayed. However, the CET calculated variables will be flagged with a question mark during the time that the QSPDS is identifying Out-of-Range and Suspect CET inputs.

   b. WHEN the QSPDS completes one (1) complete cycle of the CET algorithm without flagging or removing a flag from any CET input, THEN the inputs are considered stable AND accurate and the question mark is removed from the calculated variables.

   c. The CET calculated variables will also be flagged with all question marks if the number of valid CET inputs is less than nine (9).

3. Inputs considered invalid are either failed or deviated from the mean of the CET inputs by a specific amount.

   a. Failed inputs are displayed as all question marks.

   b. Deviated inputs are displayed as a question mark in front of the displayed value indicating it is suspect.

Attachment 7 - Core Exit Thermocouple (CET) Temperature Calculation

PROCEDURE (continued) (✓) INITIALS

4. WHEN a Saturation Alarm occurs (either RCS or Upper Head),
THEN all previous valid (before the Saturation Alarm) CET inputs will be used in calculating CET variables.

   a. Previously Suspect CET inputs will be used in the calculation if their temperature value falls within a specified band of the mean temperature.

5. The Core Exit Temperature alarms if the temperature is greater than a high setpoint. However, Suspect inputs will NOT be alarmed.

Figure 2 - Alarm Behavior

PAGE ELEMENTS	ALARM STATUS	ACK STATUS	MECHANISMS	EXAMPLES
System Alarm Indicators and Sector Numbers	Normal or Return to Normal	- - -	Normal Mode	CORE PPP 2
	Parameter Alarm on Associated Page	UNACK	Blink Mode	(CNMT PPP) (2)
		ACK	Inverse Mode	(CNMT PPP) (2)
		ACK	Normal Mode Inverse Mode	CNMT PPP (2)
Parameter Value Field	Normal or Return to Normal	- - -	Normal Mode	123.4
	Setpoint Alarm	UNACK	Inverse Mode 1 asterisk	(123.4*)
		ACK	Inverse Mode 2 asterisks	(*123.4*)
	Out-of-Range / Bad Data Alarm	UNACK	Inverse Mode Question Marks	(??????)
		ACK	Normal Mode Question Marks	??????
	Suspect Alarm	UNACK	Inverse Mode Question Mark	(? 123.4)
		ACK	Normal Mode Question Mark	? 123.4

Blink Mode: Alternation of the Normal and Inverse Modes.

5.1.2B (continued)

7. QSPDS Computer Functional Design

   a. The QSPDS computer is a dual-channel seismically-qualified display of safety parameters, designed to provide a backup function to the primary SPDS (Ref. 10.3.8, and Ref. 10.9.3). Refer to Attachment 10 of this SDBD for further design details.

   b. The QSPDS computer is designed to monitor, process and display information relevant to the core cooling conditions. The QSPDS uses inputs from the heated junction thermocouples (HJTCs), core exit thermocouples (CETs), and loop temperature and reactor pressure instruments (Attachment 6, Items 7, 8, 9, 10, 29 and 30; and Attachment 7, Items 96 through 113) to provide display of ICCI variables and indication of reactor vessel liquid inventory, reactor coolant temperature, and saturation/superheat margin (Ref. 10.4.14, Ref. 10.4.15, Ref. 10.4.35, Ref. 10.4.36, Ref. 10.4.37, and Ref. 10.9.3). In addition to the display of the ICC input variables and of the calculated variables, the QSPDS computer performs the alarming functions of impending ICC, interfacing to the ERF computer for data transmission and display on the SPDS, flagging of suspect calculations and of hardware or software malfunctions, and control of power to HJTC heaters (Ref. 10.4.5, Ref. 10.4.29, Ref. 10.7.1, and Ref. 10.9.3).

   c. Data used by the QSPDS computer is checked for an out-of-range condition, or when the configured setpoint is reached. Thermocouple inputs are checked for an open thermocouple condition. An alarm is initiated to warn of a bad data or of a setpoint that is reached. The setpoint alarm occurs when an input parameter exceeds its high or low value, around the setpoint. The bad data alarm occurs when an input parameter value exceeds its allowable range. This may result from an out-of-range sensor, a failed sensor, an electrical failure or a software error.

5.1.2B7 (continued)

d. A suspect temperature reading within a set of CETs is identified from the application of the Chauvenet Criterion using standard deviation of the valid samples. Values which fail the validity check are flagged before transmission to the ERF computer and are displayed on the ERF computer operator interface displays with a blinking white question mark (?). These values are displayed on the QSPDS computer PDU, preceded by a question mark (?) for suspect data, or are replaced by a series of question marks (???) for out-of-range data (Ref. 10.3.8, Ref. 10.4.5, Ref. 10.4.6, and Ref. 10.9.3).

e. The QSPDS computer monitors, processes and transmits to the ERF computer for display, selected Regulatory Guide 1.97 variables (Ref. 10.4.25 and Ref. 10.4.27). Refer to Attachment 12 of this SDBD for a listing of these variables.

f. The QSPDS computer is designed for an operational availability of 99 percent (Ref. 10.9.3).

C. Monitoring Instrumentation and Control Design

1. Monitoring Instrumentation and Control Design

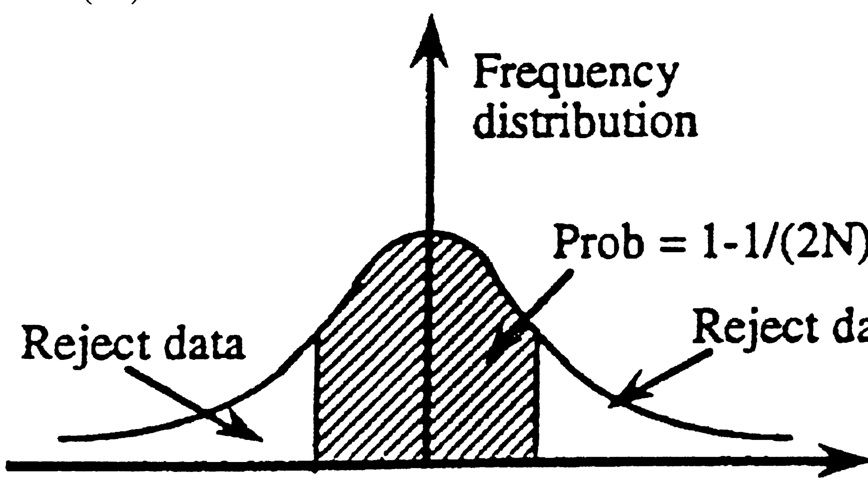
a. The SPDS software resident in the ERF computer host system uses the information transmitted to it by the DAS in order to provide a continuous indication of plant parameters or derived variables representative of Fort Calhoun status (Ref. 10.4.4, Ref. 10.4.5, Ref. 10.4.6, Ref. 10.4.31, Ref.10.4.42, Ref.10.4.43, 10.9.6 and Ref. 10.9.15). The SPDS software uses three level hierarchical paging system: Top Level (Level One), Midlevel bargraphs (Level Two) and supporting P&ID displays (Level Three) to provide the operator with information about:

- Reactivity Control (Ref. 10.9.13 and Ref. 10.9.14)
- Vital Auxiliaries (Ref. 10.9.13 and Ref. 10.9.14)
- Reactor Coolant System Inventory Control (Ref. 10.9.13 and Ref. 10.9.14)
- Reactor Coolant System Pressure Control (Ref. 10.9.13 and Ref. 10.9.14.
- Core Heat Removal (Ref. 10.9.13 and Ref. 10.9.14)
- Reactor Coolant System Heat Removal (Ref. 10.9.13 and Ref. 10.9.14)
- Containment Integrity (Ref. 10.9.13 and Ref. 10.9.14)

Statistical Rejection of "Bad" Data – Chauvenet's Criterion

Occasionally, when a sample of N measurements of a variable is obtained, there may be one or more that appear to differ markedly from the others. If some extraneous influence or mistake in experimental technique can be identified, these "bad data" or "wild points" can simply be discarded. More difficult is the common situation in which no explanation is readily available. In such situations, the experimenter may be tempted to discard the values on the basis that something must surely have gone wrong. However, this temptation must be resisted, since such data may be significant either in terms of the phenomena being studied or in detecting flaws in the experimental technique. On the other hand, one does not want an erroneous value to bias the results. **In this case, a *statistical* criterion must be used to identify points that can be considered for rejection. There is no other justifiable method to "throw away" data points.**

One method that has gained wide acceptance is *Chauvenet's criterion*; this technique defines an acceptable scatter, in a statistical sense, around the mean value from a given sample of N measurements. The criterion states that all data points should be retained that fall within a band around the mean that corresponds to a probability of 1-1/(2N). In other words, data points can be considered for rejection only if the probability of obtaining their deviation from the mean is less than 1/(2N). This is illustrated below.



The probability 1-1/(2N) for retention of data distributed about the mean can be related to a maximum deviation d_{max} away from the mean by using the Gaussian probabilities in Appendix A. For the given probability, the nondimensional maximum deviation τ_{max} can be determined from the table where

$$\tau_{max} = \frac{\left|(X_i - \overline{X})\right|_{max}}{S_X} = \frac{d_{max}}{S_X}$$

and S_X is the precision index of the sample. Therefore, all measurements that deviate from the mean by more than $\tau_{max}S_X$ can be rejected. A new mean value and a new precision index can then be calculated from the remaining measurements. No further application of the criterion to the sample is allowed; Chauvenet's criterion may be applied only *once* to a given sample of readings.

The table below gives the maximum acceptable deviations for various sample sizes. Values of d_{max}/S_X for other sample sizes can easily be determined using the Gaussian probability table in Appendix A.

Chauvenet's Criterion for Rejecting a Reading

Number of Readings (N)	Ratio of Maximum Acceptable Deviation to Precision Index (d_{max}/S_X)
3	1.38
4	1.54
5	1.65
6	1.73
7	1.80
8	1.87
9	1.91
10	1.96
15	2.13
20	2.24
25	2.33
50	2.57
100	2.81
300	3.14
500	3.29
1,000	3.48

Example 4
For the 10 temperature measurements in Example 2, determine if any should be rejected by Chauvenet's criterion.

SOLUTION
Inspecting the data from Example 2, the fifth reading of T=98.5°F appears to deviate substantially from the others and is therefore a candidate for rejection. From the table above for N=10, the maximum dimensionless deviation is $d_{max}/S_X=1.96$. Since $S_X \approx 0.49°F$ in this case, $d_{max}=1.96S_X=(1.96)(0.49°F)=0.96°F$. The deviation for the fifth reading is 1.125°F so it can indeed be rejected, although no others can. Eliminating this point and recalculating the mean and precision index results in

$$\overline{X} = 97.25°F$$

$$S_X = 0.31°F$$

Comparing these values with those calculated in Example 2, $\overline{X}$ is decreased by only about 0.13% while S_X is decreased by over 37%.

TD C490.0350

USERS GUIDE

FOR

THE QUALIFIED SAFETY PARAMETER

DISPLAY SYSTEM

CAUTION

DRAWINGS ARE FOR INFORMATION ONLY. CHECK THE DRAWING CONTROL PROGRAM FOR THE LATEST REVISION.

CHECK WITH VENDOR PRIOR TO ORDERING ANY PARTS TO VERIFY PART NUMBERS.

5.3 CORE EXIT THERMOCOUPLE TEMPERATURE

The QSPDS displays individual core exit temperatures with a core map, the highest and next highest core exit temperature in each quadrant, and a representative core exit temperature.

The representative core exit temperature is calculated based on averaging all valid CET inputs with practical checks from other inputs. Using the Representative CET Temperature (average temperature), and pressurizer Pressures the Temperature and Pressure Saturation Margins are calculated and displayed. However the CET calculated variables will be flagged with a question mark during the time that the QSPDS is identifying out of range and suspicious CET inputs. When the QSPDS completes one (1) complete cycle of the CET algorithm without flagging or removing a flag from any CET input then the inputs are considered stable, and accurate and the question mark (?) is removed from the calculated variables. Also the CET calculated variables will be flagged with all question marks if the number of valid CET inputs is less than 2 CET's per Quadrant per channel.

Those inputs considered invalid are either failed or deviated from the mean of the CET inputs by a specific amount. The failed inputs are displayed as all question marks. While the deviated inputs have a question mark in front of the displayed value indicating it is suspicious.

Whenever a saturation alarm occurs (either RCS or Upper Head) all previous valid (before the saturation alarm) CET inputs will be used in calculating CET variables. Previously suspected CET inputs will be used in the calculation if their temperature value fall within a specified band of the mean temperature.

The core exit temperature alarm if the temperature is greater than a high setpoint. However, suspected inputs will not be alarmed.

Attachment 4

Question 83: Actions for a Dropped Fuel Assembly

Original Question

(83)QUESTION NUMBER: 083

A new fuel bundle has dropped off the FH-1 Refueling Machine grapple onto the refueling cavity floor and appears to be damaged. The following conditions exist:

- The inner PAL door is being held open by an RP tech for trash removal.
- The outer PAL door is open and inoperable.
- The Equipment Hatch is installed.
- All Area and Process Monitors indicate normal background radiation.

1) Which AOP does the CRS enter, **and**
2) What is an action the operators are directed to perform?

	Procedure	Action
A.	AOP-08 Fuel Handling Incident	Trip VIAS using both CRHS test switches
B.✓	AOP-08 Fuel Handling Incident	Direct the EONA to close the inner PAL door within 1 hour
C.	AOP-12 Loss of Containment Integrity	Trip VIAS using both CRHS test switches
D.	AOP-12 Loss of Containment Integrity	Direct the EONA to close the inner PAL door within 1 hour

FCS believes this question has two correct answers. Choices A and B both describe actions that are taken in AOP-08. Although the event occurred in the containment and not the Auxiliary Building, step 8 would eventually be performed for either location. The stem states radiation monitors are not in alarm; therefore VIAS will not have occurred. If VIAS has not occurred, the contingency action for step 8 requires manual VIAS actuation making choice A correct. Choice B is also an action taken in AOP-8 as asked by the question "what is an action taken." FCS respectfully requests that choices A and B both be accepted as correct.

1.0 PURPOSE

This procedure provides guidance in the event an irradiated Fuel Assembly is dropped or otherwise damaged.

2.0 ENTRY CONDITIONS

A fuel assembly has been damaged which may be indicated by any of the following:

A. Area radiation monitors increase.

B. "RM-050 CNTMT PARTICULATE HIGH RADIATION" alarm (AI-33C; A33C).

C. "RM-051 CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

D. "RM-052 STACK/CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

E. "RM-062 AUX BLDG VENT STACK HIGH RADIATION" alarm (AI-33C; A33C).

F. Containment air particulate high radiation indication upscale.

G. Ventilation Isolation Actuation Signal (VIAS).

H. While handling fuel, the Hoist Load Indicator shows low Hoist weight.

I. Possible damage to Fuel Assembly is observed.

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS

1. Announce and repeat the following over the plant communications system:

   "Attention all personnel. Attention all personnel. A fuel handling accident has occurred in the (location). All non-essential personnel should immediately evacuate the area."

2. Direct the Radiation Protection Technician to survey the affected area.

3. IMPLEMENT the Emergency Plan.

4. **IF the incident was in the Auxiliary Building,**
   THEN GO TO Step 8.

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

It is required to close the following within 1 hour of event initiation: Room 66 Roll-up Doors or the Equipment Hatch, all containment penetrations open to the outside atmosphere and one door in the PAL.

5. Direct the Shift Outage Manager to close the Equipment Hatch within 1 hour.

5.1 **IF** the Equipment Hatch can not be closed,
THEN perform the following:

a. Direct Shift Outage Manager to close the construction access.

b. Direct the Security Shift Supervisor to close Room 66 Doors 1009-1, 1013-1 and 1013-4 within 1 hour.

6. Direct Shift Outage Manager to close all containment penetrations open to the outside atmosphere within 1 hour.

7. Direct the EONA to close at least one PAL door within 1 hour.

INSTRUCTIONS

8. Verify VIAS actuation PER Attachment A, VIAS Actuation.

CONTINGENCY ACTIONS

8.1 **IF** VIAS did **NOT** actuate, **THEN** ensure VIAS actuates by performing the following:

a. Obtain a key from panel AI-30A or AI-30B.

b. Trip VIAS using both CRHS test switches.

- 86A/CRHS Test Switch (AI-30A)
- 86B/CRHS Test Switch (AI-30B)

c. Verify VIAS actuation PER Attachment A, VIAS Actuation.

Attachment 5

Question 98: SGTR Dose at Site Boundary

(98)QUESTION NUMBER: 098

The plant was at 100% power with all TS LCOs met when a Steam Generator Tube Rupture occurred.

- Off-Site Power has been lost.
- Natural Circulation is established.
- All Safeguards equipment functioned properly.

What is the maximum dose that an individual at the site boundary could experience in the first two hours following the accident?

A. 25 mRem TEDE

B. 75 mRem TEDE

C. 5 REM TEDE

D.✓ 25 REM TEDE

Question 98 asks "What is the maximum dose that an individual at the site boundary could experience in the first two hours following the accident?" The question's choices are based on the regulatory limits and not the actual USAR Chapter 14.14 Steam Generator Tube Rupture Accident analysis. The table in USAR section 14.14.4 indicates the actual dose would be 1.0 REM for either a Pre-Accident Spike or a Concurrent Spike condition. The limits are listed as 25 REM and 2.5 REM for each condition respectively. Therefore, FCS believes this question has no correct answer and requests it be deleted.

USAR-14.14

Safety Analysis

Steam Generator Tube Rupture Accident

Rev 15

Safety Classification:	Usage Level:
Safety	Information

Change No.:	EC 51265
Reason for Change:	The USAR title pages and sections are being reformatted into the current format showing the safety classification and usage level. Additional editorial changes (e.g., font, capitalization, title corrections, etc.) are being made for consistency.
Preparer:	M. Edwards
Issued:	07-12-11 3:00 pm

Table of Contents

List of Tables

14.14 License Renewal Supplement

Since the performance of the FSAR analysis, the steam generator tube rupture accident has been evaluated using recent computer codes (Reference 14.14-6) which also addresses a release pathway through the steam driven auxiliary feedwater pump (FW-10).

14.14.1 General

The steam generator tube rupture accident is a penetration of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the radiological safety standpoint, as a leaking steam generator tube would allow transport of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with shell side water in the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the steam system dump and bypass valves. Noncondensible radioactive materials would be discharged through the condenser vacuum pumps to the atmosphere. FW-10 (Auxiliary Feedwater Pump) was considered as a release path also. Modification MR-FC-91-039 provided Operations with the ability to override containment isolation actuation signal (CIAS) from steam generator blowdown (SGBD) sampling isolation valves to draw a steam generator blowdown sample. This modification allows for establishment of an acceptable/analyzed blowdown sampling release path should plant conditions warrant the drawing of a blowdown sample. Appropriate Abnormal Operating Procedures (AOPs)/Emergency Operating Procedures (EOPs) instruct the Operator to reroute the SGBD sampling discharge flow to a lineup leading to the radioactive waste disposal system.

Diagnosis of the accident is facilitated by radiation monitors in the blowdown sample lines from each steam generator and in the condenser vacuum pump discharge line. Additionally, the CIAS signal for HCV-2506A/B and HCV-2507A/B can be overridden for a period not to exceed two hours. This allows samples to be obtained from the steam generators to aid in the identification of the affected steam generator. These monitors initiate alarms in the control room and inform the operator of abnormal activity levels and that corrective action is required; the steam generator blowdown is automatically isolated should a high activity level be reached. Additionally, steam generator blowdown is automatically isolated upon reactor trip.

The behavior of the systems varies depending upon the size of the rupture. For leak rates up to the capacity of the charging pumps in the chemical and volume control system (CVCS), reactor coolant inventory can be maintained and an automatic reactor trip would not occur. The gaseous fission products would be released to atmosphere from the secondary system at the condenser vacuum pump discharge. Those fission products not discharged in this way would be retained by the main steam, feedwater and condensate systems.

For leaks that exceed the capacity of the charging pumps, the pressurizer water level and pressure decrease and a thermal margin/low pressure (TM/LP) reactor trip results. The turbine then trips and the steam system dump and bypass valves open. The steam generator water level indicators aid in detection of these larger leaks since the water inventory in the leaking steam generator may increase more rapidly than that of the intact steam generator following reactor trip.

The amount of radioactivity released increases with break size. For this analysis, an area equivalent to a double-ended break of one steam generator tube is assumed for the rupture size. At normal operating conditions, the leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the charging pumps; the reactor coolant system pressure decreases and a low pressurizer pressure trip occurs. Following the reactor trip, the reactor coolant average temperature is reduced by exhausting steam through the steam dump and bypass system. The radioactivity exhausted through the steam dump and bypass system flows to the condenser where the noncondensible gaseous products are released to the atmosphere.

Based on guidance contained in the EOPs, which serve to minimize releases and off-site doses, the operator will use the steam dump and bypass valves to reduce the reactor coolant system hot leg temperature. Once the dump valves are closed, operation of the bypass valve (PCV-910) allows steam to pass to the condenser to reduce this temperature to 510°F. When the hot leg temperature is below 510°F, the affected steam generator is isolated to terminate the release source. This hot leg isolation temperature of 510°F bounds the minimum cold leg temperature requirements for adequate reactor coolant pump (RCP) net positive suction head (NPSH). This steam generator hot leg isolation temperature value is consistently used in the EOPs and this analysis, regardless of whether RCPs are available (i.e., off-site power is available). When the reactor coolant temperature is 300°F the operator is assumed to place the shutdown heat removal system into operation and isolate both steam generators. Although Shutdown Cooling entry can occur at 350°F, a lower entry temperature is conservative for this analysis.

14.14.2 Method of Analysis

The analysis of a steam generator tube rupture was performed using a digital computer simulation of the system. The simulation includes neutron kinetics with fuel and moderator temperature feedback, the effect of the shutdown group of control element assemblies (CEAs) and the reactor coolant and main steam systems including the pressurizer, steam generators, and steam dump and bypass valves. The method of analysis used provides radiological consequences results that bound operator actions that may be taken in accordance with the EOPs (Reference 14.14-7). This results from the analysis assumption that the operator will feed and steam both steam generators for cooldown, for the first two hours. Cooldown to less than 510°F would occur much sooner than this when using the EOPs.

14.14.3 Radiological Consequences for Steam Generator Tube Rupture Accident

The Steam Generator Tube Rupture (SGTR) Radiological Consequences assessment followed the guidance provided in Regulatory Guide (RG) 1.183 with exceptions noted in Section 14.1.6 and discussed in Reference 14.14-4. Table 14.14-1 lists the assumptions/parameters used to develop the radiological consequences following SGTR and were documented in References 14.14-4 and 14.14-10.

Two cases of thermal hydraulic data of a postulated SGTR were generated for input to the radiological consequences evaluation (Reference 14.14-11). One case assumed the rupture tube is at the top of the tubesheet on the cold side of the steam generator and is called the cold side break case; the other case assumed the rupture is at the top of the tubesheet on the hot leg side of the steam generator and is called the hot side break case. Due to the assumed simultaneous loss of offsite power with the reactor trip, the reactor is cooled down by releasing steam via the main steam safety valves (MSSVs)/atmospheric dump valve (ADV). The primary coolant with elevated iodine concentrations (pre-accident or concurrent iodine spike) flows into the faulted steam generator and the associated activities are released to the environment due to the secondary side steam releases.

Before the reactor trip, the activities are released from the main condenser air ejector (AEJ). After the reactor trip the steam release is via the MSSVs/ADV. A portion of this steam is released via the turbine exhaust of the turbine driven auxiliary feedwater (AFW) pump. However, the atmospheric dispersion factor of this release point is bounded by that of the MSSVs/ADV (see Section 14.1). Consequently, the dose analyses conservatively assumes that all of the steam is discharged via the MSSVs/ADV.

The spiking primary coolant activities leaked into the intact steam generator at a conservative 1 gpm primary to secondary leakage rate are also released to the environment via secondary steam releases.

Both the cold side break case and the hot side break case were evaluated. The doses from the worst case are calculated in detail and are reported in the results. The most critical parameter that affects the doses is the total break flow that flashes from the time of the reactor trip until the isolation of the affected steam generator at t = 2 hours. This directly released primary coolant for the hot side break is greater than the cold side break. Additionally, the total steam release from the intact SG is essentially the same for the two break cases. Therefore, the hot side break case will produce greater radiological consequences and is evaluated and described in detail.

Since there is no postulated fuel damage associated with this accident, the main source of radioactivity is the activity in the primary coolant system. Two spiking cases were addressed: a pre-accident iodine spike and a concurrent iodine spike per RG 1.183.

a. Pre-Accident spike – the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DEI-131, which is the transient TS limit for full power operation. The initial primary coolant noble gas activity is assumed to be at TS levels.

b. Concurrent spike – the initial primary coolant iodine activity is assumed to be at TS of 1 μCi/gm DEI-131 (equilibrium TS limit for full power operation). Immediately following the accident, the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 335 times the equilibrium appearance rate corresponding to the 1 μCi/gm DEI-131 coolant concentration. The duration of the assumed spike is eight hours. The initial primary coolant noble gas activity is assumed to be at TS levels.

The initial secondary side liquid and steam activity is relatively small and its contribution to the total dose is negligible compared to that contributed by the rupture flow and is therefore, not considered in this assessment.

Faulted SG Release

Per Reference 14.14-4 a postulated SGTR will result in a large amount of primary coolant being released to the faulted steam generator via the break location with a significant portion of it flashed to the steam space. The noble gases in the entire break flow and the iodine in the flashed flow are assumed immediately available for release from the steam generator without retention. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and partition factor. Before the reactor trip at 629.5 seconds, the activities in the steam are released to the environment via the main condenser air ejector. All steam noble gases and organic iodine are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returned to both steam generators and assumed to be available for future steaming release. After the reactor trip, the break flow continues until the primary system is fully depressurized and in equilibrium with the secondary side of the faulted SG. To maximize the calculated off-site doses, the condenser is assumed unavailable after reactor trip. The steam is released from the MSSVs/ADV. All activity releases from the faulted steam generator cease when it is isolated at 120 minutes after the accident. Table 14.14-2 provides the break flow of primary coolant into the faulted SG and Table 14.14-3 provides the steam releases from the faulted SG as a function of time (References 14.14-4 and 14.14-11). Reference 14.14-11 also provides the integrated break flow that flashes in the faulted SG.

Intact SG Release

The activity release from the intact SG is due to normal primary to secondary leakage and steam release from the secondary side. A portion of the break flow activity that is transferred to the intact SG via the condenser before reactor trip will also be released from the intact SG during the cool down phase. The primary to secondary leak rate is conservatively assumed to be 1 gpm at STP.

Steam generator tube uncovery occurs at t = 679.5 seconds for a period of 112 minutes (Reference 14.14-12). During this tube uncovery period, the halogens in the flashed portion of the primary to secondary tube leakage are assumed immediately available for release to the environment. The flash fractions associated with the SG tube leakage during the tube uncovery period are provided in Reference 14.14-11.

The halogens in the non-flashed portion of the leakage mixes uniformly with the SG liquid mass and is released into the steam space in proportion to the steaming rate and the partition factor. When the tubes are totally submerged, the halogens in the tube leakage are assumed to mix uniformly with the SG liquid and are released as discussed above for the non-flashed portion of the tube leakage. The noble gases are released freely to the environment without retention in the SG for the entire duration of the event. The steam releases from the MSSVs/ADV continue for 144.6 hours, at which time shutdown cooling (SDC) is initiated via operation of the SDC system and environmental releases are terminated. Table 14.14-4 provides the cumulative steam releases from the intact SG (Reference 14.14-4 and 14.14-11).

Per RG 1.183, the 2-hour EAB dose must reflect the "worst case" 2-hour activity release period following the accident. The worst 2-hour EAB dose is expected to occur during the first two hours of the event because it coincides with the break flow in the faulted SG and tube uncovery in the intact SG. Also, the noble gas release rate is at its highest at the onset of the event. Reference 14.14-10 shows that the maximum 2-hour EAB dose occurs during t = 0 to 2 hours for both the pre-accident and concurrent iodine spike cases.

Accident Specific Control Room Assumptions

The SGTR will result in a safety injection actuation signal (SIAS), 678.5 seconds into the event, which will result in the initiation of the CR emergency ventilation. An additional delay of 44 seconds was assumed to account for a coincident LOOP. Due to single failure of the recirculation damper, the emergency recirculation filtration system is assumed to be unavailable for the first 2 hours after the event. The remaining CR parameters were discussed in Section 14.1.

14.14.4 Results

The total effective dose equivalent (TEDE) doses for the various locations are shown below.

	Pre-Accident Spike (Rem)	**Reg Limit**	**Concurrent Spike (Rem)**	**Reg Limit**
EA-B (2-hour dose)	1.0	25	1.0	2.5
LPZ (for duration of event)	0.5	25	0.5	2.5
Control Room (30-day dose)	1.0	5.0	1.5	5.0

The results remain below the regulatory limits set by 10 CFR 50.67 and RG 1.183.

Doses were rounded up to the nearest 0.5 Rem. The maximum 2 hour dose period for the EAB dose was in the 0 to 2 hours time period.

14.14.5 Specific References

14.14-1 Not Used

14.14-2 Not Used

14.14-3 Not Used

14.14-4 Application for Amendment of Operating License, Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components, LIC-05-107, October 26, 2005, OPPD to USNRC Document Control Desk

14.14-5 Not Used

14.14-6 Letter from OPPD (R.L. Andrews) to NRC (J.M. Taylor), dated April 10, 1987

14.14-7 Response to Fort Calhoun Station Condition Report 199700976 Action Item 1 (including 10CFR50.59 evaluation), April 1998

14.14-8 OPPD-NA-8303-P, Revision 1, Omaha Public Power District Reload Core Analysis Methodology: Transient and Accident Methods and Verification, January 1993

14.14-9 Letter NRC -93-0301 from NRC (S.D. Bloom) to OPPD (T.L. Patterson), dated November 2, 1993

14.14-10 OPPD Calculation FC06820, Revision 1, Site Boundary and Control Room Dose following a Steam Generator Tube Rupture Accident Using Alternate Source Terms

14.14-11 AREVA Document 32-5051831-01, FCS RSG - Steam Generator Tube Rupture with Cooldown to SDC Entry Conditions, March 11, 2005

14.14-12 AREVA Document 32-5048718-02, FCS RSG: Inputs for Radiological Consequences Analysis - Tube Uncovery, March 30, 2005

Table 14.14-1 – Radiological Analysis Assumptions and Key Parameter Values-SGTR

Power Level	1530 MWt
Reactor Coolant Mass	250,000 lbm
Break Flow to Affected Steam Generator	Table 14.14-2
Time of Reactor Trip	629.5 sec
Termination of Release to Affected SG	2 hours
Amount of Break Flow that Flashes	Reference 14.14-11
Leakrate to intact SG	1 gpm at STP
Failed/Melted Fuel Percentage	0%
RCS TS Iodine Concentration	Table 14.1-6
RCS TS Noble Gas Concentration	Table 14.1-6
RCS Equilibrium Iodine Appearance Rates	Table 14.1-7
Pre-Accident Iodine Spike Activity	Table 14.1-7
Accident Initiated Spike Appearance Rate	335 Times Equilibrium
Duration of Accident Initiated Spike	8 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (post-accident minimum)	45,708 lbm
Faulted SG Liquid mass (post-accident minimum)	70,261 lbm
Initial Mass in SGs (total)	157,185 lbm
Form of All Iodine Released to the Environment via SGs (total)	97% elemental; 3% organic
Time period of Tube Recovery (Intact SG)	t = 679.5 sec to t = 7399.5 sec
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion)	1.0 (Released to Environ w/o holdup)
Fraction of Noble Gas Released either SG	1.0 (Released to Environ w/o holdup)
Partition Factor in Condenser AEJ	2000 (elemental iodine), 1 (organic iodine and noble gases)
Cumulative Steam Releases from Faulted SG	Table 14.14-3
Cumulative Steam Releases from Intact SG	Table 14.14-4
Steam Flowrate to Condenser from Faulted SG Before Trip	944.8 lb/s (0-629.5 secs)
Steam Flowrate to Condenser from Intact SG Before Trip	940.3 lb/s (0-629.5 secs)
Termination of Release from SGs	144.6 hours
Environmental Release Points	Condenser Evac. Discharge (0-629.5 secs) MSSVs/ADV (629.5 secs - 144.6 hr)
CR Emergency Ventilation: Initiation Signal/Timing	
Initiation time (SIAS)	678.5 sec

Table 14.14-2 – Integrated Break Flow from Primary Coolant to Faulted Steam Generator-SGTR

Time (sec)	Integrated Break Flow (lbs)
0	0
629.5	26852.7
722.5	30048.1
3600	119070.9
7200	284570.7

Table 14.14-3 – Cumulative Steam Releases from Faulted Steam Generator-SGTR

Time (sec)	MSSV Release (lbs)
0	0
629.5	0.0
722.5	7338.8
3600	106262.6
7200	146230.0

Table 14.14-4 – Cumulative Steam Releases from Intact Steam Generator-SGTR

Time (hr)	MSSV Release (lbs)	ADV Release (lbs)	Total Release (lbs)
0	0	0	0
0.6736	0.0	0.0	0.0
1	24686.0	0.0	24686.0
2	64111.8	0.0	64111.8
4	104000.3	69221.8	173222.1
8	171279.7	186449.0	357728.7
24	365063.0	524165.6	889228.7
30	425470.0	629425.0	1054895.0
144.6	----	3080000.0	3505470.0