INITIALS

CAUTION

- Do **NOT** Reduce RCS level Below Elv 39 ft. 4.9 in. using this Addendum.
- As pressurizer level lowers to 10% AND before the PZR level goes off scale low, COMPARE Pressurizer level with RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" level indication.

3.0 PERFORM the following to lower RCS inventory:

3.1 The following instrumentation **SHALL** be operable prior to RCS Draindown to 39' 4.9":

3.1.1 RHR heat exchanger inlet and outlet temperature with indication on QDPS OR chart recorder for all operable trains. _____

3.1.2 RHR pump flow with indication on QDPS for all operable trains. _____

3.1.3 RHR pump motor current indication (amps) for all operable trains. _____

3.1.4 "RHR PUMP CURRENT LO" Annunciators on Lampbox 1M02 for the operable RHR pumps is **NOT** removed from service. _____

3.1.5 RCS level sightglass has been walked down in last twelve hours and satisfies requirements. _____

3.1.6 Both trains of RVWL are operable with at least two QDPS displays available. _____

3.1.7 Core Exit Thermocouples, five per train, two trains. _____

3.2 ENSURE Adequate capacity is available in radwaste to receive volume drained from RCS. REFER TO Addendum 3, Determination of RCS Volume to be Drained. _____

3.3 A dedicated, reliable communication line, headphones being the preferred method, is established between Control Room personnel and RCB sightglass watch. _____

INITIALS

NOTE

- Venting and draining operations should be coordinated with the Radwaste Operator and Health Physics.
- RCS level changes SHALL be made slowly in a controlled manner to minimize effects on reactor vessel level indications.
- The RCS level sightglass SHALL be continuously monitored during all draining and refill operations.
- One of the following pressurizer vent paths SHALL be established prior to draining the pressurizer:
    - The Pressurizer spray line vent valves RC-0502 and RC-0503 OPEN to atmosphere. (Preferred Method)
    - A minimum of one Pressurizer Code Safety Valve REMOVED.
- In addition to the specified temperature limits, the intent is to maintain RCS temperature as low as allowed by existing plant conditions, core cooling capabilities or other limiting criteria to maximize margin to core boiling:
    - (Mode 5) MAINTAIN RCS core exit temperature (RHR HX inlet temp when CETs NOT available) less than 140°F.

3.4 IF PZR level wll be lowered below 10% Cold Calibrated level, THEN ENSURE RCS level sightglass is in service. ______

INITIALS

NOTE

WHEN RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, THEN ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup" PERFORMED daily and DOCUMENTED in a temporary log.

3.5 COMMENCE temporary logging of LG-3662 sightglass valves once per shift (Ref. Procedure Step 3.37). _____

CAUTION

Prior to draining the pressurizer, a vent path SHALL be established to prevent drawing a vacuum.

3.6 IF lowering pressurizer level to below 10% Cold Calibrated level (55 ft 6 inch elevation), THEN PERFORM the following:

3.6.1 ENSURE RCS level sightglass is being monitored. _____

3.6.2 ENSURE pressurizer vent path established. _____

3.6.3 DOCUMENT Vent Path. ____________________ _____

3.6.4 OPEN PZR Spray Valves RC-PCV-655B and RC-PCV-655C. _____

3.6.5 DETERMINE RCS volume to be drained using Addendum 3, Determination of RCS Volume to be Drained. _____

3.6.6 VERIFY "DIVERT LCV-0112A" in the AUTO position. {CP004} _____

3.6.7 Manually RAISE letdown flow using "PRESS CONT PCV-0135". {CP004} _____

3.6.8 COMPARE Pressurizer level with RCS level sightglass level indication (Should agree within 6 inches. 1% of Pressurizer Cold Calibrated level equals 4 inches). _____

INITIALS

CAUTION

IF RVWL system indicates less than 100%, THEN the draindown SHALL be stopped and the level difference between RVWL and RCS level sightglass investigated.

3.6.9 IF pressurizer Cold Calibrated level AND RCS level sightglass do **NOT** agree within 6 inches, THEN STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. ______

3.6.10 IF pressurizer level will be maintained, THEN REFER TO Step 3.22 of this Addednum. ______

CAUTION

IF during any RCS draining process, fluctuations are observed in RHR pump flow, amps, OR discharge pressure, THEN any RCS drain in progress SHALL be stopped to allow RCS water level to stabilize and any RCS water level recovery SHALL be initiated as necessary to ensure RHR system operation.

3.7 PLACE Reactor Vessel head temperature on trend display. (Plant Computer points IITE2040 and IITE3040) ______

3.8 IF lowering of RCS level is to continue, THEN DRAIN to between 47 ft. 4 in. and 46 ft. 5 in. elevation as indicated on RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)". ______

INITIALS

NOTE

WHEN RVWL Sensor Point 1 has been uncovered, THEN indicated temperature will rise to about 750°F due to heating from the heated junction thermocouple.

3.9 VERIFY water in reactor vessel head less than 180°F as indicated by Plant Computer display RC12 (8112). _____

3.10 PLACE the Reactor Vessel head to pressurizer equalizing line in service as follows:

3.10.1 ENSURE the reactor vessel head venting manifold is connected per 0POP02-RC-0003, Addendum 1, Filling and Venting the RCS. {RCB on RV Head} _____

3.10.2 ENSURE the RV to PZR Equalizing Line is aligned IAW 0POP03-ZG-0009, Mid-Loop Operation. _____

INITIALS

3.11 WHEN RCS level is between 47 ft. 4 in. and 46 ft. 5 in. as indicated on RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)", THEN PERFORM the following:

3.11.1 OPEN the following valves to remove water plug (Loop Seal) from Head and Pressurizer vent manifold and Rx Head Vent line:

- "1(2)-RC-0509 RX VESSEL HEAD" "VENTING MANIFOLD DRAIN" {RCB On RV Head} _____
- "1(2)-RC-0507 RX VESSEL HEAD" "VENTING MANIFOLD VENT VALVE" {RCB On RV Head} _____

3.11.2 IF the reactor vessel head vent valves are operable, THEN OPEN the Reactor Vessel head vent valves. {CP005}

- "ISOL HV-3657A" _____
- "ISOL HV-3657B" _____
- "ISOL HV-3658A" _____
- "ISOL HV-3658B" _____
- "HEAD VENT THROT VLV HCV-0601" _____
- "HEAD VENT THROT VLV HCV-0602" _____

3.11.3 IF desired, THEN OPEN PRT "1(2)-RC-0025 N2 SUPPLY ISOL". {RCB 6 ft E of PRT} _____

3.11.4 IF RCS level will be maintained, THEN REFER TO Step 3.22. _____

INITIALS

3.12 IF RCS level is to remain below 47 ft. 4 in, THEN PERFORM the following:

- CLOSE "1(2)-RC-0507 RX VESSEL HEAD" "VENTING MANIFOLD VENT VALVE" {RCB On RV Head} _____
- ENSURE "1(2)-RC-0509 RX VESSEL HEAD" "VENTING MANIFOLD DRAIN"{RCB On RV Head} remains OPEN to vent the HEAD. _____

NOTE

- The temperature rise will occur when sensor is uncovered prior to RVWL point indicating dry.
- WHEN RVWL Sensor Point 1 has been uncovered, THEN indicated temperature will rise to about 750°F due to heating from the heated junction thermocouple.
- WHEN RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, THEN ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup" PERFORMED daily and DOCUMENTED in a temporary log.

CAUTION

Inadequate head venting due to rapid draining can cause Reactor Vessel water level to remain higher than loop level.

3.13 IF lowering of RCS level is to continue, THEN DRAIN RCS until RVWL Sensor Point 1 (45' 3.4") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2004, A-Train, IITE3004, C-Train) _____

3.14 COMPARE RCS level sightglass indication with RVWL level. _____

3.15 IF RVWL AND RCS level sightglass do **NOT** agree within 6 inches, THEN STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. _____

3.16 IF RCS level will be maintained, THEN REFER TO Step 3.22. _____

INITIALS

3.17 REFER TO Addendum 3, Determination of RCS Volume to be Drained, to determine draindown volume. ______

NOTE

- The temperature rise will occur when the sensor is uncovered prior to RVWL point indicating dry.
- WHEN RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, THEN ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup" PERFORMED daily and DOCUMENTED in a temporary log.
- Do **NOT** Reduce RCS level Below Elv 39 ft. 4.9 in. using this Addendum.

3.18 IF lowering of RCS level is to continue, THEN DRAIN RCS until RVWL Sensor Point 2 (39' 4.9") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2007, A-Train, IITE3007, C-Train) ______

3.19 COMPARE RCS level sightglass indication with RVWL level. ______

3.20 IF RVWL AND RCS level sightglass do **NOT** agree within 6 inches, THEN STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. ______

3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. ______

3.22 MAINTAIN RCS level using any of the following:

- Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train ______
- Raise RCS level as necessary using CCP normal charging or seal injection ______
- Reduce RCS level as necessary using low pressure letdown ______

3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) ______

3.24 To raise RCS level, GO TO Step 2.0 of this addendum. ______

3.25 RETURN TO Procedure Step in effect. ______

INITIALS

NOTE

- This addendum vents a voided Reactor Vessel head when the RCS in Modes 5 or 6.
- HJTC Train "A" or Train "C" RVWL Sensor 1 indications are found at Computer Points IITE2004 and IITE3004, respectively.

1.0 CHECK the following indications for Reactor Vessel Head voiding: _____

- Pressurizer level rising

  AND

- VCT level constant OR rising

  AND

- RVWL Sensor 1 temperature rising

2.0 IF Reactor Vessel Head voiding is indicated, THEN vent the Reactor Vessel Head to the PRT as follows:

2.1 OPEN Head Vent Isolation Valves:

- ISOL HV-3657A and ISOL HV-3658A _____

  OR

- ISOL HV 3657B and ISOL HV-3658B _____

2.2 OPEN Head Vent Throttle Valve(s):

- HCV-0601 _____
- HCV-0602 _____

INITIALS

2.3 MONITOR for the following indications of the Reactor Vessel Head being vented: _____

- Pressurizer level lowering

AND

- RVWL Sensor 1 temperature lowering

AND

- PRT pressure rising

2.4 WHEN the Reactor Vessel Head void is vented, THEN PERFORM the following:

2.4.1 ENSURE CLOSED the Head Vent Throttle Valve:

- HCV-0601 _____
- HCV-0602 _____

2.4.2 ENSURE CLOSED the Head Vent Isolation Valves:

- HV-3657A _____
- HV-3657B _____
- HV-3658A _____
- HV-3658B _____

1.0 Purpose:

1.1 Provides instructions for closing Personnel Air Lock (PAL) doors to establish containment closure during work activities inside Reactor Containment Building (RCB) that require both PAL doors to be open.

2.0 Instructions:

2.1 PERFORM the following to close Personnel Air Lock doors (M-90):

2.1.1 ENSURE the following conditions at RCB PAL door:

- Door-seating surfaces are clear of all obstructions. ______
- Handwheel position indication in "OPEN" position. ______
- Both door latch pins fully retracted to ensure NO interference with door closure. ______

2.1.2 PULL RCB PAL door CLOSED. ______

2.1.3 WHEN RCB PAL door is flush with door jam, THEN ROTATE RCB PAL door hand wheel to "CLOSED" position to engage door latch pins. ______

2.1.4 ENSURE the following conditions at MAB PAL door:

- Door-seating surfaces are clear of all obstructions. ______
- Handwheel position indication in "OPEN" position. ______
- Both door latch pins fully retracted to ensure NO interference with door closure. ______

2.1.5 PULL MAB PAL door CLOSED. ______

2.1.6 WHEN MAB PAL door is flush with door jam, THEN ROTATE MAB PAL door hand wheel to "CLOSED" position to engage door latch pins. ______

NOTE

An ECO Caution Tag will be hanging on 1(2)-XC-0037 and MCB switch for PAL Seal OCIVs to prevent pressurizing door seals with the doors open.

2.2 PERFORM the following to pressurize PAL door seals:

2.2.1 OPEN "1(2)-XC-0037 RCB PERSONNEL AIRLOCK SEAL AIR SUPPLY ISOLATION VALVE". (MAB 60′, Room 326) ____

2.2.2 NOTIFY Control Room {8614, 8610, 1111(7953, 8683, 2222)} that BOTH PAL doors are in the CLOSED position **AND** valve 1(2)-XC-0037 is OPEN. ____

2.2.3 REQUEST Control Room to pressurize PAL door seals by taking "PERS AIR LOCK SEAL OCIVS" switch to OPEN. (CP005) ____

Unit:_____ Date:_____ Initial Time:_____

1. RECORD data at 15 minute intervals.

2. Pressurizer cooldown rate SHALL **NOT** exceed 200°F (160°F/hr ADMIN LIMIT) in any one hour. (TRM 3.4.9.2) Cooldown rate is the actual cooldown over the hour period. (i.e.: $°F_{\text{Temperature recorded 60 minutes prior to current time}} - °F_{\text{Temperature current}} = \Delta°F/\text{hour}$)

3. OBTAIN RCS pressure from QDPS Detail Data Menu Page 1 (PT403, PT404, PT405, PT406 or PT407)

4. OBTAIN RCS temperature from QDPS Detail Menu Page 1 (TE414, TE424, TE434, or TE444)

5. RCS cooldown rate SHALL **NOT** exceed 100°F (80°F/hr ADMIN LIMIT) in any one hour period during cooldown within the limits of Addendum 1. (Technical Specification 3.4.9.1) (Ref 2.70) Cooldown rate is the actual cooldown over the hour period. (i.e.: $°F_{\text{Temperature recorded 60 minutes prior to current time}} - °F_{\text{Temperature current}} = \Delta°F/\text{hour}$)

6. RECORD the differential temperature between the Pressurizer water space TI-0608 and RCS temperature (4). IF cooldown is due to an unisolable RCS leak, THEN differential temperature limit is less than or equal to 250°F. Differential temperature limit is less than or equal to 320°F for normal cooldowns. (Ref 2.62)

7. INITIAL the appropriate column.

8. 15 minute cooldown rate will be converted to an hourly rate.
Calculation: (i.e.: $(°F_{\text{Temperature recorded 15 minutes prior to current time}} - °F_{\text{Temperature current}}) \times 4 = \Delta°F/\text{hour}$)

9. IF any readings obtained are outside the specified limits, THEN PERFORM the following: (Ref 2.7)

   9.1 Immediately NOTIFY the Shift Manager/Unit Supervisor of the out of specification reading.

   9.2 STOP the plant cooldown.

   9.3 ENSURE RCS temperature/pressure within the specified limits from the QDPS.

   9.4 MAINTAIN existing RCS temperature and pressure.

NOTE
The cooldown SHALL **NOT** be resumed until Engineering authorizes the cooldown to be restarted.

   9.5 NOTIFY Engineering to perform an Engineering Evaluation to determine the effects of the out-of-limit (**NOT** ADMIN LIMIT) condition on the structural integrity of the RCS or Pressurizer, as applicable.

10. WHEN Pressurizer vapor space temperature TI-0607 is **NOT** functional, THEN use the associated functional Pressurizer water space temperature TI-0608 for all Pressurizer temperature indications called out in this procedure. Use of the liquid temperature element alone is more conservative [will provide higher indicated change for a given actual system change] and better represents actual metal temperature. Use of the liquid temperature indication alone will provide assurance that cooldown limits will **NOT** be exceeded. (CREE 02-3367)

    Example:

    - Pressurizer vapor space temperature TI-0607 is non-functional, THEN substitute Pressurizer water space temperature TI-0608, for Pressurizer vapor space temperature TI-0607 in this procedure.

Unit:____ Date:____ Initial Time:____ Page __ of __

TIME (1)	PRZR VAPOR SPACE (10)			PRZR WATER SPACE			RCS PRESS QDPS (3)	RCS			PRZR-RCS WTR	INITIAL (7)
	TI-0607	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate	TI-0608	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate		QDPS (4)	°F/HR (8) 15 minute rate	°F/HR (5) Rolling Hourly rate	Delta T (6)	

Personnel Participating in cooldown:

Name	Initials	Name	Initials

Cooldown completed by: __________ Operator _____ Date _____ Time

Data Sheet 1 Reviewed by: __________ Shift Manager/Unit Supervisor _____ Date

This procedure, when completed, SHALL be retained.

Unit:_____ Date:_____ Initial Time:_____ Page__ of __

(CONTINUATION SHEET)

TIME (1)	PRZR VAPOR SPACE (10)			PRZR WATER SPACE			RCS PRESS QDPS (3)	RCS			PRZR-RCS WTR	INITIAL (7)
	TI-0607	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate	TI-0608	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate		QDPS (4)	°F/HR (8) 15 minute rate	°F/HR (5) Rolling Hourly rate	Delta T (6)	

UNIT 1 **(Circle Unit Performing Lineup)** **UNIT 2**

EXCEPTIONS

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	REMARKS

Personnel participating in device manipulation: ______________________ ______________________

Name Initials

______________________ ______________________

______________________ ______________________

______________________ ______________________

Device lineup completed by: ______________ ______________ ______________

Operator Date Time

Lineup 1 Reviewed: ______________________ ______________________

Unit Supervisor Date

This procedure, when completed, SHALL be retained.

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	LOCATION	POSITION REQUIRED	ALIGNED BY	VERIFIED BY	NEW TAG NEEDED
1(2)-RC-0507	RX VESSEL HEAD VENTING MANIFOLD VENT VALVE	RCB On RV Head	CLOSED			
1(2)-RC-0509	RX VESSEL VENTING MANIFOLD DRAIN	RCB On RV Head	** OPEN/ CLOSED			
1(2)-RC-0504	RV HEAD/PZR EQUALIZING LINE DRAIN VLV	RCB 73' Outside On SG 1A(2A) N Shield Wall	CLOSED			
1(2)-RC-0132	RX VESSEL HEAD ATMOSPHERIC VENT VALVE	RCB On RV Head	** OPEN/ CLOSED			
1(2)-RC-0508	RX VESSEL HEAD VENTING MANIFOLD PZR EQUAL LINE ISOL	RCB On RV head	OPEN			
1(2)-RC-0501	RV HEAD/PZR EQUALIZING LINE ISOL VLV	RCB 73' Outside On SG 1A(2A) N Shield Wall	OPEN			
1(2)-RC-0163	PZR SPRAY LINE VENT VALVE	RCB Top of PZR	OPEN			
1(2)-RC-0103	PZR SPRAY LINE VENT VALVE	RCB Top of PZR	OPEN			
1(2)-RC-0506	RX VESSEL HEAD VENTING MANIFOLD PI-3636 ISOL	RCB On Head	* OPEN/ CLOSED			
1(2)-RC-0070	RX VESSEL HEAD VENT ISOL	RCB On RV Head inside shield wall	LOCKED OPEN			

* IF PI-3636 is installed, THEN OPEN "1(2)-RC-0506 RX VESSEL HEAD VENTING MANIFOLD PI-3636 ISOL".

** OPEN IF RCS LEVEL below 47 ft. 4 in.

UNIT 1 (Circle Unit Performing Form) **UNIT 2**

INITIALS

NOTE

- Any component and line **NOT** to be borated to 2800 ppm or greater requires Unit Operations Manager prior permission. (Form 1, Step 3.0)
- The volumes in Steps 2.0 and 3.0 of this form purge five dead leg volumes through the associated component and piping.
- With the RCS borated to greater than or equal to 2875 ppm, CVCS system design will mix the following lines ensuring the concentration will be 2800 ppm or greater prior to entering the RCS piping. (Reference 2.112)

CVCS sections that do **NOT** require prior boration	Mixing volume
1(2)-CV-FCV-0201, CCP 1A(2A) recirc to VCT	Volume Control Tank
1(2)-CV-FCV-0202, CCP 1B(2B) recirc to VCT	Volume Control Tank
1(2)-CV-LCV-3119, Aux Spray line	Pressurizer
1(2)-CV-0255, CVCS charging discharge FCV-0205 bypass	Charging pump discharge header to RCS including Regenerative Heat Exchanger
1(2)-CV-0206, CCP 1B(2B) discharge bypass	Charging pump discharge header to RCS including Regenerative Heat Exchanger

1.0 IF a component will **NOT** be borated, THEN PERMISSION from Unit Operations Manager is required, AND N/A components in Step 2.0 and 3.0 of this addendum **NOT** to be borated. _____ US/SM

2.0 ENSURE the following components borated to 2800 ppm or greater as follows:

- ENSURE CCP 1A(2A) borated with 600 gallons of 2800 ppm or greater of borated water from VCT through CCP 1A(2A) "DISCH ISOL MOV-8377A" per 0POP02-CV-0004. _____
- ENSURE CCP 1B(2B) borated with 600 gallons of 2800 ppm or greater of borated water from VCT through CCP 1B(2B) "DISCH ISOL MOV-8377B" per 0POP02-CV-0004. _____
- ENSURE 1(2)-CV-MOV-0003 and associated lines to RCS borated with 300 gallons of 2800 ppm or greater borated water per Step 7.12 of this procedure. _____
- ENSURE 1(2)-CV-MOV-0006 and associated lines to RCS borated with 300 gallons of 2800 ppm or greater borated water per Step 7.12 of this procedure. _____

3.0 RECORD Time/Date when each component and associated lines were determined to be borated.

- CCP 1A(2A) and associated lines have been inservice with 600 gallons of 2800 ppm or greater borated water flushed. ______/_______
- CCP 1B(2B) and associated lines have been inservice with 600 gallons of 2800 ppm or greater borated water flushed. ______/_______
- 1(2)-CV-MOV-0003 and associated lines to RCS have been inservice with 300 gallons of 2800 ppm or greater borated water flushed through it. ______/_______
- 1(2)-CV-MOV-0006 and associated lines to RCS have been inservice with 300 gallons of 2800 ppm or greater borated water flushed through it. ______/_______

RS1

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC. NO.STPNOC013-CALC-002 REV. 1 PAGE NO. 1 of 49

Title:	Radiological Release Thresholds for Emergency Action Levels	**Client:** South Texas Project **Project:** STPNOC013

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions that require confirmation? (If **YES**, Identify the assumptions) ______		✓
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, Identify the design verified Calculation.) **Design Verified Calculation No.** ______		✓
3	Does this calculation Supersede an existing Calculation? (If **YES**, identify the superseded Calculation.) **Superseded Calculation No.** ______		✓

Scope of Revision: Incorporate decay time of one hour from shutdown as well as migration into Attachment 1. Change statement of no decay in the STAMPEDE runs.

Revision Impact on Results: Values calculated in Attachment 1 decreased and have become the limiting values.

~~**Study Calculation**~~ ☐ ~~**Final Calculation**~~ ☑

Safety-Related ☑ **Non-Safety Related** ☐

(Print Name and Sign)

Originator:	Caleb Trainor	**Date:** 3/21/2014
Design Verifier:	Chad Cramer	**Date:** 3/21/14
Approver:	Marvin Morris	**Date:** 3/21/14

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	03/03/2014	Initial Issue
1	3/21/2014	Resolve inconsistency in decay times for the two calculations

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
1-11	1		

ATTACHMENT REVISION STATUS

ATTACHMENT NO.	PAGE NO.	REVISION NO.	ATTACHMENT NO.	PAGE NO.	REVISION NO.
1	12-24	1			
2	25-31	1			
3	32-49	1			

ENERCON Excellence—Every project. Every day.	CALCULATION DESIGN VERIFICATION CHECKLIST	CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 3 of 49

Item	CHECKLIST ITEMS	Yes	No	N/A
1	**Design Inputs** - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis and incorporated in the calculation?	✓		
2	**Assumptions** – Were the assumptions reasonable and adequately described, justified and/or verified, and documented?	✓		
3	**Quality Assurance** – Were the appropriate QA classification and requirements assigned to the calculation?	✓		
4	**Codes, Standard and Regulatory Requirements** – Were the applicable codes, standards and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied?	✓		
5	**Construction and Operating Experience** – Have applicable construction and operating experience been considered?			✓
6	**Interfaces** – Have the design interface requirements been satisfied, including interactions with other calculations?	✓		
7	**Methods** – Was the calculation methodology appropriate and properly applied to satisfy the calculation objective?	✓		
8	**Design Outputs** – Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives and are the results reasonable compared to the inputs?	✓		
9	**Radiation Exposure** – Has the calculation properly considered radiation exposure to the public and plant personnel?	✓		
10	**Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	✓		
11	**Computer Software** – Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?			✓

COMMENTS:

None

(Print Name and Sign)

Design Verifier:	Chad Cramer	**Date:** 3/21/14
Others:		**Date:**

Calculation Design Verification Plan:

Calculation shall be verified by comparing the documented input with the references and checking the validity of the references for the intended use. As necessary, assumptions shall be evaluated and verified to determine if they are based on sound engineering principles and practices. Verify the applicable methodology, inputs, results, and conclusions.

(Print Name and Sign for Approval – mark "N/A" if not required)

Approver: Marvin Morris *Marvin Morris*	**Date** 3/21/14

Calculation Design Verification Summary:

Design inputs, assumptions, methodology, results and conclusions were evaluated/verified and found to be acceptable. All comments have been incorporated.

Based On The Above Summary, The Calculation Is Determined To Be Acceptable.

(Print Name and Sign)

Design Verifier: Chad Cramer [signature]	**Date:** 3/21/14
Others:	**Date:**

Table of Contents

1.0 OBJECTIVE/SCOPE

The purpose of this calculation is to determine the Emergency Action Level (EAL) threshold values of a radiological release from the Unit Vent or Main Steam Lines for an Unusual Event, Alert, Site Area Emergency, or General Emergency. The calculated threshold values are to be included in the STP EAL Technical Basis document, which implements the new NEI 99-01, Revision 6, Emergency Action Level Scheme and will be submitted to the NRC for approval. Upon NRC approval, the values will be used in 0ERP01-ZV-IN01, Revision 10, Emergency Classification.

Both a hand calculation and the South Texas Assessment Model Projecting Emergency Dose Evaluation (STAMPEDE) software program were used to generate the results. The hand calculation is included as Attachment 1.

Revision 1 of this calculation incorporated decay for a release taking place one hour after reactor shutdown. This was done to create continuity between the two methodologies present.

2.0 SUMMARY OF RESULTS

The results of the calculations for the radiation monitors specified in the STP EAL Basis Document and are listed in Table 2.1, below.

Table 2.1: Summary of Calculation Results

Emergency Action Level		RT-8010B, Unit Vent (µCi/sec)	RT-8046 through 8049, Main Steam Lines (µCi/cc)
RU1	Unusual Event		
	Hand Calculation	1.40E+05	5.00E-02
	STAMPEDE	N/A*	N/A*
RA1	Alert		
	Hand Calculation	1.57E+06	4.10 E+00
	STAMPEDE	2.50E+06	4.50E+00
RS1	Site Area Emergency		
	Hand Calculation	1.57E+07	4.105E+01
	STAMPEDE	2.50E+07	4.50E+01
RG1	General Emergency		
	Hand Calculation	1.57E+08	4.10E+02
	STAMPEDE	2.50E+08	4.50E+02

*STAMPEDE was not used to determine the threshold for RU1. Reference 5.10 indicates that the ODCM methodology should be used to determine the threshold value.

This calculation will be used to establish the threshold values for abnormal radiation based emergencies in the STP EAL Technical Basis document.

3.0 METHOD OF ANALYSIS

Previously, STAMPEDE was used to calculate the Emergency Action Level threshold values for effluent releases. A hand calculation will verify the STAMPEDE calculations. The hand calculation is described in Attachment 1 of this document STAMPEDE conforms to the requirements of STP Procedure 0PGP07-ZA-0014, Software Quality Assurance Program. STAMPEDE was run at STP on an STP computer and under the supervision of an ENERCON employee with access to the STP site as a critical worker.

4.0 INPUTS

4.1 Per NEI 99-01, Revision 6, Initiating condition AU1, EAL 1, the Notice of Unusual Event initiating condition is a release of gaseous or liquid radioactivity greater than two times the ODCM limit for sixty minutes or longer (Reference 5.10).

4.2 The ODCM offsite dose limit is exceeded if the Xe-133 release concentration exceeds 7.41E-04 μCi/cc (Reference 5.6).

4.3 The Unit Vent flow rate is 9.4E+07 cc/sec (Reference 5.1).

4.4 The main steam line pressure and PORV choke flow rate are 1285 psig and 1.05E+06 lbm/hr, respectively (Reference 5.2).

~~4.5 The specific volume of saturated steam at 1285 psig is 0.338 ft^3/lbm (Reference 5.3).~~

4.6 The release concentration is varied to find the release concentration which correlates to each emergency action level. Emergency action levels are taken from NEI 99-01, Revision 6 (Reference 5.10) for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The doses at the Site Boundary that correlate to the threshold concentrations are listed in Table 4.1.

Table 4.1 EAL Offsite Dose Initiating Conditions

	Alert	Site Area	General
TEDE	10 mrem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

5.0 REFERENCES

~~5.1 Offsite Dose Calculation Manual, Revision 17, March 2011~~
5.2 Main Steam PORV Capacity Verification MC05591, Revision: 1
5.3 NIST Steam Tables, 2011
5.4 0ERP01-ZV-IN01, Emergency Classification Draft Revision 10
5.5 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21
5.6 STP Calculation NC-9012, CRMS Rad Monitor Setpoints, Revision 7
5.7 STP Calculation NC-9011, Revision 2
5.8 STAMPEDE Computer Program, Revision 7.0.3.3
5.9 STAMPEDE User's Manual
5.10 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
5.11 0PGP07-ZA-0014 Quality Assurance Program
5.12 ITWMS Call Number 1000010987 Design Document, Revision 0

6.0 ASSUMPTIONS

6.1 Unit Vent Noble Gas Monitor

To be consistent with the ODCM methodology, the unit vent release is assumed to be entirely Xe-133. The unit vent noble gas monitor is calibrated to Xe-133 (Reference 5.1) therefore; the monitor reading accurately reflects the Xe-133 release magnitude.

To be consistent with ODCM methodology, the main steam line release is assumed to be entirely Xe-133. The noble gas monitor is calibrated to Xe-133 (Reference 5.6).

6.2 Release Duration

Per Reference 5.10, Sections IC AA1, AS1, and AG1 developer notes, the release should be assumed to last one hour.

6.3 Release following Reactor Shutdown

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

6.4 Source Term

Per Reference 5.1, any unit vent release with increased RCS activity and no core melt should be calculated using the gap inventory. Therefore, the gap inventory is used for all unit vent releases.

Per Reference 5.1, for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of noble gases plus 0.2% iodine. A steam generator tube rupture is the only scenario which would create significant offsite doses through a main steam line release.

6.5 Default STAMPEDE Input Values

Reference 5.10 developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. STAMPEDE is used for emergency dose assessment. Per Reference 5.1, when actual meteorology is not available, the default STAMPEDE values should be used. Had the ODCM methodology been used, the 500 hour peak χ/Q value would be used which is less conservative than the χ/Q value produced by STAMPEDE using default meteorological conditions. Therefore, the use of STAMPEDE default values provides a more conservative estimate than that of the alternative method outlined in Reference 5.10.

6.6 Average Effluent Concentration (χ/Q)

The same χ/Q is used for the unit vent and main steam line release. Reference 5.1 applies the same unit vent χ/Q to Units 1 and 2 which would also be applicable to the main steam line. All releases are considered to be ground level releases.

7.0 STAMPEDE CALCULATIONS

7.1 Unusual Event – RU1

7.1.1 Unit Vent Monitor

AU1 recommends declaring an unusual event due to a release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Reference 5.10).

STP sets the ODCM limit at 7.41E-04 μCi/cc (Reference 5.6, pg. 16). Two times the limit would be 1.48E-03 μCi/cc. The threshold is listed in μCi/sec so that variations in flow rate do not change the threshold. The normal flow rate from the unit vent is 9.4E+07 cc/sec (Reference 5.1).

$$Concentration\left(\frac{\mu Ci}{cc}\right) * Flow\ Rate\left(\frac{cc}{sec}\right) = Release\ Rate\left(\frac{\mu Ci}{sec}\right)$$

$$(1.48E-03)\left(\frac{\mu Ci}{cc}\right) * (9.4\text{E}+07)\left(\frac{cc}{sec}\right) = 1.4E+05\left(\frac{\mu Ci}{sec}\right)$$

Equation 7.1.1.1

7.1.2 Main Steam Line Monitor

The ODCM does not calculate a release corresponding to allowable limits for the main steam line monitors. Since the unit vent release calculated in the ODCM was assumed to be primarily Xe-133, the assumption is made in the ODCM that other noble gases and iodine may be ignored in the calculation. This assumption is equally justifiable for the main steam line and the same limiting release will be used.

The magnitude of the release calculated for the unit vent Unusual Event applies to the main steam lines as well. The main steam line PORV's will create a dose exceeding two times the ODCM limit by releasing 1.4E+05 μCi/sec of activity which is equivalent to the release from the unit vent.

The steam lines hold saturated steam at 1285 psig, per Reference 5.2, which has a specific volume of 0.338 ft^3/lbm (Reference 5.3). The PORVs will release the steam at 1.05E+06 lbm/hr per Reference 5.2. This creates a set flow rate of steam from the main steam lines of 2.79E+06 cc/sec as shown below.

$$F\left(\frac{lbm}{hr}\right) * Density\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = \frac{cc}{sec}$$

$$1.05E+06\left(\frac{lbm}{hr}\right) * 0.338\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = 2.79E+06\frac{cc}{sec}$$

Equation 7.1.2.1

Since the flow rate is set, the concentration will determine the limit. Equation 7.1.1.1 solves for the limiting concentration of 5.00E-02 μCi/cc as shown below.

$$\frac{Limiting\ Release\left(\frac{\mu Ci}{sec}\right)}{Release\ Rate\left(\frac{cc}{sec}\right)} = Limiting\ Concentration\left(\frac{\mu Ci}{cc}\right)$$

$$\frac{1.40 * 10^5\left(\frac{\mu Ci}{sec}\right)}{2.79 * 10^6\left(\frac{cc}{sec}\right)} = 5.00E - 02\left(\frac{\mu Ci}{cc}\right)$$

Equation 7.1.2.2

7.2 Alert, Site Area and General Emergencies – RA1, RS1, RG1

7.2.1 Unit Vent Monitor

Input

The Alert EAL is set to 10 mrem TEDE and 50 mrem Thyroid CDE per Reference 5.10. The emergency offsite dose calculation software STAMPEDE was used to calculate the release which corresponds to this dose. A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for the calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Gap inventory source term
- Default STAMPEDE input values
  - Windspeed = 13.2 mph
  - Stability class D

Results

Given a monitored unit vent release of 2.50E+06 μCi/sec, the Thyroid CDE is 51 mrem/hr at the closest portion of the site boundary and the EAL Initiating Condition is exceeded.

Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 2.50E+07 and 2.50E+08 μCi/sec for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

The input and output files can be found at the end of this document in Attachment 3.

7.2.2 Main Steam Line Monitor

Input

A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for this calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Noble gas + iodine with 0.2% iodine source term
- Default STAMPEDE input values
  - Windspeed = 13.2 mph
  - Stability class D

Results

Given a monitored main steam line release of 4.5 µCi/cc, the Thyroid CDE is 50 mrem/hr and the EAL Initiating Condition is exceeded.

The input and output files can be found at the end of this document in Attachment 3.

7.3 Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 45 and 450 µCi/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

Attachment 1 – Hand Calculations

1.0 OBJECTIVE/SCOPE

Each release calculated using STAMPEDE in the main document is calculated by hand in this attachment and the results compared to STAMPEDE.

2.0 SUMMARY OF RESULTS

Table 2.1 is displayed again below showing the results from all the calculations. The minor difference is due to STAMPEDE using decay factors over a one hour period after shutdown. This also accounts for the change in the limiting dose being TEDE in the hand calculations and Thyroid CDE in the STAMPEDE calculations. The accuracy of the hand calculation is considered sufficient and recommended for use in Emergency Action Levels.

Table 2.1 Results

Emergency Action Level		RT-8010b, Unit Vent (µCi/sec)	RT-8046 through 8049, Main Steam Line (µCi/cc)
RU1	**Unusual Event**		
	Hand Calculation	1.40E+05	5.00E-02
	STAMPEDE	N/A	N/A
RA1	**Alert**		
	Hand Calculation	1.57E+06	3.90E+00
	STAMPEDE	2.50E+06	4.50E+00
RS1	**Site Area Emergency**		
	Hand Calculation	1.57E+07	3.90E+01
	STAMPEDE	2.50E+07	4.50E+01
RG1	**General Emergency**		
	Hand Calculation	1.57E+08	3.90E+02
	STAMPEDE	2.50E+08	4.50E+02

3.0 METHOD OF ANALYSIS

Using the limiting dose at the site boundary, the release is back calculated using atmospheric dispersion models. The X/Q value used is calculated from Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power* Plants. Rather than using the most conservative meteorology, average meteorological conditions are used as inputs to most closely agree with STP emergency dose assessment methodology per the ODCM and STAMPEDE. Assumed nuclide inventories are taken from Reference 5.4. The dose conversion factors are taken from Reference 5.2. A release concentration is used to find an initial projected dose at the Site Boundary. Using the projected dose at the Site Boundary, the release concentration is scaled to find the limiting dose for each EAL.

4.0 INPUTS

- The Unit Vent flow rate is taken from the Offsite Dose Calculation Manual; Revision 17, March 2011 and is 9.44E+07 cc/sec.
- The main steam line pressure and PORV choke flow rate were taken from Reference 5.5 and are 1285 psig and 1.05E+06 lbm/hr respectively.
- The specific volume of saturated steam at this pressure is taken from the NIST steam tables and is 0.338 ft^3/lbm.
- The release concentration is varied to find the release concentration which correlates to each emergency action level dose. Emergency action level doses are taken from NEI 99-01 Revision 6 for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The limiting doses are listed in Table 4.1. NEI 990-01 Revision 6 states that these values are based on fractions of the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) and the General Emergency represents the protective action values recommended by the EPA.

Table 4.1 EAL Thresholds

	Alert	Site Area	General
TEDE	10 mrem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

- A release lasting one hour is selected per NEI 99-01 Revision 6 developer notes.
- Atmospheric dispersion factors are calculated per Regulatory Guide 1.145 (Reference 5.1). The reactor building dimensions used as inputs for this calculation are taken from Reference 5.13.
- Nuclide inventories are taken from TGX/THX 3-1, (Reference 5.4) which is the source document for the nuclide inventories used in STAMPEDE. The release inventories are a gap release and noble gases plus 0.2% iodine which are listed below. Each nuclide inventory was normalized to one so it could be scaled to various release activities.

Table 4.2 Gap Inventory

Nuclide	Activity (µCi/cc)	Normalized	Nuclide	Activity (µCi/cc)	Normalized
I-131	1.10E+05	1.12E-03	Xe-135	5.50E+06	5.62E-02
I-132	1.50E+05	1.53E-03	Xe-137	1.90E+07	1.94E-01
I-133	2.20E+05	2.25E-03	Xe-138	1.80E+07	1.84E-01
I-134	2.40E+05	2.45E-03	Cs-134	3.70E+01	3.78E-07
I-135	2.00E+05	2.05E-03	Cs-137	2.90E+01	2.97E-07
Kr-83m	1.30E+06	1.33E-02	Te132	4.80E+00	4.91E-08
Kr-85m	2.90E+06	2.97E-02	Mo99	1.22E+01	1.25E-07
Kr-85	3.70E+05	3.78E-03	Ru103	8.80E-03	9.00E-11
Kr-87	5.50E+06	5.62E-02	Ru106	2.90E-03	2.97E-11
Kr-88	7.80E+06	7.98E-02	Zr95	1.10E-02	1.12E-10
Kr-89	9.50E+06	9.72E-02	La140	1.90E-02	1.94E-10
Xe-131m	1.10E+05	1.12E-03	Ce144	7.40E-03	7.57E-11
Xe-133m	6.80E+05	6.95E-03	Ce-141	1.00E-02	1.02E-10
Xe-133	2.20E+07	2.25E-01	Sr89	6.40E-02	6.55E-10
Xe-135m	4.20E+06	4.30E-02	Sr90	3.20E-03	3.27E-11

Table 4.3 Noble Gases+0.2% Iodine Inventory

Nuclide	Inventory	Normalized
I-131	6.10E-02	2.26E-04
I-132	8.61E-02	3.19E-04
I-133	1.00E-01	3.72E-04
I-134	1.86E-02	6.92E-05
I-135	2.73E-01	1.01E-03
Xe-131m	2.80E+00	1.04E-02
Xe-133	2.40E+02	8.90E-01
Xe-133m	4.20E+00	1.56E-02
Xe-135	7.60E+00	2.82E-02
Xe-135m	4.00E-01	1.48E-03
Xe-137	1.60E-01	5.93E-04
Xe-138	5.80E-01	2.15E-03
Kr-83m	3.70E-01	1.37E-03
Kr-85	7.60E+00	2.82E-02
Kr-85m	1.50E+00	5.56E-03
Kr-87	9.80E-01	3.63E-03
Kr-88	2.80E+00	1.04E-02
Kr-89	8.40E-02	3.12E-04

- The dose conversion factors taken from EPA 400R92001 (Reference 5.2) are listed in Tables 4.4 and 4.5 below.

Table 4.4 TEDE Dose Conversion Factors

Nuclide	Dose Conversion Factor (rem per uCi*hr/cc)	Nuclide	Dose Conversion Factor (rem per uCi*hr/cc)
I-131	5.30E+04	Xe-135	1.40E+02
I-132	4.90E+03	Xe-137	1.10E+02
I-133	1.50E+04	Xe-138	7.20E+02
I-134	3.10E+03	Cs-134	6.30E+04
I-135	8.10E+03	Cs-137	4.10E+04
Kr-83m		Te132	1.20E+04
Kr-85m	9.30E+01	Mo99	5.20E+03
Kr-85	1.30E+00	Ru103	1.30E+04
Kr-87	5.10E+02	Ru106	5.70E+05
Kr-88	1.30E+03	Zr95	3.20E+04
Kr-89	1.20E+03	La140	1.10E+04
Xe-131m	4.9	Ce144	4.50E+05
Xe-133m	1.70E+01	Ce-141	1.10E+04
Xe-133	2.00E+01	Sr89	5.00E+04
Xe-135m	2.50E+02	Sr90	1.60E+06

Table 4.5 Thyroid CDE Dose Conversion Factors

Nuclide	Thyroid CDE DCF (rem per uCi*hr/cc)
I-131	1.30E+06
I-132	7.70E+03
I-133	2.20E+05
I-134	1.30E+03
I-135	3.80E+04

- The unit vent noble gas monitor energy efficiency by nuclide is taken from Offsite Dose Calculation Manual (Reference 5.3). The values are relative to Xe-133 efficiency since the monitor is calibrated to Xe-133. Table 4.6 displays the energy efficiency by nuclide relative to Xe-133.

Table 4.6 Energy Efficiency Relative to Xe-133

Nuclide	Efficiency Relative to Xe-133 $(uCi/cc)_{equivalent}$
Kr-83m	*
Kr-85m	1.9
Kr-85	2.4
Kr-87	2.8
Kr-88	2.3
Kr-89	2.8
Xe-131m	0.015
Xe-133m	0.14
Xe-133	1
Xe-135m	0.042
Xe-135	2.5
Xe-137	2.8
Xe-138	2.8

*There is no relative efficiency available for Kr-83m. Assumption 6.4 further justifies the omission.

Table 4.7 Nuclide Half Lives

Nuclide	Half Life (hr)	Nuclide	Half Life (hr)
I-131	1.93E+02	Xe-135	9.08E+00
I-132	2.38E+00	Xe-137	6.38E-02
I-133	2.03E+01	Xe-138	2.36E-01
I-134	8.77E-01	Cs-134	1.80E+04
I-135	6.61E+00	Cs-137	2.60E+05
Kr-83m	1.83E+00	Te132	7.79E+01
Kr-85m	4.48E+00	Mo99	6.62E+01
Kr-85	9.40E+04	Ru103	9.44E+02
Kr-87	1.27E+00	Ru106	8.84E+03
Kr-88	2.84E+00	Zr95	1.55E+03
Kr-89	5.10E-02	La140	4.03E+01
Xe-131m	2.83E+02	Ce144	6.82E+03
Xe-133m	5.42E+01	Ce-141	7.77E+02
Xe-133	1.27E+02	Sr89	1.21E+03
Xe-135m	2.60E-01	Sr90	2.50E+05

- The half-lives are taken from Reference 5.15 which lists the input data used by STAMPEDE.

5.0 REFERENCES

5.1 Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.
5.2 EPA 400R92001, Manual of Protective Action Guides and Protective actions for Nuclear Incidents, Revision 1, May 1992.
5.3 Offsite Dose Calculation Manual, Revision 17, March 2011.
5.4 TGX/THX 3-1, Revision 5, Westinghouse Radiation Analysis Manual.
5.5 MC05591, Main Steam PORV Capacity Verification, Revision 1.
5.6 NIST Steam Tables, 2011.
5.7 0ERP01-ZV-IN01, Emergency Classification, Revision 10.
5.8 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21.
5.9 STP Calculation NC-9012, Process and Effluent Radiation Monitor Set Points, Revision 7
5.10 STP Calculation NC-9011, CRMS Rad Monitor Setpoints, Revision 2.
5.11 STAMPEDE Computer Program, Revision 7.0.3.3.
5.12 STAMPEDE User's Manual
5.13 STP Drawing 6C189N5007, General Arrangement Reactor Containment Building, Revision 6
5.14 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
5.15 ITWMS Call Number 1000010987 Design Document, Revision 0

6.0 ASSUMPTIONS

6.1 Release lasts for one hour

Per NEI 99-01 (Reference 5.14), IC AA1, AS1, AG1 developer notes, the release should be assumed to last one hour.

For this to be true for the main steam line, it is assumed that the PORV is open for one hour. To calculate the most limiting case, it is assumed that the maximum flow possible is being released from the PORV.

6.2 Nuclide mix

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) any unit vent release with increased RCS activity and no core melt should be calculated using a gap inventory. It is conservative to assume an increased RCS activity and not within the intended scope of the relevant initiating conditions to assume core melt. Therefore, a gap inventory is used for all unit vent releases.

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of 100 percent noble gases plus 0.2 percent iodine. Since a steam generator tube rupture releasing through the PORVs is the only steam generator tube rupture scenario which would create offsite doses large enough to meet or exceed the EALs, this assumption is made.

6.3 Atmospheric Dispersion

NEI 99-01 (Reference 5.14) developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8), when actual meteorology is not available, the default STAMPEDE values should be used. The default STAMPEDE values assume a stability class D for atmospheric dispersion and a windspeed of 13.2 mph. These values were used as inputs for the atmospheric dispersion calculation.

It is clear that STAMPEDE uses the same method for calculating atmospheric dispersion factor (X/Q) outlined in section 7.1.1 of this Attachment. However, STAMPEDE does not follow the same logic in selecting the appropriate result from the three calculations. The STAMPEDE value printed in the results found in attachment 3 is consistent with the largest of the three hand calculated X/Q values. This suggests that STAMPEDE simply selects the largest of the three X/Q values resulting in a much more conservative estimate. This calculation will deviate from the recommendations of Regulatory Guide 1.145 and conform to the methodology STAMPEDE uses.

The close proximity of all release points allows for a single atmospheric dispersion coefficient to be used. This assumption is also made by STAMPEDE.

6.4 Exposure Pathways

The dose conversion factors used in table 4.4 and 4.5 represent a summation of dose conversion factors for external plume exposure, inhalation from the plume, and external exposure from deposition. Because the dose estimations are used for implementing early phase protective actions, conversion factors using limited pathways are appropriate.

The EPA does not provide a dose conversion factor for Kr-83m. Because the PAGs are based on EPA dose calculations, it is appropriate to only use the nuclides for which dose conversion factors are provided. Additionally, Kr-83m represents only 1.33% of the nuclide inventory activity and its exclusion would not significantly affect the final dose.

6.5 The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A ~~release at reactor shutdown would have a significantly higher activity at the monitor location than~~ at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

Decay is incorporated for one hour from reactor shutdown as well as migration time. Half-lives are taken from Reference 5.15. Migration time is assumed to be the reciprocal of the wind speed.

7.0 HAND CALCULATIONS

7.1 Unit Vent Monitor

7.1.1 X/Q

The atmospheric dispersion factor, X/Q, determines the change in concentration between the unit vent discharge and the dose reception site. This value is based on meteorological conditions and will vary with wind speed and stability class. The ODCM uses the highest annual average X/Q value at the site boundary which is 5.3E-06 sec/m^3. However, for an accident related release STAMPEDE is used rather than the ODCM. STAMPEDE uses real time, user entered, or default meteorological conditions to calculate the X/Q for a specific accident. Default values will be used as inputs into the Regulatory Guide 1.145 method for calculating X/Q as described below. Default values are identified in section 6.0, Atmospheric Dispersion.

For a neutral atmospheric stability class, which is the default in STAMPEDE, X/Q values can be determined through the following set of equations.

$$\frac{X}{Q} = \frac{1}{\bar{U}_{10}\left(\pi\sigma_y\sigma_z + \frac{A}{2}\right)}$$

Equation 7.1.1.1

$$\frac{X}{Q} = \frac{1}{\bar{U}_{10}(3\pi\sigma_y\sigma_z)}$$

Equation 7.1.1.2

$$\frac{X}{Q} = \frac{1}{\bar{U}_{10}\pi\Sigma_y\sigma_z}$$

Equation 7.1.1.3

Where

X/Q = relative concentration (sec/m^3)

π = 3.14159

$\bar{U}_{10}$ = windspeed at 10 meters above plant grade (m/s)

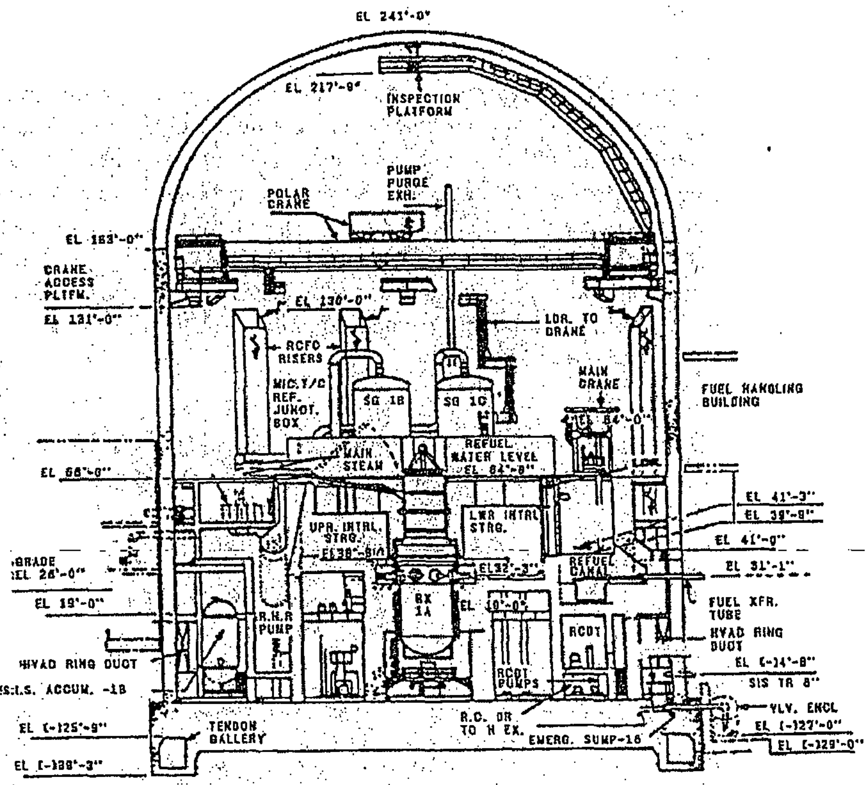
σ_y = lateral plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 1

σ_z = vertical plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 2

Σ_y = $(M-1)\sigma_{y800m} + \sigma_y$ = lateral plume spread with meander and building wake effects (m), a function of atmospheric stability, windspeed $\bar{U}_{10}$, and distance; M is determined from Regulatory Guide 1.145 Figure 3

A = the smallest vertical-plane cross-sectional area of the reactor building (m^2), taken from Reference 5.13 and shown below

Figure 7.1.1.1: Reactor Building Dimensions



Assuming the reactor building cross section to be a perfect rectangle and half sphere, the variables are defined as follows;

$\overline{U}_{10} = 13.2 \text{ mph} = 5.9 \text{ m/s}$

$\sigma_y \ = 1200 \text{ m}$

$\sigma_z \ = 4.2 \text{ m}$

$\Sigma_y \ = (M-1)\sigma_{y800m} + \sigma_y$; M=1 $\rightarrow \sigma_y = 1200$ m

$\text{A} \ = (135' * 158') + \left(\frac{\pi * 79^2}{2}\right) = 31128.37$

The three equations become;

$$\frac{X}{Q} = \frac{1}{5.9\left(\pi 1200 * 4.2 * \frac{31128.37}{2}\right)} = 5.398 * 10^{-6}$$

$$\frac{X}{Q} = \frac{1}{5.9(3\pi * 1200 * 4.2)} = 3.568 * 10^{-6}$$

$$\frac{X}{Q} = \frac{1}{5.9 * \pi * [(1-1)\sigma_{y800m} + 1200] * 4.2} = 1.07 * 10^{-5}$$

To select the appropriate X/Q value, the first two X/Q values should be compared and the higher value selected. This value is then compared with the third X/Q value and the lower of those two is the appropriate X/Q value. The appropriate X/Q is 5.39E-06 sec/m^3 for default meteorological conditions by the methodology recommended in Regulatory Guide 1.145.

This calculated value is very similar to the ODCM highest average value of 5.3E-06 sec/m^3 which was not selected for use. Additionally, the value shown in the STAMPEDE output file at one mile is 1.032E-05 sec/m^3. This suggests that STAMPEDE uses the same methodology and simply selects the largest atmospheric dispersion value to remain conservative. This methodology will be replicated and 1.07E-05 will be used as the X/Q.

7.1.2 Nuclide Inventory

As previously stated, a gap inventory is appropriate for this problem. The gap inventory is taken from TGX/THX 3-1 (Reference 5.4) which is used as the source term for STAMPEDE inventories. The concentrations were then normalized so they could be scaled to the varying emergency classifications. The values for the normalized inventory can be found in Table 4.2.

7.1.3 Dose Conversion Factors

As stated in NEI99-01 (Reference 5.14) developer notes, the purpose of dose projections is to check if the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) have been exceeded. The dose conversion factors provided by the EPA in EPA 400R92001 are used. These dose conversion factors account for external plume exposure, inhalation from the plume, and external exposure from deposition and are listed Tables 4.4 and 4.5, and taken from tables 5-1, 5-2 in EPA 400R92001 (Reference 5.2).

The EPA does not provide a dose conversion factor for Kr-83m. This nuclide contributes 1.33% of the inventory activity. The lack of this nuclide's contribution to the final dose will not significantly affect the outcome.

7.1.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.1.5 Dose Calculations

The dose rate at the site boundary is calculated using Equation 7.1.5.1.

$$\dot{D} = \frac{X}{Q} F \sum_{i}^{n} C_i * 0.5^{\frac{1.07575}{T_{1/2i}}} * DCF_i$$

Equation 7.1.5.1

Where

$\dot{D}$ = dose rate per hour at the site boundary

$\frac{X}{Q}$ = atmospheric dispersion coefficient as calculated in section 7.1.1

F = unit vent flow rate

C_i= concentration of nuclide i at the time of shutdown

1.07575 = the total decay time of interest from section 7.1.4

$T_{1/2i}$ = the half-life of nuclide i

DCF_i = the dose conversion factor for nuclide i listed in tables 4.4 and 4.5

The total concentration of the nuclides is varied to find the dose rate of interest. Beginning with an arbitrary release concentration of 1 µCi/cc, the dose rate is calculated. Since the dose is linearly correlated to concentration, the release concentration may be scaled to find the dose rate of interest.

The Alert EAL is 10 mrem TEDE or 50 mrem Thyroid CDE. Using the above method to calculate TEDE with the appropriate conversion factors, a limiting release rate of 2.33E+06 µCi/sec from the unit vent results in 5.7 mrem TEDE. Using the calculated release rate to find Thyroid CDE with the appropriate conversion factors, the same release results in a 50 mrem Thyroid CDE at the site boundary. Thus, 2.33E+06 µCi/sec is the limiting release rate based on the 50 mrem Thyroid CDE EAL initiating condition.

The limiting release rate threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert release rate threshold value.

These calculations can be found in Attachment 2.

7.1.6 Monitor Response

The unit vent noble gas monitor is calibrated to Xe-133. Monitor efficiencies relative to Xe-133 by nuclide are listed in ODCM Table B3-2. To find the monitor reading associated with each limiting release, the noble gas concentrations must be multiplied by the monitor response and summed. Table 4.6 shows the indicated response of the unit vent noble gas monitor by nuclide and Equation 7.1.5.1 shows how the monitor response was calculated.

$$Monitor\ Response = \sum_{i}^{n} C_i * Re_i$$

Equation 7.1.5.1

Where

C_i = concentration of nuclide i (µCi/cc)

Re_i = monitor response to nuclide i $(\mu Ci/cc)_{Xe\text{-}133\ equivalent}$

In the case of an Alert, the 2.33E+06 µCi/sec release rate will read as 1.57E+06 µCi/sec on the monitor. Kr-83m does not have an indicated monitor response coefficient. Because Kr-83m is only 1.34% of the noble gases and does not contribute to the dose calculation, its exclusion is acceptable.

This again is a linear correlation and the SAE and GE scale by factors of 10 and 100 respectively.

These calculations can be found in Attachment 2.

7.2 Main Steam Line Monitors

7.2.1 X/Q

Since the atmospheric dispersion is independent of nuclide inventory or release rate and the close proximity of the releases, the X/Q value will be the same for a main steam line release as it is for a unit vent release. This assumption is also taken by STAMPEDE and outline in Assumption 6.3.

7.2.2 Nuclide Inventory

Per 0ERP01-ZV-TP01, if the release path is the main steam line with a steam generator tube rupture, the nuclide inventory should be 100% noble gas and 0.2% of the iodine from the reactor coolant.

The secondary steam concentration for noble gases and iodine after a steam generator tube rupture are taken from TGX/THX 3-1 (Reference 5.4). Values for the reactor coolant inventory are listed in table 4.3. All of the noble gases are used and the iodine concentration from the coolant inventory is scaled to total 0.2% of iodine in the total coolant inventory. These inventories are then normalized to one. These values are listed in Table 4.3.

7.2.3 Dose Conversion Factors

The dose conversion factors used are found in Tables 4.4 and 4.5, taken from tables 5-1, 5-2 in EPA 400R92001.

7.2.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.2.5 Dose Calculations

Equation 7.1.5.1 applies to the release from the main steam lines. The main steam line flow rate is used instead of the unit vent flow rate for the value F. The main steam line flow rate was calculated in Equation 7.1.2.2 of the STAMPEDE CALCULATIONS section of this document as 2.79E+06 cc/sec.

The Alert EAL threshold is 10 mrem TEDE or 50 mrem Thyroid CDE at the site boundary (Table 4.2). Using the method in Equation 7.1.5.1 to calculate TEDE with the appropriate conversion factors, a concentration at time of shutdown of 4.10 µCi/cc would result in 6.89 mrem TEDE at the site boundary if the steam line PORV was open for an hour. Using the same steam line concentration to calculate Thyroid CDE results in 50 mrem Thyroid CDE at the site boundary.

The steam line concentrations at the time of shutdown for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, values for the steam line concentration at time of shutdown are 41.0 and 410 µCi/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE.

These calculations can be found in Attachment 2.

7.2.6 Monitor Response

Because the main steam line monitor is adjacent to the main steam line, significant shielding takes place between the source and monitor. STP calculation NC-9011 Revision 2 calculates a conversion factor for the main steam lines for a noble gas inventory which is incorporated into the monitor readout. No monitor response needs to be calculated.

The concentration of the main steam line one hour after shutdown given a concentration of 4.10 µCi/cc at the time of shutdown is 3.90 µCi/cc. This calculation is also found in Attachment 2. Additionally, the monitor readings for the SAE and GE one hour after shutdown are 39.0 and 390 µCi/cc respectively. These values are the thresholds for the main steam line monitor.

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:24 **User Name:** Unit Vent Alert

Comments:

User Supplied Information

Meteorological Data Inputs:

- Ground level wind velocity: 13.2 mi/hr
- Ground level wind from: 180 degrees
- User-selected Stability Class
- Stability Class: "D - Neutral"

Monitored Unit Vent Release:

Unit Vent Release Rate entered: 2.50E+006 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:24

Release Start Date/Time: 12/17/2013 15:24

Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+006 uCi/sec

NOBLE GAS

Nuclide	uCi/sec
Kr-83M:	2.53E+004
Kr-85:	1.05E+004
Kr-85M:	7.06E+004
Kr-87:	9.03E+004
Kr-88:	1.74E+005
Kr-89:	3.06E-001
Xe-131M:	3.12E+003
Xe-133:	6.22E+005
Xe-133M:	1.91E+004
Xe-135:	1.45E+005
Xe-135M:	8.14E+003
Xe-137:	9.53E+000
Xe-138:	2.66E+004

IODINE

Nuclide	uCi/sec
I-131:	3.12E+003
I-132:	3.18E+003
I-133:	6.05E+003
I-134:	3.08E+003
I-135:	5.12E+003

PARTICULATE

Nuclide	uCi/sec
Cs-134:	1.05E+000
Cs-137:	8.25E-001
Ce/Pr-144:	2.10E-004
Ce-141:	2.84E-004
La-140:	5.31E-004
Mo-99:	3.43E-001
Ru/Rh-106:	8.25E-005
Ru-103:	2.50E-004
Sr/Y-90:	9.10E-005
Sr-89:	1.82E-003
Te-132:	1.35E-001
Zr-95:	3.13E-004

12/17/2013 3:24:46 PM

Table A2-1: Unusual Event Emergency Calculations

Unit Vent Limiting Concentration (uCi/cc)	Flow Rate (cc/sec)	Release Rate (uCi/sec)
1.48E-03	9.44E+07	**1.40E+05**
MSL Limiting Release Rate (uCi/sec)	**Flow Rate (cc/sec)**	**Limiting Concentration (uCi/cc)**
1.40E+05	2.79E+06	**5.00E-02**

Table A2-2: Input Values for Calculations

X/Q (s/m^3)	duration (s)	Release Rate (cc/sec)	Release Constant (s*cc/m^3)	Unit Conversion for Release Constant (hr*m^3/s*cc)	Total Release Variable (uCi/cc)	Decay Time (hr)
5.40E-06	3600	9.44E+07	1.83E+06	5.10E-04	1.79E-02	1.07575

Table A2-3: Calculations for Boundary Concentrations and TEDE dose due to Unit Vent Release

Nuclide	Inventory (Ci)	Normalized	Varied concentration (uCi/cc)	Release Constant (hr*cc/cc)	Concentration @ Boundary (uCi*hr/cc)	Half Life (hr)	Decayed Concentration (uCi*hr/cc)	Dose Conversion Factor (rem per uCi*hr/cc)	Dose Contribution (rem)
I-131	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	1.93E+02	2.78E-08	5.30E+04	1.47E-03
I-132	1.50E+05	1.53E-03	3.77E-05	1.01E-03	3.81E-08	2.38E+00	2.79E-08	4.90E+03	1.37E-04
I-133	2.20E+05	2.25E-03	5.55E-05	1.01E-03	5.61E-08	2.03E+01	5.40E-08	1.50E+04	8.11E-04
I-134	2.40E+05	2.45E-03	6.04E-05	1.01E-03	6.11E-08	8.77E-01	2.61E-08	3.10E+03	8.09E-05
I-135	2.00E+05	2.05E-03	5.06E-05	1.01E-03	5.11E-08	6.61E+00	4.56E-08	8.10E+03	3.70E-04
Kr-83m	1.30E+06	1.33E-02	3.28E-04	1.01E-03	3.31E-07	1.83E+00	2.21E-07		0.00E+00
Kr-85m	2.90E+06	2.97E-02	7.33E-04	1.01E-03	7.40E-07	4.48E+00	6.27E-07	9.30E+01	5.83E-05
Kr-85	3.70E+05	3.78E-03	9.33E-05	1.01E-03	9.42E-08	9.40E+04	9.42E-08	1.30E+00	1.22E-07
Kr-87	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	1.27E+00	7.79E-07	5.10E+02	3.97E-04
Kr-88	7.80E+06	7.98E-02	1.97E-03	1.01E-03	1.99E-06	2.84E+00	1.53E-06	1.30E+03	1.99E-03
Kr-89	9.50E+06	9.72E-02	2.40E-03	1.01E-03	2.42E-06	5.10E-02	1.08E-12	1.20E+03	1.30E-09
Xe-131m	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	2.83E+02	2.78E-08	2.50E+02	1.36E-07
Xe-133m	6.80E+05	6.95E-03	1.71E-04	1.01E-03	1.73E-07	5.42E+01	1.71E-07	1.40E+02	2.90E-06
Xe-133	2.20E+07	2.25E-01	5.55E-03	1.01E-03	5.61E-06	1.27E+02	5.57E-06	1.10E+02	1.11E-04
Xe-135m	4.20E+06	4.30E-02	1.06E-03	1.01E-03	1.07E-06	2.60E-01	6.09E-08	7.20E+02	1.52E-05
Xe-135	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	9.08E+00	1.29E-06	5.30E+04	1.81E-04
Xe-137	1.90E+07	1.94E-01	4.79E-03	1.01E-03	4.83E-06	6.38E-02	4.06E-11	4.90E+03	4.47E-09
Xe-138	1.80E+07	1.84E-01	4.54E-03	1.01E-03	4.59E-06	2.36E-01	1.95E-07	1.50E+04	1.40E-04

Nuclide	Inventory	Normalized	Varied concentration	Release Constant	Concentration @ Boundary	Half Life	Decayed Concentration	Dose Conversion Factor	Dose Contribution
	(μCi/cc)		(μCi/cc)	(hr/cc/cc)	(μCi*hr/cc)	(hr)	(μCi*hr/cc)	rem per μCi*hr/cc	(rem)
Cs-134	3.70E+01	3.78E-07	9.33E-09	1.01E-03	9.42E-12	1.80E+04	9.42E-12	6.30E+04	5.93E-07
Cs-137	2.90E+01	2.97E-07	7.33E-09	1.01E-03	7.40E-12	2.60E+05	7.40E-12	4.10E+04	3.03E-07
Te132	4.80E+00	4.91E-08	1.21E-09	1.01E-03	1.22E-12	7.79E+01	1.21E-12	1.20E+04	1.45E-08
Mo99	1.22E+01	1.25E-07	3.08E-09	1.01E-03	3.11E-12	6.62E+01	3.08E-12	5.20E+03	1.60E-08
Ru103	8.80E-03	9.00E-11	2.22E-12	1.01E-03	2.24E-15	9.44E+02	2.24E-15	1.30E+04	2.91E-11
Ru106	2.90E-03	2.97E-11	7.33E-13	1.01E-03	7.40E-16	8.84E+03	7.40E-16	5.70E+05	4.22E-10
Zr95	1.10E-02	1.12E-10	2.76E-12	1.01E-03	2.79E-15	1.55E+03	2.79E-15	3.20E+04	8.93E-11
La140	1.90E-02	1.94E-10	4.79E-12	1.01E-03	4.83E-15	4.03E+01	4.75E-15	1.10E+04	5.22E-11
Ce144	7.40E-03	7.57E-11	1.87E-12	1.01E-03	1.89E-15	6.82E+03	1.89E-15	4.50E+05	8.49E-10
Ce-141	1.00E-02	1.02E-10	2.52E-12	1.01E-03	2.54E-15	7.77E+02	2.54E-15	1.10E+04	2.79E-11
Sr89	6.40E-02	6.55E-10	1.62E-11	1.01E-03	1.63E-14	1.21E+03	1.63E-14	5.00E+04	8.16E-10
Sr90	3.20E-03	3.27E-11	8.07E-13	1.01E-03	8.15E-16	2.50E+05	8.15E-16	1.60E+06	1.30E-09
								Total TEDE Dose	**5.77E-03**

Table A2-4: Thyroid Dose Calculation for Unit Vent Release

Nuclide	Decayed Concentration	Thyroid DCF	Thyroid Dose
	(uCi*hr/cc)	rem per uCi-hr/cc	(rem)
I-131	2.78E-08	1.30E+06	3.61E-02
I-132	2.79E-08	7.70E+03	2.15E-04
I-133	5.40E-08	2.20E+05	1.19E-02
I-134	2.61E-08	1.30E+03	3.39E-05
I-135	4.56E-08	3.80E+04	1.73E-03
		Total Thyroid Dose	5.00E-02

Table A2-5: Unit Vent Monitor Response to Nuclide Inventory

Nuclide	Concentration	Half Life	Concentration After 1 hr	Response Coeff	Response	
	(uCi/cc)	Hours	(uCi/cc)	(uCi/cc)	(uCi/cc)	
Kr-83m	3.28E-04	1.83E+00	2.25E-04		0.00E+00	
Kr-85m	7.33E-04	4.48E+00	6.28E-04	1.9	1.19E-03	
Kr-85	9.33E-05	9.40E+04	9.33E-05	2.4	2.24E-04	
Kr-87	1.39E-03	1.27E+00	8.03E-04	2.8	2.25E-03	
Kr-88	1.97E-03	2.84E+00	1.54E-03	2.3	3.55E-03	
Kr-89	2.40E-03	5.10E-02	3.00E-09	2.8	8.40E-09	
Xe-131m	2.76E-05	2.83E+02	2.76E-05	0.015	4.13E-07	
Xe-133m	1.71E-04	5.42E+01	1.69E-04	0.14	2.37E-05	
Xe-133	5.55E-03	1.27E+02	5.52E-03	1	5.52E-03	
Xe-135m	1.06E-03	2.60E-01	7.38E-05	0.042	3.10E-06	
Xe-135	1.39E-03	9.08E+00	1.28E-03	2.5	3.21E-03	
Xe-137	4.79E-03	6.38E-02	9.15E-08	2.8	2.56E-07	
Xe-138	4.54E-03	2.36E-01	2.41E-04	2.8	6.74E-04	
				Monitor Reading:	1.66E-02	1.57E+06
					(uCi/cc)	(uCi/sec)

Table A2-6: Input for Main Steam Line Release Calculation

Choke Flow	Limiting Release	Steam Specific Volume	Volumetric Release	Release Rate unit conversion	Variable concentration	Decay Time
(lbm/hr)	(mCi/sec)	(ft^3/lbm)	(ft^3/hr)	(cc/sec)	(µCi/cc)	(hr)
1.05E+06	3.37E+06	0.338	3.55E+05	2.79E+06	4.05	1.07575

Table A2-7: Calculations for Boundary Concentrations and TEDE dose due to Main Steam Line Release

Nuclide	Steam Inventory	Normalized	Varied Concentration	Release Constant*	Concentration @ Boundary	Half life	Decayed Concentration	DCF	Dose Contribution
			(µCi/cc)	(hr-cc/cc)	(µCi-hr/cc)	(hr)	(µCi-hr/cc)	(rem per µCi-hr/cc)	(rem)
I-131	6.10E-02	2.26E-04	9.27E-04	2.9853E-05	2.77E-08	1.93E+02	2.76E-08	5.30E+04	1.46E-03
I-132	8.61E-02	3.19E-04	1.31E-03	2.9853E-05	3.90E-08	2.38E+00	2.85E-08	4.90E+03	1.40E-04
I-133	1.00E-01	3.72E-04	1.53E-03	2.9853E-05	4.55E-08	2.03E+01	4.39E-08	1.50E+04	6.58E-04
I-134	1.86E-02	6.92E-05	2.84E-04	2.9853E-05	8.47E-09	8.77E-01	3.62E-09	3.10E+03	1.12E-05
I-135	2.73E-01	1.01E-03	4.14E-03	2.9853E-05	1.24E-07	6.61E+00	1.10E-07	8.10E+03	8.95E-04
Xe-131m	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.83E+02	1.27E-06	4.90E+00	6.22E-06
Xe-133	2.40E+02	8.90E-01	3.65E+00	2.9853E-05	1.09E-04	5.42E+01	1.07E-04	2.00E+01	2.15E-03
Xe-133m	4.20E+00	1.56E-02	6.40E-02	2.9853E-05	1.91E-06	1.27E+02	1.90E-06	1.70E+01	3.23E-05
Xe-135	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	2.60E-01	1.96E-07	1.40E+02	2.75E-05
Xe-135m	4.00E-01	1.48E-03	6.07E-03	2.9853E-05	1.81E-07	9.08E+00	1.67E-07	2.50E+02	4.17E-05
Xe-137	1.60E-01	5.93E-04	2.43E-03	2.9853E-05	7.26E-08	6.38E-02	6.10E-13	1.40E+02	8.53E-11
Xe-138	5.80E-01	2.15E-03	8.82E-03	2.9853E-05	2.63E-07	2.36E-01	1.12E-08	7.20E+02	8.04E-06
Kr-83m	3.70E-01	1.37E-03	5.62E-03	2.9853E-05	1.68E-07	1.83E+00	1.12E-07		0.00E+00
Kr-85	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	4.48E+00	2.92E-06	1.30E+00	3.80E-06
Kr-85m	1.50E+00	5.56E-03	2.28E-02	2.9853E-05	6.81E-07	9.40E+04	6.81E-07	9.30E+01	6.33E-05
Kr-87	9.80E-01	3.63E-03	1.49E-02	2.9853E-05	4.44E-07	1.27E+00	2.47E-07	5.10E+02	1.26E-04
Kr-88	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.84E+00	9.79E-07	1.30E+03	1.27E-03
Kr-89	8.40E-02	3.12E-04	1.28E-03	2.9853E-05	3.82E-08	5.10E-02	1.71E-14	1.20E+03	2.05E-11
								Total Dose	6.89E-03

*Release Constant = X/Q * duration * release rate

Table A2-8: Main Steam Line Release Thyroid Dose Calculation

Nuclide	Concentration @ Boundary	Thyroid DCF	Thyroid Dose
	(µCi*hr/cc)	rem per µCi*hr/cc	(rem)
I-131	2.76E-08	1.30E+06	3.58E-02
I-132	2.85E-08	7.70E+03	2.20E-04
I-133	4.39E-08	2.20E+05	9.66E-03
I-134	3.62E-09	1.30E+03	4.71E-06
I-135	1.10E-07	3.80E+04	4.20E-03
		Total	4.99E-02

Table A2-9: Main Steam Line Reading at Release

Nuclide	Concentration	Half Life	Concentration 1 Hour after Shutdown
	(µCi/cc)	Hours	(µCi/cc)
I-131	9.27E-04	1.93E+02	9.23E-04
I-132	1.31E-03	2.38E+00	9.77E-04
I-133	1.53E-03	2.03E+01	1.47E-03
I-134	2.84E-04	8.77E-01	1.29E-04
I-135	4.14E-03	6.61E+00	3.73E-03
Xe-131m	4.26E-02	2.83E+02	4.25E-02
Xe-133	3.65E+00	5.42E+01	3.60E+00
Xe-133m	6.40E-02	1.27E+02	6.36E-02
Xe-135	1.16E-01	2.60E-01	8.04E-03
Xe-135m	6.07E-03	9.08E+00	5.62E-03
Xe-137	2.43E-03	6.38E-02	4.65E-08
Xe-138	8.82E-03	2.36E-01	4.67E-04
Kr-83m	5.62E-03	1.83E+00	3.85E-03
Kr-85	1.16E-01	4.48E+00	9.90E-02
Kr-85m	2.28E-02	9.40E+04	2.28E-02
Kr-87	1.49E-02	1.27E+00	8.62E-03
Kr-88	4.26E-02	2.84E+00	3.34E-02
Kr-89	1.28E-03	5.10E-02	1.60E-09
		Total Activity	3.90E+00

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:24 User Name: Unit Vent Alert

Comments:

User Supplied Information

Meteorological Data Inputs:

Groundlevel wind velocity: 13.2 mi/hr

Groundlevel wind from: 180 degrees

User-selected Stability Class

Stability Class: "D - Neutral"

Monitored Unit Vent Release:

Unit Vent Release Rate entered: 2.50E+006 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:24

Release Start Date/Time: 12/17/2013 15:24

Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+006 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	2.53E+004	I-131:	3.12E+003	Cs-134:	1.05E+000
Kr-85:	1.05E+004	I-132:	3.18E+003	Cs-137:	8.25E-001
Kr-85M:	7.06E+004	I-133:	6.05E+003	Ce/Pr-144:	3.10E-004
Kr-87:	9.03E+004	I-134:	3.08E+003	Ce-141:	2.84E-004
Kr-88:	1.74E+005	I-135:	5.12E+003	La-140:	5.31E-004
Kr-89:	3.06E-001			Mo-99:	3.43E-001
Xe-131M:	3.12E+003			Ru/Rh-106:	8.25E-005
Xe-133:	6.22E+005			Ru-103:	2.50E-004
Xe-133M:	1.91E+004			Sr/Y-90:	9.10E-005
Xe-135:	1.45E+005			Sr-89:	1.82E-003
Xe-135M:	8.14E+003			Te-132:	1.35E-001
Xe-137:	9.53E+000			Zr-95:	3.13E-004
Xe-138:	2.66E+004				

12/17/2013 3:24:46 PM

DRILL **STAMPEDE Results Information** **DRILL**

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:24 User Name: Unit Vent Alert

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.540E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.009	0.016	0.137
1.0	0.003	0.006	0.051
2.0	0.001	0.002	0.018
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.009	0.016	0.137
1.0	0.003	0.006	0.051
2.0	0.001	0.002	0.018
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

DRILL **STAMPEDE Results Information** DRILL
Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE
Wind Velocity: 13.2 mi/hr
Wind Direction: 180
Release Rate: 1.19E+006 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.006	0.002	0.001	0.000
CDE	0.051	0.018	0.004	0.001

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of >3 mrem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:24:44 PM
Name Date/Time

REVIEWED BY: ______________________ ______________
Rad Manager/Radiological Director Date/Time

12/17/2013 3:24:28 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/18/2013 07:54 **User Name:** SteamLine Site Alert
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 138 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+000 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/18/2013 06:54
Release Start Date/Time: 12/18/2013 07:54
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.19E+007 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+004	I-131:	3.05E+003	Cs-134:	0.00E+000
Kr-85:	3.43E+005	I-132:	3.22E+003	Cs-137:	0.00E+000
Kr-85M:	5.79E+004	I-133:	4.88E+003	Ce/Pr-144:	0.00E+000
Kr-87:	2.55E+004	I-134:	4.22E+002	Ce-141:	0.00E+000
Kr-88:	9.80E+004	I-135:	1.23E+004	La-140:	0.00E+000
Kr-89:	4.25E-003			Mo-99:	0.00E+000
Xe-131M:	1.26E+005			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+007			Ru-103:	0.00E+000
Xe-133M:	1.87E+005			Sr/Y-90:	0.00E+000
Xe-135:	3.18E+005			Sr-89:	0.00E+000
Xe-135M:	1.23E+003			Te-132:	0.00E+000
Xe-137:	1.27E-001			Zr-95:	0.00E+000
Xe-138:	1.36E+003				

12/18/2013 7:55:19 AM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/18/2013 07:54 User Name: Steam Line Site Alert

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.082E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.011	0.019	0.135
1.0	0.0042	0.007	0.050
2.0	0.002	0.003	0.017
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.011	0.019	0.135
1.0	0.004	0.007	0.050
2.0	0.002	0.003	0.017
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

12/18/2013 7:54:42 AM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+007 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.007	0.003	0.001	0.000
CDE	0.050	0.017	0.004	0.001

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of ≥ 3 mrem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY:

Name	Date/Time
	12/18/2013 7:55:14 AM

REVIEWED BY:

Rad Manager/Radiological Director	Date/Time

12/18/2013 7:54:42 AM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:25 **User Name:** Unit Vent Site Area

Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored Unit Vent Release:
Unit Vent Release Rate entered: 2.50E+007 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:25
Release Start Date/Time: 12/17/2013 15:25
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+007 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	2.52E+005	I-131:	3.12E+004	Cs-134:	1.05E+001
Kr-85:	1.05E+005	I-132:	3.18E+004	Cs-137:	6.24E+000
Kr-85M:	7.06E+005	I-133:	6.04E+004	Ce/Pr-144:	2.10E-003
Kr-87:	9.03E+005	I-134:	3.08E+004	Ce-141:	2.84E-003
Kr-88:	1.73E+006	I-135:	5.12E+004	La-140:	5.31E-003
Kr-89:	3.14E+000			Mo-99:	3.43E+000
Xe-131M:	3.12E+004			Ru/Rh-106:	8.24E-004
Xe-133:	6.22E+006			Ru-103:	2.90E-003
Xe-133M:	1.91E+005			Sr/Y-90:	9.10E-004
Xe-135:	1.45E+006			Sr-89:	1.82E-002
Xe-135M:	8.18E+004			Te-132:	1.35E+000
Xe-137:	9.74E+001			Zr-95:	3.13E-003
Xe-138:	2.67E+005				

12/17/2013 3:25:33 PM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:25 User Name: Unit Vent Site Area
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-006
1.0	0:05	1.082E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

Distance (miles)	Measurable Dose Rates: Immersion Whole Body noble gas gamma (rem/hr)	PAG Dose Rates: TEDE external + internal (rem/hr)	PAG Dose Rates: Iodine CDE Thyroid (rem/hr)
0.5	0.088	0.160	1.364
1.0	0.0332	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.014
20.0	0.000	0.001	0.005

Distance (miles)	Measurable Doses: Immersion Whole Body noble gas gamma (rem)	PAG Doses: TEDE external + internal (rem)	PAG Doses: Iodine CDE Thyroid (rem)
0.5	0.088	0.160	1.364
1.0	0.033	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.014
20.0	0.000	0.001	0.005

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.5.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+007 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.060	0.021	0.005	0.002
CDE	0.510	0.176	0.041	0.014

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RS1 (SITE AREA EMERGENCY) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:25:28 PM

Name Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director Date/Time

12/17/2013 3:25:21 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:28 **User Name:** Steam Line Site Area

Comments:

User Supplied Information

Meteorological Data Inputs:

Ground level wind velocity:	13.2 mi/hr
Ground level wind from:	180 degrees
User-selected Stability Class	
Stability Class:	"D - Neutral"

Monitored S/G Tube Rupture Release:

Steam Activity:	4.50E+001 uCi/cc
Steam Flow Rate:	1.050 mlb/hr
Reactor Shutdown Date/Time:	12/17/2013 14:28
Release Start Date/Time:	12/17/2013 15:28
Estimated Release Duration:	1.00 hours
Nuclide Mixture:	Noble Gas + Iodine
Iodine as percent of noble gas:	0.2%

Calculated NOBLE GAS release rate: 1.20E+008 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	**uCi/sec**	**Nuclide**	**uCi/sec**	**Nuclide**	**uCi/sec**
Kr-83M:	1.14E+005	I-131:	3.07E+004	Cs-134:	0.00E+000
Kr-85:	3.45E+006	I-132:	3.23E+004	Cs-137:	0.00E+000
Kr-85M:	5.81E+005	I-133:	4.90E+004	Ce/Pr-144:	0.00E+000
Kr-87:	2.55E+005	I-134:	4.22E+003	Ce-141:	0.00E+000
Kr-88:	9.91E+005	I-135:	1.24E+005	La-140:	0.00E+000
Kr-89:	3.92E-002			Mo-99:	0.00E+000
Xe-131M:	1.27E+006			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+008			Ru-103:	0.00E+000
Xe-133M:	1.88E+006			Sr/Y-90:	0.00E+000
Xe-135:	3.19E+006			Sr-89:	0.00E+000
Xe-135M:	1.21E+004			Te-132:	0.00E+000
Xe-137:	1.19E+000			Zr-95:	0.00E+000
Xe-138:	1.34E+004				

12/17/2013 3:29:03 PM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:28 **User Name:** Steam Line Site Area

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.431E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.111	0.189	1.354
1.0	0.0423	0.072	0.506
2.0	0.015	0.025	0.175
5.0	0.004	0.006	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.013
20.0	0.001	0.001	0.005

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.111	0.189	1.354
1.0	0.042	0.072	0.506
2.0	0.015	0.025	0.175
5.0	0.004	0.006	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.013
20.0	0.001	0.001	0.005

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.20E+008 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.072	0.025	0.006	0.002
CDE	0.506	0.175	0.041	0.013

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RS1 (SITE AREA EMERGENCY) has been met.

PERFORMED BY:

_______________________ 12/17/2013 3:29:00 PM

Name Date/Time

REVIEWED BY:

_______________________ _______________________

~~Rad Manager/Radiological Director~~ ~~Date/Time~~

12/17/2013 3:28:53 PM

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:26 User Name: Unit Vent General

Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored Unit Vent Release:
Unit Vent Release Rate entered: 1.50E+008 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:26
Release Start Date/Time: 12/17/2013 15:26
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+008 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	2.52E+006	I-131:	3.12E+005	Cs-134:	1.05E+002
Kr-85:	1.09E+006	I-132:	3.18E+006	Cs-137:	8.25E+001
Kr-85M:	7.06E+006	I-133:	6.04E+005	Ce/Pr-144:	2.10E-002
Kr-87:	9.03E+006	I-134:	3.08E+005	Ce-141:	2.84E-002
Kr-88:	1.73E+007	I-135:	5.12E+005	La-140:	5.31E-002
Kr-89:	3.10E+001			Mo-99:	3.43E+001
Xe-131M:	3.12E+005			Ru/Rh-106:	8.25E-003
Xe-133:	6.22E+007			Ru-103:	2.50E-002
Xe-133M:	1.91E+006			Sr/Y-90:	9.10E-003
Xe-135:	1.45E+007			Sr-89:	1.82E-001
Xe-135M:	8.16E+005			Te-132:	1.35E+001
Xe-137:	9.64E+002			Zr-95:	3.13E-002
Xe-138:	2.66E+006				

12/17/2013 3:26:37 PM

DRILL STAMPEDE Results Information DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:26 User Name: Unit Vent General
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m^3)	CHI/Q DEPL (sec/m^3)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.879	1.598	13.646
1.0	0.3322	0.601	5.099
2.0	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.006	0.050

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.879	1.598	13.646
1.0	0.332	0.601	5.099
2.0	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.006	0.050

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+008 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.601	0.210	0.050	0.017
CDE	5.099	1.762	0.411	0.135

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1, 2

SHELTER IN PLACE ZONE(S): 6, 11

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 1 rem TEDE and/or 5 rem Thyroid CDE the Emergency Classification Initiating Condition RG1 (GENERAL EMERGENCY) has been met

PERFORMED BY: ______________________ 12/17/2013 3:26:33 PM

Name — Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director — Date/Time

12/17/2013 3:26:25 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:30 User Name: SteamLine General

Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+002 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/17/2013 14:30
Release Start Date/Time: 12/17/2013 15:30
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.20E+009 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+006	I-131:	3.07E+005	Cs-134:	0.00E+000
Kr-85:	3.45E+007	I-132:	3.24E+005	Cs-137:	0.00E+000
Kr-85M:	5.82E+006	I-133:	4.90E+005	Ce/Pr-144:	0.00E+000
Kr-87:	2.56E+006	I-134:	4.24E+004	Ce-141:	0.00E+000
Kr-88:	9.92E+006	I-135:	1.24E+006	La-140:	0.00E+000
Kr-89:	4.13E-001			Mo-99:	0.00E+000
Xe-131M:	1.27E+007			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+009			Ru-103:	0.00E+000
Xe-133M:	1.38E+007			Sr/Y-90:	0.00E+000
Xe-135:	3.19E+007			Sr-89:	0.00E+000
Xe-135M:	1.23E+005			Te-132:	0.00E+000
Xe-137:	1.24E+001			Zr-95:	0.00E+000
Xe-138:	1.35E+005				

12/17/2013 3:30:50 PM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:30 User Name: SteamLine General
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.945E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	1.108	1.895	13.537
1.0	0.4236	0.717	5.058
2.0	0.153	0.254	1.747
5.0	0.040	0.063	0.407
7.5	0.022	0.034	0.211
10.0	0.015	0.022	0.134
20.0	0.006	0.008	0.049

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	1.168	1.895	13.537
1.0	0.424	0.717	5.058
2.0	0.153	0.254	1.747
5.0	0.040	0.063	0.407
7.5	0.022	0.034	0.211
10.0	0.015	0.022	0.134
20.0	0.006	0.008	0.049

DRILL — STAMPEDE Results Information — DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.20E+009 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.727	0.254	0.063	0.022
CDE	$\boxed{5.068}$	1.747	0.407	0.134

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1, 2

SHELTER IN PLACE ZONE(S): 6, 11

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

> Based on a Site Boundary (1 Mile) Dose Projection ≥ 1 rem TEDE and/or 5 rem Thyroid CDE the Emergency Classification Initiating Condition RG1 (GENERAL EMERGENCY) has been met

PERFORMED BY: ______________________ 12/17/2013 3:30:48 PM

Name Date/Time

REVIEWED BY: ______________________ ______________

~~Rad Manager/Radiological Director~~ ~~Date/Time~~

CN-2883

The particulate channel is used as part of the Reactor Coolant Pressure Boundary (RCPB) leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the Containment, are sufficient to detect an increase in leakage rate of the equivalent of one gal/min within one hour. Elements of this monitor, including the indicator mounted in the RMS CR cabinet, are designed and qualified to remain functional following a Safe Shutdown Earthquake (SSE), in compliance with RG 1.45. Further information on the RCPB leakage detection system is presented in Section 5.2.5.

11.5.2.3.3 Unit Vent Monitor: The unit vent monitor samples the plant vent stack prior to discharge to the environment and monitor for particulates, iodine, and noble gases.

The unit vent particulate and iodine monitor draws representative air samples from the plant vent stack via isokinetic nozzles in the stack, and directs them through a moving filter paper monitored by a shielded beta-sensitive scintillation detector. The sample stream then passes through a charcoal collector, where collected iodine is monitored by a shielded gamma-sensitive scintillation detector. The sample is then returned to the vent stack.

A separate wide-range gas monitor is provided for the unit vent. The monitor has two isokinetic nozzles for sampling during both normal and accident conditions. The stack samples pass first through a sample conditioning unit which filters particulates and iodine and may be used to take grab samples. The samples then pass through the shielded detector assembly, which uses three detectors to cover the complete range required. The low range detector uses a beta-sensitive plastic scintillator-photomultiplier (PM) tube. The mid-range and high-range detectors use cadmium telluride (CdTe), chlorine-doped, solid-state sensors. This wide-range gas monitor satisfies the requirements of NUREG-0737, Item II.F.1 for provisions for sampling plant effluents for iodines and particulates and for noble gas effluents from the plant vent.

11.5.2.3.4 Control Room Electrical Auxiliary Building Ventilation Monitors: The CR/EAB ventilation monitors are Class 1E monitors which continuously assess the intake air to the CR for indication of abnormal airborne radioactivity concentration. Each monitor assembly is powered from a separate electrical power source. In the event of high radiation CR emergency ventilation operation is initiated (Section 7.3.2). Failure of a monitor is alarmed in the CR.

Each monitor assembly is comprised of a recirculation pump, beta-sensitive scintillation detector, four-pi lead shielding, check source, stainless steel sample gas receiving chamber, and associated electronics.

11.5.2.3.5 Condenser Vacuum Pump Monitor: Gaseous samples are drawn through an off-~~line system by a pump from the discharge of the vacuum pump exhaust header of the condenser.~~ This channel monitors the gaseous sample for radioactivity which would be indicative of an SG tube leak, allowing reactor coolant to enter the secondary side fluid; this monitor complements the SGBD monitors in indication of a SG tube leak. The gaseous radioactivity levels are monitored by a single detector in a manner similar to the unit vent wide range gas monitor.

11.5.2.3.6 Spent Fuel Pool Exhaust Monitors: The SFPE monitors are Class 1E and are identical to the CR/EAB ventilation monitors described in Section 11.5.2.3.4 except that they sample the exhaust from the FHB. In the event of high radiation the monitors initiate emergency operation

11.5.2.5.1 Gaseous Waste Processing System Inlet Monitor: The GWPS inlet monitor employs a gamma (NaI crystal) scintillator/photomultiplier tube combination to measure the radioactivity level of the waste gases entering the GWPS. The monitor is used in conjunction with the GWPS discharge monitor to measure overall effectiveness of the GWPS.

11.5.2.5.2 GWPS Discharge Monitor: This monitor is similar to the GWPS inlet monitor and is installed upstream of the GWPS discharge valve. Upon detection of high radioactivity or monitor failure, the GWPS discharge valve, FV-4671, is automatically closed.

11.5.2.5.3 Main Steam Line Monitors: Each MS line is monitored by an ATL monitor consisting of a Geiger Mueller (GM) tube detector and an ion chamber detector with overlapping ranges. The detectors are shielded by 3 in. of lead.

The monitors are designed to monitor gross gamma activity in the steam line and provide a basis for determining possible atmospheric releases from the MS power-operated relief valve (PORV), SG safety valves, and/or auxiliary feedwater pump turbine.

The monitors provide a dose rate range equivalent to 10^{-1} to 10^{3} $\mu Ci/cm^3$ xenon-133. Based upon core inventory, the ratio of xenon-133 to other nuclides in the fuel can be detemined. In order to obtain the above concentrations of xenon-133 in the main steam line, a large primary-to-secondary leak must be present coincident with a large amount of fuel failure. The presence of xenon-133 indicates other radioactive isotopes are present.

Using the relative ratios of isotopes present in the MS line, a computer model for determination of dose rates from these isotopes, detector response curves, the thickness of the MS line, and the geometry of the MS line relative to the detector, the dose rate equivalent to MS line concentration is obtained. The quantity of radioactive effluents released is obtained by multiplying the xenon-133 equivalent MS line concentrations by the isotope ratio times the steam release rate.

These detectors are safety-related Class 1E and meet the requirements of RG 1.97 and NUREG-0737.

11.5.2.5.4 Steam Generator Blowdown Monitors: These monitors are identical to the MS line monitors and are adjacent to the SG blowdown lines in the Isolation Valve Cubicle (IVC).

The monitors are used as an aid in determining the source of SG blowdown radioactivity due to SG tube rupture or a large primary-to-secondary leak.

~~These detectors are safety-related Class 1E and meet the requirements of RG 1.97.~~

11.5.2.5.5 Main Steam Line High Energy Gamma (N-16) Monitors: Each main steam line is monitored by an ATL NaI scintillation detector. These detectors were installed to monitor the status of steam generator primary to secondary tube leaks and to provide a diagnostic tool for all individuals concerned with steam generator condition. These detectors are designed to detect high energy gamma activity in the 6 to 7.2 MEV energy range. High energy gamma activity in the main steam lines indicates the presence of N-16. The level of N-16 in the main steam lines is used to

RS2

The new fuel assemblies are transported to the new fuel storage pit or to the new fuel elevator by the 15/2-ton, dual-service FHB crane. The 2-ton hoist of this crane is designed to handle new fuel assemblies. New fuel handling is discussed in detail in Section 9.1.4. Use of the 2-ton hoist of the 15/2-ton crane or of the fuel-handling machine to handle new fuel ensures that the design uplift of the racks will not be exceeded.

The new fuel storage pit is situated in the approximate center of each FHB. The floor of the new fuel storage pit is at El. 50 ft-3 inches. The new fuel storage pit access hatch is provided with a three-section protective cover at El. 68 ft. The fuel assemblies are loaded into the new fuel storage racks through the top and stored vertically.

9.1.1.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities.

Flood protection of each FHB is discussed in Section 3.4.1. Flooding of the new fuel storage pit from fluid sources inside either FHB is not considered credible since all fluid systems components are located well below the elevation of the new fuel storage pit access hatch. A floor drain is provided in the new fuel storage pit to minimize collection of water.

The applicable design codes and the ability of the FHB to withstand various external loads and forces are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7. Missile protection of the FHBs is discussed in Section 3.5. Failure of nonseismic systems or structures will not decrease the degree of subcriticality provided in the new fuel storage pit.

In accordance with American National Standards Institute (ANSI) N18.2, the design of the normally dry new fuel storage racks is such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place, assuming optimum moderation (under dry or fogged conditions). For the unborated flooded condition, assuming new fuel of the highest anticipated enrichment in place, the effective multiplication factor does not exceed 0.95. Credit may be taken for the inherent neutron-absorbing effect of the materials of construction.

The new fuel assemblies are stored dry, the 21-in. spacing ensuring a safe geometric array. Under these conditions, a criticality accident during refueling and storage is not considered credible. Consideration of criticality safety analysis is discussed in Section 4.3.

Design of the facility in accordance with RG 1.13 ensures adequate safety under both normal and postulated accident conditions. The new fuel storage racks also meet the requirements of General Design Criterion (GDC) 62.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases. The spent fuel pool (SFP) is a stainless steel-lined reinforced concrete pool and is an integral part of each FHB. All spent fuel racks are classified as seismic Category I, as defined by RG 1.29, and as ANS SC 3.

The spent fuel storage facility provides storage capacity for 1,969 high density absorber spent fuel racks in a honeycomb array in each unit. Two storage regions are provided in the SFP. Two of the

Region 2 rack modules on the south end of the pool (modules #12 and #16) have not been installed. A Fuel Ultrasonic Cleaning system may be used in the open space designated for modules #12 and #16. The Fuel Ultrasonic Cleaning system is freestanding and is seismically qualified. It has no adverse effect on the fuel assemblies that are selected for cleaning; nor does it have an adverse effect on the design function of the spent fuel pool or its associated support systems. Figure 9.1.2-2 shows the pool layout for both Units 1 and 2. The six Region 1 rack modules are located in the northwest corner of the spent fuel pool.

The Region 1 racks have 10.95-in. nominal center-to-center spacing between the cells. This region is conservatively designed to accommodate unirradiated fuel at enrichments to 4.95 weight percent. Region 1 storage cells are each bounded on four sides by a water box except on the periphery of the pool. The Region 1 spent fuel racks include a lead-in-guide to assist in depositing fuel assemblies into the fuel cell. Figure 9.1.2-3 shows a typical Region 1 spent fuel rack.

The reactivity characteristics of fuel assemblies which are to be placed in the spent fuel storage racks are determined and the assemblies are categorized by reactivity. Alternately, if necessary, all assemblies may be treated as if each assembly is of the highest reactivity class until the actual assembly reactivity classification is determined. Section 5.6 of the Technical Specifications provides the definitions of the reactivity classifications and the allowed storage patterns. Fuel assemblies are loaded into the racks in a geometrically safe configuration to ensure rack subcriticality.

Fuel assembly reactivity requirements for close packed storage and checkerboard storage are specified in the Technical Specifications. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum value needed to ensure that the rack K_{eff} is less than or equal to 0.95 in the event of misplaced assemblies in the close packed storage areas or in checkerboard storage areas. Consideration of criticality safety is discussed in Section 4.3.

The Region 2 racks have a 9.15-in. nominal center-to-center spacing with fixed absorber material surrounding each cell. A sheet of neutron absorber material is captured between the side walls of all adjacent boxes. To provide space for the absorber sheet between boxes, a double row of matching flat round raised areas are coined into the side walls of all boxes. The raised dimension of these locally formed areas on each box wall is half the thickness of the absorber sheet. The boxes are fusion welded together at all these local areas. The absorber sheets are scalloped along their edges to clear these areas. Figure 9.1.2-4 shows a typical Region 2 spent fuel rack.

The axial location of the absorber with respect to the active fuel region is provided and maintained by the structure of each box. At the outside periphery of each rack, a sheet of absorber material is captured under thin stainless sheets which are intermittently welded all around to the box.

All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion during normal and emergency water quality conditions. The racks are designed to withstand normal operating loads as well as to remain functional with the occurrence of an SSE. The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist. The racks are designed to withstand a maximum uplift force equal to the uplift force of the bridge hoist. The 14-in. and 16-in. racks also meet the requirements of ASME Code, Section III, Appendix XVII. The high-density spent fuel racks meet the criteria of Appendix D to Standard Review Plan (SRP) 3.8.4.

Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 Facilities Description. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum

NRC ORDER EA-12-051 (SFP LEVELS)

REQUIREMENTS FOR RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION AT OPERATING REACTOR SITES AND CONSTRUCTION PERMIT HOLDERS

All licensees identified in Attachment 1 to this Order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

1. The spent fuel pool level instrumentation shall include the following design features:

   1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.

   1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

   1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

   1.4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

   1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.

   1.6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

1.7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

1.8 Testing: The instrument channel design shall provide for routine testing and calibration.

1.9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.

2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

The three critical levels that must be monitored in a spent fuel pool are discussed below. It should be noted that continuous indication from a single instrument over the entire span from level 1 to level 3 is not required but could be used. If more than one instrument is used to monitor the entire span, that set of instruments constitutes a single channel satisfying either the primary or backup instrument channel requirement (refer to Figure 1 below).

A visual representation of monitoring levels 1, 2 and 3 and the associated requirements for monitoring between the points are presented in Figure 1. The minimum requirements apply to the separation distance between level indications and support development of appropriate response procedures. These requirements are separate from the instrument channel design accuracy discussed in section 3, which apply to either discrete or to continuous instruments.

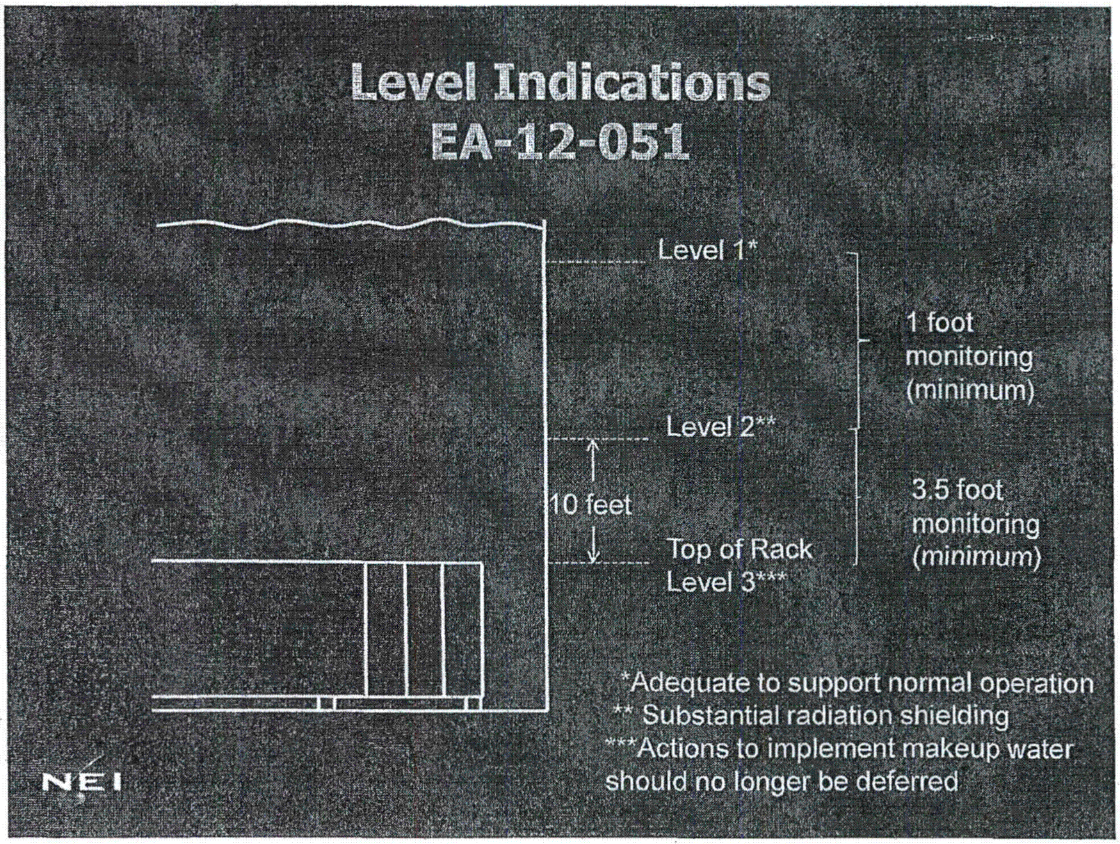


Figure 1

2.3.1. Level 1 – level that is adequate to support operation of the normal fuel pool cooling system

A typical fuel pool cooling system design includes a combination of weirs and/or vacuum breakers that prevent siphoning of the pool water level, below a minimum level, in the event of a piping rupture that can affect the SFP level. Level 1 represents the HIGHER of the following two points:

- The level at which reliable suction loss occurs due to uncovering of the coolant inlet pipe, weir or vacuum breaker (depending on the design), or
- The level at which the water height, assuming saturated conditions, above the centerline of the cooling pump suction provides the required net positive suction head specified by the pump manufacturer or engineering analysis.

This level will vary from plant to plant and the instrument designer will need to consult plant-specific design information to determine the actual point that supports adequate cooling system performance.

2.3.2. Level 2 – level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck

Level 2 represents the range of water level where any necessary operations in the vicinity of the spent fuel pool can be completed without significant dose consequences from direct gamma radiation from the stored spent fuel. Level 2 is based on either of the following:

- 10 feet (+/- 1 foot) above the highest point of any fuel rack seated in the spent fuel pools, or
- a designated level that provides adequate radiation shielding to maintain personnel radiological dose levels within acceptable limits while performing local operations in the vicinity of the pool. This level shall be based on either plant-specific or appropriate generic shielding calculations, considering the emergency conditions that may apply at the time and the scope of necessary local operations, including installation of portable SFP instrument channel components. Additional guidance can be found in EPA-400 (Reference 4), USNRC Regulatory Guide 1.13 (Reference 5) and ANSI/ANS-57.2-1983 (Reference 6).

Designation of this level should not be interpreted to imply that actions to initiate water make-up must be delayed until SFP water levels have reached or are lower than this point.

2.3.3. Level 3 – level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

Level 3 corresponds nominally (i.e., +/- 1 foot) to the highest point of any fuel rack seated in the spent fuel pool. Level 3 is defined in this manner to provide the maximum range of information to operators, decision makers and emergency response personnel. Designation of this level should not be interpreted to imply that actions to initiate water make-up must or should be delayed until this level is reached.

Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 28, 2013
NOC-AE-13002959
10 CFR 50.4
10 CFR 2.202

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1&2
Docket Nos. STN 50-498, STN 50-499
Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to
Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

References:
1. Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051)
2. NRC Interim Staff Guidance JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, August 29, 2012
3. Letter D. W. Rencurrel to NRC, "Initial Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)", dated October 24, 2012

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued an order (Reference 1) to STP Nuclear Operating Company (STPNOC). Reference 1 directs STP Nuclear Operating Company to provide a reliable indication of the water level in associated spent fuel storage pools. Specific requirements are outlined in Attachment 2 of Reference 1.

Reference 1 required submission of an overall integrated plan, including how compliance will be achieved. The final interim staff guidance (Reference 2) was issued August 29, 2012 providing licensees an acceptable approach for complying with the order. The purpose of this letter is to provide the overall integrated plan, including a description of how compliance will be achieved pursuant to Section IV, Condition C.1.a, of Reference 1 in accordance with the guidance in Attachment 2 to Reference 1 and the guidance in Reference 2. See the Enclosure for STPNOC's response to the requested information.

There are no new commitments in this letter.

33650640

If there are any questions regarding this letter, please contact Robyn Savage at (361) 972-7438.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2/28/13

Dennis L. Koehl

Dennis L. Koehl
President and CEO/CNO

Enclosure: South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051

cc:
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South Texas Project (STP)
Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation
to Meet NRC Order EA-12-051

1.0 OVERALL INTEGRATED PLAN INTRODUCTION

This document provides the overall Integrated Plan (the "Plan") which the STP Nuclear Operating Company ("STPNOC") will implement for Units 1 and 2 to comply with the requirements of NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Ref.2), (the "ORDER"), NRC Interim Staff Guidance JLD-ISG-2012-003 [Rev.0] (Ref.3), (the "ISG"), and NEI Report 12-02 [Rev.1] ("NEI 12-02").

This Plan follows the format and provides all of the information on the STP 1 & 2 Integrated Plan that is required in NEI 12-02 [Rev.1] (Ref.1), Section A-2-2. Throughout this Plan, any reference to NEI 12-02 and the ISG will be based on the revisions above. Any reference to NEI 12-02 will include compliance to the clarifications and exceptions to NEI 12-02 required by the Interim Staff Guidance, Rev. 0.

In response to the NRC requirements, STPNOC will provide two channels of independent, permanently-installed, wide-range spent fuel pool level instrumentation ("SFPLI"), for the spent fuel pool ("SFP") of each unit. The spent fuel pool for each unit is independent and not interconnected in any way. For each SFP, the instrumentation provided for each channel will utilize the same technology, as permitted by the NEI 12-02 [Rev.1]. The spent fuel pool level instrumentation will provide continuous level indication for each SFP on both the Primary and Backup Channels.

Both the Primary and Backup Channel/Instrument location and display of the SFP level will be independently mounted in each unit's Radwaste Control Room in the Mechanical Electrical Auxiliary Building (MEAB), which is an accessible post-event location. Other locations are still being considered.

Both the Primary and Backup Channel remote, non-safety related indication of the SFP level will also be provided in each unit's Control Room via input to the Plant Computer.

The instrumentation systems will not be safety-related, but will meet the requirements for augmented quality in accordance with NEI 12-02 [Rev.1] and the ISG as described below.

Since all of the potential suppliers have not completed development, the information in this Plan is based on the overall strategy and on information which, based on current information from potential suppliers, is thought to envelope the systems being developed for this application.

If there are any changes to the requirements in NRC JLD-ISG-2012-003 [Rev.0] and NEI 12-02 [Rev.1], relief from the requirements and schedule documented in this Plan may be required, in accordance with Section 12.0. Any required changes to this Plan will be defined in the periodic status reports submitted to the NRC.

2.0 APPLICABILITY:

This Plan applies to the spent fuel pools for South Texas Project Unit 1 and Unit 2.

3.0 SCHEDULE:

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 1 is scheduled for completion prior to 10/28/2015, which is the end of the second refueling outage (1RE19) following submittal of this Plan.

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 2 is scheduled for completion prior to 4/29/2015, which is the end of the second refueling outage (2RE17) following submittal of this Plan.

Unit 1 Milestones are as follows:

- Design/Engineering – September of 2014
- Purchase of instruments & equipment – February of 2015
- Receipt of equipment – June of 2015
- Unit 1 Installation & Functional Testing – October of 2015

Unit 2 Milestones are as follows:

- Design/Engineering – December of 2013
- Purchase of instruments & equipment – August of 2014
- Receipt of equipment – November of 2014
- Installation & Functional Testing – April of 2015

Consistent with the requirements of the ORDER and the guidance from NEI 12-02 [Rev.1], status reports will be generated in six (6) month intervals from the submittal of this Plan.

4.0 IDENTIFICATION OF SPENT FUEL POOL WATER LEVELS:

The STP Unit 1 and 2 spent fuel pools are essentially identical. The following SFP elevations are identified:

- The bottom of the pool is at Plant El. 21 ft. 11 in.
- The top of the SFP racks is approximately at Plant El. 39 ft. 10 in.
- The minimum Limiting Condition for Operation SFP level is Plant El. 62 ft.
- Normal SFP water level is at Plant El. 66 ft. 6 in.
- Non-safety related level switch alarms are activated at Plant El. 67 ft. on high level and Plant El. 66 ft. on low level.
- The spent fuel pool deck is at Plant El. 68 ft.

The required key SFP water levels, per guidance of NEI 12-02 [Rev.1] and ISG (with clarifications and exceptions), are as follows:

4.1 LEVEL 1: Level adequate to support operation of the normal fuel pool cooling system.

LEVEL 1 represents the HIGHER of either the level at which reliable suction loss to the cooling pump occurs, or, the required NPSH (Nominal Pump Suction Head) of the cooling pump.

Loss of reliable suction to SFP cooling pumps. For the purposes of this Plan, this level will conservatively be placed at Plant El. 64 ft. 2 in. This allows for just over 1 ft. of SFP water level above the top of the suction inlet flange (SFP Cooling Pump 14 in. suction line with centerline of suction inlet flange at Plant El. 62 ft. 6 in.) which will be sufficient for NPSH. (Ref. 9)

Therefore, considering the top of SFP fuel storage rack is at Plant El. 39 ft. 10 in., the indicated level on either the Primary or Backup Instrument Channel of greater than 24 ft. 4 in. above the top of the SFP fuel storage racks based on the design accuracy for the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, is adequate for normal SFP cooling system operation.

LEVEL 1 = Plant El. 64 ft. 2 in or 24 ft. 4 in. water level above the top of the SFP fuel storage rack

4.2 LEVEL 2: Level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck.

Indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of SFP stored fuel assemblies based on current guidance in NRC RG 1.13 [Rev.2] (Ref. 4) will achieve substantial radiation shielding. Requirements on substantial SFP radiation shielding is also given in ANSI/ANS-57.2-1983 (Ref. 5), and states that radiation shall not exceed 2.5 mRem/hr, but the minimum water depth to achieve this is undefined. NRC RG 1.13 [Rev.2] took exception to using dose rates as design input for minimum SFP water level, and instead defined the minimum level as 10 ft. above the stored fuel assemblies.

STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. Therefore, indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of the SFP fuel storage rack, based on the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, ensures there is adequate water level to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck.

LEVEL 2 = Plant El 49 ft. 10 in. or 10 ft. water level above the top of the SFP fuel storage rack.

4.3 LEVEL 3: Level where the fuel remains covered.

As stated above, STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. The installation of the SFPLI sensor will be such that it will measure as close as possible to the top of the SFP fuel rack. Indicated level on either the Primary or Backup Instrument Channel of greater than ½ ft. above the top of SFP fuel storage racks based upon the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, satisfies the NEI 12-02 [Rev.2] requirement of ± 1 ft. from the top of the fuel rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the SFP fuel storage rack.

LEVEL 3 = Plant El 40 ft. 4 in. or 6 in. water level above the top of the SFP fuel storage rack.

5.0 INSTRUMENTS:

Both the Primary and Backup Instrument Channels will utilize permanently-installed instruments. The design of the primary and backup instruments will be consistent with the requirements by NEI 12-02 [Rev.1], the ISG, and this Plan.

The current plan is for both channels to utilize Guided Wave Radar, which functions according to the principle of Time Domain Reflectometry (TDR). A generated pulse of electromagnetic energy travels down the probe. Upon reaching the liquid surface the pulse is reflected and based upon reflection times level is inferred. The measured range will be continuous from the high pool level elevation (67') to the top of the spent fuel racks (Ref. 8). However, STP is still evaluating other designs for this application. Any changes to the current plan will be reported in the 6 month update letter.

The supplier for the SFP instrumentation will be chosen by a competitive bidding process completed after submittal of this Plan, so the information in this Plan is based on the overall strategy and on available information from potential supplier's information on systems being developed for this application.

5.1 Primary (fixed) Instrument Channel

The Primary Instrument Channel level sensing components will be located in the northeast corner of the Spent Fuel Pool, as shown in Attachment 1. The primary instrument channel will provide continuous level indication over a range from Plant El. 40 ft. 4 in. (LEVEL 3) to Plant El. 67 ft. (SFP high level alarm) or a range of 26 ft. 8 in. In addition, the capability for discrete level indications at LEVEL1, LEVEL 2 and LEVEL 3, as described in Section 4.0, will be available.

Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

June 25, 2013
NOC-AE-13003008
File No.: G25
10 CFR 2.202

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Unit 1 & 2
Docket Nos. STN 50-498, STN 50-499
Response to Request for Additional Information
Regarding the Overall Integrated Plan in Response to Order EA-12-051,
"Reliable Spent Fuel Pool Instrumentation" (TAC Nos. MF0827 and MF0828)

References:

1. Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051) (ST-AE-NOC-12002271) (ML12054A679)

2. Letter, D. L. Koehl to NRC Document Control Desk, "Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated February 28, 2013 (NOC-AE-13002959) (ML13070A006)

3. NRC letter dated June 7, 2013, "South Texas Project, Units 1 and 2 – Request for Additional Information RE: Overall Integrated Plan in Response to Order EA-12-051, "Reliable Spent Fuel Pool Instrumentation" (TAC Nos. MF0827 and MF0828) (ST-AE-NOC-13002439) (ML131149A09)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued an Order (Reference 1) modifying licenses with regard to requirements for reliable spent fuel pool instrumentation. On February 28, 2013, STP Nuclear Operating Company (STPNOC) submitted an Overall Integrated Plan (OIP) (Reference 2) in response to the NRC Order. By a letter (Reference 3) dated June 7, 2013, the NRC staff determined that additional information is needed to complete their review of the OIP. The STPNOC response to Reference 3 is provided in the attachment to this letter.

There are no regulatory commitments in this letter.

STI: 33704694

If there are any questions, please contact Ken Taplett at 361-972-8416.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: June 25, 2013

G. T. Powell

G. T. Powell
Site Vice President

Attachment: Response to Request for Additional Information Regarding Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

kjt

cc:

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Response to Request for Additional Information Regarding Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

References:

1. Letter, D. L. Koehl to NRC Document Control Desk, "Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated February 28, 2013 (NOC-AE-13002959) (ML13070A006)
2. Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051) (ST-AE-NOC-12002271) (ML12054A679)
3. NRC Japan Lessons-Learned Project Directorate Interim Staff Guidance JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, August 29, 2012 (ML12221A339)
4. NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1, dated August 2012 (ML122400399)

Reference 1 provided the Overall Integrated Plan (OIP) which the STP Nuclear Operating Company ("STPNOC") will implement for Units 1 and 2 to comply with the requirements of NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Reference 2), NRC Interim Staff Guidance JLD-ISG-2012-003, Revision 0, (Reference 3) and NEI Report 12-02, Revision 1 (Reference 4).

As discussed in Reference 1, any changes to the requirements in NRC JLD-ISG-2012-003 or NEI 12-02 may require relief from the requirements and schedule documented in the OIP.

As provided in the OIP, the Milestones for completing the design and engineering work for Unit 1 are September 2014 and for Unit 2 is December 2013.

The following responses to the request for additional information are based on information developed to date. Any changes to the following information that occur after completing and approving the final design for reliable spent fuel pool instrumentation will be provided in the periodic 6-month status reports submitted to the NRC required by Order EA-12-051.

REQUEST FOR ADDITIONAL INFORMATION

OVERALL INTEGRATED PLAN IN RESPONSE TO

ORDER EA-12-051, "RELIABLE SPENT FUEL POOL INSTRUMENTATION"

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 Introduction

By letter dated February 28, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13070A006), STP Nuclear Operating Company (STPNOC, the licensee), submitted an Overall Integrated Plan (OIP) in response to the March 12, 2012, U.S. Nuclear Regulatory Commission (NRC), Commission Order modifying licenses with regard to requirements for Reliable Spent Fuel Pool (SFP) Instrumentation (Order Number EA-12-051; ADAMS Accession No. ML12054A679) for South Texas Project (STP), Units 1 and 2. The NRC staff endorsed Nuclear Energy Institute (NEI) 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, to Modify Licenses with Regard to Reliable SFP Instrumentation," Revision 1, dated August 2012 (ADAMS Accession No. ML12240A307), with exceptions as documented in Interim Staff Guidance (ISG) 2012-03, "Compliance with Order EA-12-051, Reliable SFP Instrumentation," Revision 0, dated August 29, 2012 (ADAMS Accession No. ML12221A339).

The NRC staff has reviewed the February 28, 2013, response by the licensee and determined that the following request for additional information (RAI) is needed to complete its technical review. Please provide the response to the following RAIs.

2.0 Levels of Required Monitoring

The OIP states, in part, that

> **LEVEL 1: Level adequate to support operation of the normal fuel pool cooling system.**
>
> Plant El. 64 ft. 2 in or 24 ft. 4 in. water level above the top of the SFP fuel storage rack.
>
> **LEVEL 2: Level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck.**
>
> Plant El 49 ft. 10 in. or 10 ft. water level above the top of the SFP fuel storage rack.
>
> **LEVEL 3: Level where the fuel remains covered.**
>
> Plant El 40 ft. 4 in. or 6 in. water level above the top of the SFP fuel storage rack.
>
> ...The installation of the SFPLI [spent fuel pool level instrumentation] sensor will be such that it will measure as close as possible to the top of the SFP fuel rack. Indicated level on either the Primary or Backup Instrument Channel of greater than ½ ft. above the top of SFP fuel storage racks based upon the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, satisfies the NEI 12-02 [Rev.2] requirement of ±1 ft. from the top of the fuel rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the SFP fuel storage rack.

NRC RAI-1a

Please provide the following:

a) **For Level 1, please specify how the identified location represents the HIGHER of the two points described in the NEI 12-02 guidance for this level.**

STPNOC Response

LEVEL 1 represents the HIGHER of either the level at which reliable suction loss to the spent fuel pool (SFP) cooling pump occurs, or the required net positive suction head (NPSH) of the SFP cooling pump

Required NPSH.

The SFP cooling pumps were analyzed for the conservative worst case operation of the SFP cooling pumps. Maximum values for line resistance, fluid temperature, suction flow and static head were used to calculate NPSH parameters for both required and available NPSH ($NPSH_R$ and $NPSH_A$). It was determined that for the worst case scenario, the $NPSH_A$ was significantly higher than $NPSH_R$. The $NPSH_A$ was calculated to be 42.67 feet (ft) and $NPSH_R$ was calculated to be 18.75 ft.

Therefore, $NPSH_R$ is not the determining value to be used for LEVEL 1.

Loss of reliable suction to SFP cooling pumps.

For the purposes of the OIP, this level is conservatively placed at Plant elevation (El.) 64 ft, 2 inches (in). This level provides for more than one foot of water above the top of the SFP cooling pump suction inlet flange (the centerline of the 14 inch suction line flange to the pump is at Plant El. 62 ft. 6 in.) which will be sufficient for NPSH.

A vortex calculation shows 0.134% air entrainment at an elevation one foot above the suction pipe centerline. Level 1 at 64 ft. 2 in. is adequate for normal SFP cooling system operation. Therefore, Level 1 represents the HIGHER of the two points described in the NEI 12-02 guidance.

NRC RAI-1b

b) A clearly labeled sketch depicting the elevation view of the proposed typical mounting arrangement for the portions of instrument channel consisting of permanent measurement channel equipment (e.g., fixed level sensors and/or stilling wells, and mounting brackets). Please indicate on this sketch the datum values representing Level 1, Level 2, and Level 3 as well as the top of the fuel. Indicate on this sketch the portion of the level sensor measurement range that is sensitive to measurement of the fuel pool level, with respect to the Level 1, Level 2, and Level 3 datum points.

STPNOC Response

See Figures 1 and 2 of this Attachment.

3.0 Instrumentation and Design Features

3.1 Instruments and Arrangement

The OIP states, in part, that

> Both the Primary and Backup Instrument Channels will utilize permanently-installed instruments....
>
> The Primary Instrument Channel level sensing components will be located in the northeast corner of the Spent Fuel Pool, as shown in Attachment 1....

> The Backup Instrument Channel level sensing components will be located in the northwest corner of the Spent Fuel Pool, as shown in Attachment 1....
>
> The current Plan is to mount the supporting electronic instruments outside of the spent fuel pool area, to provide a more benign radiation and environmental conditions, and also provide for reasonable and accessible locations for operators.
>
> SFP Primary and Backup Channel Level Instruments are currently planned to be located in Radwaste Control Room of the Mechanical Auxiliary Building (MAB); however, STPNOC is still evaluating other possible locations (i.e. relay room).

<u>NRC RAI-2</u>

Please provide a clearly labeled sketch or marked-up plant drawing of the plan view of the SFP area, depicting the SFP inside dimensions, the planned locations/ placement of the primary and back-up SFP level sensor, and the proposed routing of the cables that will extend from the sensors toward the location of the read-out/display device.

<u>STPNOC Response</u>

See Figure 3 of this Attachment.

3.2 Mounting

The OIP states, in part, that

> Consideration will be given to the maximum seismic ground motion that occurs at the installation location for the permanently installed equipment which is documented in the UFSAR [Updated Final Safety Analysis Report] Section 3.7. The mountings shall be designed consistent with the highest safety or seismic classification of the SFP. The level sensors will be mounted on seismically qualified brackets.

<u>NRC RAI-3a</u>

Please provide the following:

a) **The design criteria that will be used to estimate the total loading on the mounting device(s), including static weight loads and dynamic loads. Please describe the methodology that will be used to estimate the total loading, inclusive of design basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing or other effects that could accompany such seismic forces.**

Figure 1
Elevation View Orientation of Monitored Levels

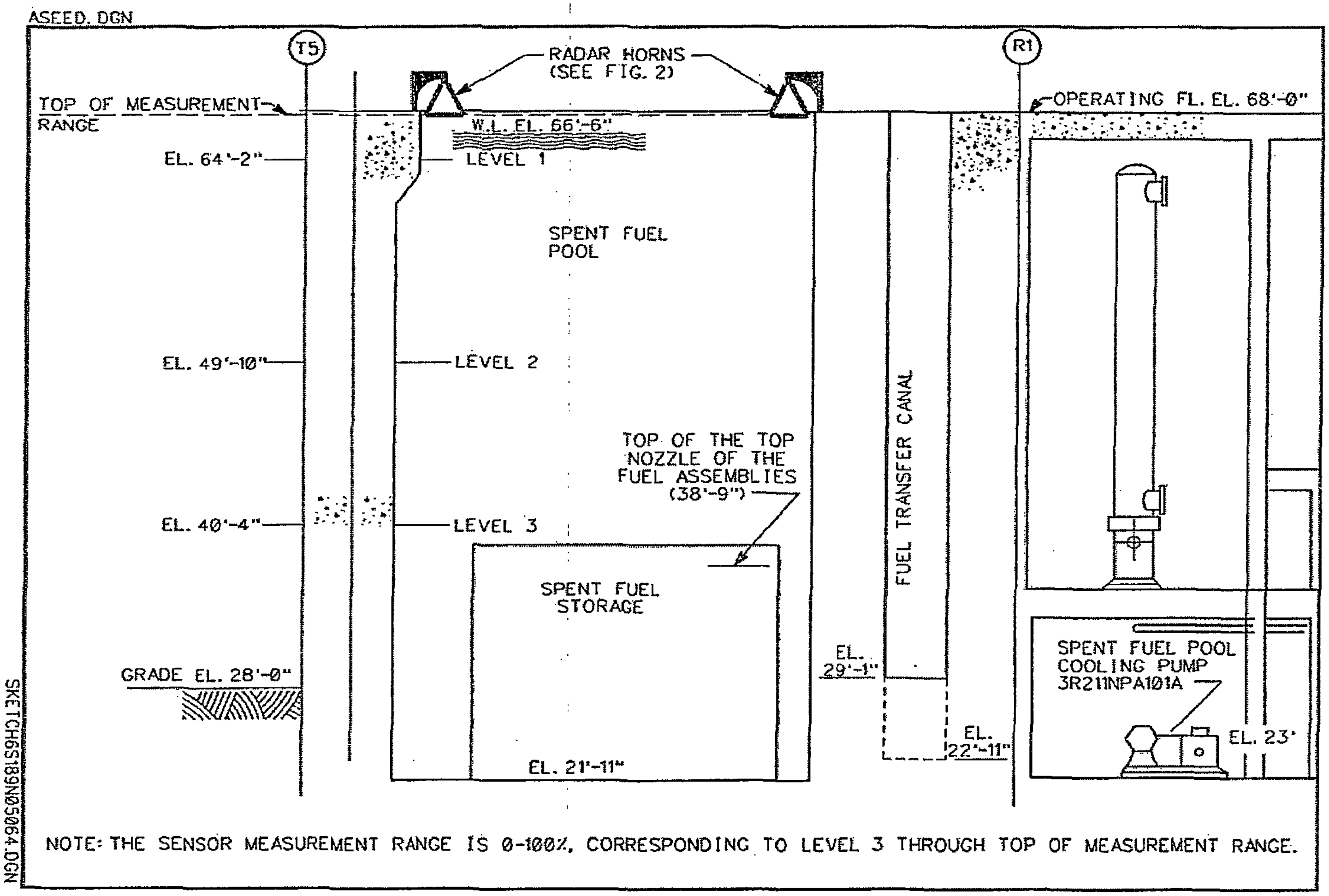
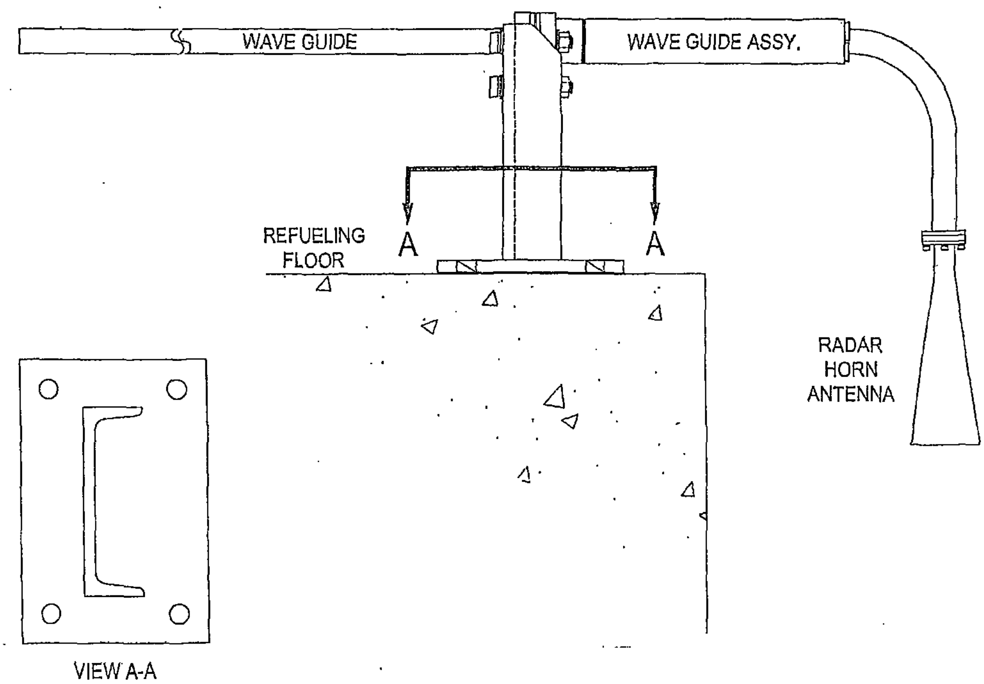


Figure 2
Proposed Mounting Arrangement



RG1

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC. NO.STPNOC013-CALC-002
		REV. 1

Title:	Radiological Release Thresholds for Emergency Action Levels	Client: South Texas Project
		Project: STPNOC013

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions that require confirmation? (If YES, Identify the assumptions)		✓
2	Does this calculation serve as an "Alternate Calculation"? (If YES, Identify the design verified Calculation.) **Design Verified Calculation No.**		✓
3	Does this calculation Supersede an existing Calculation? (If YES, identify the superseded Calculation.) **Superseded Calculation No.**		✓

Scope of Revision: Incorporate decay time of one hour from shutdown as well as migration into Attachment 1. Change statement of no decay in the STAMPEDE runs.

Revision Impact on Results: Values calculated in Attachment 1 decreased and have become the limiting values.

~~Study Calculation~~ ☐ ~~Final Calculation~~ ☑

Safety-Related ☑ Non-Safety Related ☐

(Print Name and Sign)

Originator:	Caleb Trainor	**Date:** 3/21/2014
Design Verifier:	Chad Cramer	**Date:** 3/21/14
Approver:	Marvin Morris	**Date:** 3/21/14

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 2 of 49

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	03/03/2014	Initial Issue
1	3/21/2014	Resolve inconsistency in decay times for the two calculations

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
1-11	1		

ATTACHMENT REVISION STATUS

ATTACHMENT NO.	PAGE NO.	REVISION NO.	ATTACHMENT NO.	PAGE NO.	REVISION NO.
1	12-24	1			
2	25-31	1			
3	32-49	1			

ENERCON Excellence—Every project. Every day.	CALCULATION DESIGN VERIFICATION CHECKLIST	CALC. NO. STPNOC013-CALC-002
		REV. 1
		PAGE NO. 3 of 49

Item	CHECKLIST ITEMS	Yes	No	N/A
1	**Design Inputs** - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis and incorporated in the calculation?	✓		
2	**Assumptions** – Were the assumptions reasonable and adequately described, justified and/or verified, and documented?	✓		
3	**Quality Assurance** – Were the appropriate QA classification and requirements assigned to the calculation?	✓		
4	**Codes, Standard and Regulatory Requirements** – Were the applicable codes, standards and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied?	✓		
5	**Construction and Operating Experience** – Have applicable construction and operating experience been considered?			✓
6	**Interfaces** – Have the design interface requirements been satisfied, including interactions with other calculations?	✓		
7	**Methods** – Was the calculation methodology appropriate and properly applied to satisfy the calculation objective?	✓		
8	**Design Outputs** – Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives and are the results reasonable compared to the inputs?	✓		
9	**Radiation Exposure** – Has the calculation properly considered radiation exposure to the public and plant personnel?	✓		
10	**Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	✓		
11	**Computer Software** – Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?			✓

COMMENTS:
None

	(Print Name and Sign)	
Design Verifier:	Chad Cramer [signature]	**Date:** 3/21/14
Others:		**Date:**

ENERCON Excellence—Every project. Every day.	CALCULATION DESIGN VERIFICATION PLAN AND SUMMARY SHEET	CALC. NO. STPNOC013-CALC-002
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Calculation Design Verification Plan:

Calculation shall be verified by comparing the documented input with the references and checking the validity of the references for the intended use. As necessary, assumptions shall be evaluated and verified to determine if they are based on sound engineering principles and practices. Verify the applicable methodology, inputs, results, and conclusions.

(Print Name and Sign for Approval – mark "N/A" if not required)

Approver: Marvin Morris Marvin Morris	**Date** 3/21/14

Calculation Design Verification Summary:

Design inputs, assumptions, methodology, results and conclusions were evaluated/verified and found to be acceptable. All comments have been incorporated.

Based On The Above Summary, The Calculation Is Determined To Be Acceptable.

(Print Name and Sign)

Design Verifier: Chad Cramer	**Date:** 3/21/14
Others:	**Date:**

Table of Contents

1.0 OBJECTIVE/SCOPE

The purpose of this calculation is to determine the Emergency Action Level (EAL) threshold values of a radiological release from the Unit Vent or Main Steam Lines for an Unusual Event, Alert, Site Area Emergency, or General Emergency. The calculated threshold values are to be included in the STP EAL Technical Basis document, which implements the new NEI 99-01, Revision 6, Emergency Action Level Scheme and will be submitted to the NRC for approval. Upon NRC approval, the values will be used in 0ERP01-ZV-IN01, Revision 10, Emergency Classification.

Both a hand calculation and the South Texas Assessment Model Projecting Emergency Dose Evaluation (STAMPEDE) software program were used to generate the results. The hand calculation is included as Attachment 1.

Revision 1 of this calculation incorporated decay for a release taking place one hour after reactor shutdown. This was done to create continuity between the two methodologies present.

2.0 SUMMARY OF RESULTS

The results of the calculations for the radiation monitors specified in the STP EAL Basis Document and are listed in Table 2.1, below.

Table 2.1: Summary of Calculation Results

Emergency Action Level		RT-8010B, Unit Vent (μCi/sec)	RT-8046 through 8049, Main Steam Lines (μCi/cc)
RU1	**Unusual Event**		
	Hand Calculation	1.40E+05	5.00E-02
	STAMPEDE	N/A*	N/A*
RA1	**Alert**		
	Hand Calculation	1.57E+06	4.10E+00
	STAMPEDE	2.50E+06	4.50E+00
RS1	**Site Area Emergency**		
	Hand Calculation	1.57E+07	4.105E+01
	STAMPEDE	2.50E+07	4.50E+01
RG1	**General Emergency**		
	Hand Calculation	1.57E+08	4.10E+02
	STAMPEDE	2.50E+08	4.50E+02

*STAMPEDE was not used to determine the threshold for RU1. Reference 5.10 indicates that the ODCM methodology should be used to determine the threshold value.

This calculation will be used to establish the threshold values for abnormal radiation based emergencies in the STP EAL Technical Basis document.

3.0 METHOD OF ANALYSIS

Previously, STAMPEDE was used to calculate the Emergency Action Level threshold values for effluent releases. A hand calculation will verify the STAMPEDE calculations. The hand calculation is described in Attachment 1 of this document STAMPEDE conforms to the requirements of STP Procedure 0PGP07-ZA-0014, Software Quality Assurance Program. STAMPEDE was run at STP on an STP computer and under the supervision of an ENERCON employee with access to the STP site as a critical worker.

4.0 INPUTS

4.1 Per NEI 99-01, Revision 6, Initiating condition AU1, EAL 1, the Notice of Unusual Event initiating condition is a release of gaseous or liquid radioactivity greater than two times the ODCM limit for sixty minutes or longer (Reference 5.10).

4.2 The ODCM offsite dose limit is exceeded if the Xe-133 release concentration exceeds 7.41E-04 μCi/cc (Reference 5.6).

4.3 The Unit Vent flow rate is 9.4E+07 cc/sec (Reference 5.1).

4.4 The main steam line pressure and PORV choke flow rate are 1285 psig and 1.05E+06 lbm/hr, respectively (Reference 5.2).

~~4.5 The specific volume of saturated steam at 1285 psig is 0.338 ft^3/lbm (Reference 5.3).~~

4.6 The release concentration is varied to find the release concentration which correlates to each emergency action level. Emergency action levels are taken from NEI 99-01, Revision 6 (Reference 5.10) for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The doses at the Site Boundary that correlate to the threshold concentrations are listed in Table 4.1.

Table 4.1 EAL Offsite Dose Initiating Conditions

	Alert	Site Area	General
TEDE	10 mrem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

5.0 REFERENCES

~~5.1 Offsite Dose Calculation Manual, Revision 17, March 2011~~

5.2 Main Steam PORV Capacity Verification MC05591, Revision: 1

5.3 NIST Steam Tables, 2011

5.4 0ERP01-ZV-IN01, Emergency Classification Draft Revision 10

5.5 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21

5.6 STP Calculation NC-9012, CRMS Rad Monitor Setpoints, Revision 7

5.7 STP Calculation NC-9011, Revision 2

5.8 STAMPEDE Computer Program, Revision 7.0.3.3

5.9 STAMPEDE User's Manual

5.10 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors

5.11 0PGP07-ZA-0014 Quality Assurance Program

5.12 ITWMS Call Number 1000010987 Design Document, Revision 0

6.0 ASSUMPTIONS

6.1 Unit Vent Noble Gas Monitor

To be consistent with the ODCM methodology, the unit vent release is assumed to be entirely Xe-133. The unit vent noble gas monitor is calibrated to Xe-133 (Reference 5.1) therefore; the monitor reading accurately reflects the Xe-133 release magnitude.

To be consistent with ODCM methodology, the main steam line release is assumed to be entirely Xe-133. The noble gas monitor is calibrated to Xe-133 (Reference 5.6).

6.2 Release Duration

Per Reference 5.10, Sections IC AA1, AS1, and AG1 developer notes, the release should be assumed to last one hour.

6.3 Release following Reactor Shutdown

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

6.4 Source Term

Per Reference 5.1, any unit vent release with increased RCS activity and no core melt should be calculated using the gap inventory. Therefore, the gap inventory is used for all unit vent releases.

Per Reference 5.1, for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of noble gases plus 0.2% iodine. A steam generator tube rupture is the only scenario which would create significant offsite doses through a main steam line release.

6.5 Default STAMPEDE Input Values

Reference 5.10 developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. STAMPEDE is used for emergency dose assessment. Per Reference 5.1, when actual meteorology is not available, the default STAMPEDE values should be used. Had the ODCM methodology been used, the 500 hour peak χ/Q value would be used which is less conservative than the χ/Q value produced by STAMPEDE using default meteorological conditions. Therefore, the use of STAMPEDE default values provides a more conservative estimate than that of the alternative method outlined in Reference 5.10.

6.6 Average Effluent Concentration (χ/Q)

The same χ/Q is used for the unit vent and main steam line release. Reference 5.1 applies the same unit vent χ/Q to Units 1 and 2 which would also be applicable to the main steam line. All releases are considered to be ground level releases.

7.0 STAMPEDE CALCULATIONS

7.1 Unusual Event – RU1

7.1.1 Unit Vent Monitor

AU1 recommends declaring an unusual event due to a release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Reference 5.10).

STP sets the ODCM limit at 7.41E-04 μCi/cc (Reference 5.6, pg. 16). Two times the limit would be 1.48E-03 μCi/cc. The threshold is listed in μCi/sec so that variations in flow rate do not change the threshold. The normal flow rate from the unit vent is 9.4E+07 cc/sec (Reference 5.1).

$$Concentration\left(\frac{\mu Ci}{cc}\right) * Flow\ Rate\left(\frac{cc}{sec}\right) = Release\ Rate\left(\frac{\mu Ci}{sec}\right)$$

$$(1.48E-03)\left(\frac{\mu Ci}{cc}\right) * (9.4\text{E}+07)\left(\frac{cc}{sec}\right) = 1.4E+05\left(\frac{\mu Ci}{sec}\right)$$

Equation 7.1.1.1

7.1.2 Main Steam Line Monitor

The ODCM does not calculate a release corresponding to allowable limits for the main steam line monitors. Since the unit vent release calculated in the ODCM was assumed to be primarily Xe-133, the assumption is made in the ODCM that other noble gases and iodine may be ignored in the calculation. This assumption is equally justifiable for the main steam line and the same limiting release will be used.

The magnitude of the release calculated for the unit vent Unusual Event applies to the main steam lines as well. The main steam line PORV's will create a dose exceeding two times the ODCM limit by releasing 1.4E+05 μCi/sec of activity which is equivalent to the release from the unit vent.

The steam lines hold saturated steam at 1285 psig, per Reference 5.2, which has a specific volume of 0.338 ft^3/lbm (Reference 5.3). The PORVs will release the steam at 1.05E+06 lbm/hr per Reference 5.2. This creates a set flow rate of steam from the main steam lines of 2.79E+06 cc/sec as shown below.

$$F\left(\frac{lbm}{hr}\right) * Density\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\ \left(\frac{sec}{hr}\right) = \frac{cc}{sec}$$

$$1.05E+06\left(\frac{lbm}{hr}\right) * 0.338\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\ \left(\frac{sec}{hr}\right) = 2.79E+06\frac{cc}{sec}$$

Equation 7.1.2.1

Since the flow rate is set, the concentration will determine the limit. Equation 7.1.1.1 solves for the limiting concentration of 5.00E-02 µCi/cc as shown below.

$$\frac{Limiting\ Release\left(\frac{\mu Ci}{sec}\right)}{Release\ Rate\left(\frac{cc}{sec}\right)} = Limiting\ Concentration\left(\frac{\mu Ci}{cc}\right)$$

$$\frac{1.40 * 10^5\left(\frac{\mu Ci}{sec}\right)}{2.79 * 10^6\left(\frac{cc}{sec}\right)} = 5.00E-02\left(\frac{\mu Ci}{cc}\right)$$

Equation 7.1.2.2

7.2 Alert, Site Area and General Emergencies – RA1, RS1, RG1

7.2.1 Unit Vent Monitor

Input

The Alert EAL is set to 10 mrem TEDE and 50 mrem Thyroid CDE per Reference 5.10. The emergency offsite dose calculation software STAMPEDE was used to calculate the release which corresponds to this dose. A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for the calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Gap inventory source term
- Default STAMPEDE input values
  - Windspeed = 13.2 mph
  - Stability class D

Results

Given a monitored unit vent release of 2.50E+06 µCi/sec, the Thyroid CDE is 51 mrem/hr at the closest portion of the site boundary and the EAL Initiating Condition is exceeded.

Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 2.50E+07 and 2.50E+08 µCi/sec for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

The input and output files can be found at the end of this document in Attachment 3.

7.2.2 Main Steam Line Monitor

Input

A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for this calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Noble gas + iodine with 0.2% iodine source term
- Default STAMPEDE input values
  - Windspeed = 13.2 mph
  - Stability class D

Results

Given a monitored main steam line release of 4.5 μCi/cc, the Thyroid CDE is 50 mrem/hr and the EAL Initiating Condition is exceeded.

The input and output files can be found at the end of this document in Attachment 3.

7.3 Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 45 and 450 μCi/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

Attachment 1 – Hand Calculations

1.0 OBJECTIVE/SCOPE

Each release calculated using STAMPEDE in the main document is calculated by hand in this attachment and the results compared to STAMPEDE.

2.0 SUMMARY OF RESULTS

Table 2.1 is displayed again below showing the results from all the calculations. The minor difference is due to STAMPEDE using decay factors over a one hour period after shutdown. This also accounts for the change in the limiting dose being TEDE in the hand calculations and Thyroid CDE in the STAMPEDE calculations. The accuracy of the hand calculation is considered sufficient and recommended for use in Emergency Action Levels.

Table 2.1 Results

Emergency Action Level		RT-8010b, Unit Vent (µCi/sec)	RT-8046 through 8049, Main Steam Line (µCi/cc)
RU1	**Unusual Event**		
	Hand Calculation	1.40E+05	5.00E-02
	STAMPEDE	N/A	N/A
RA1	**Alert**		
	Hand Calculation	1.57E+06	3.90E+00
	STAMPEDE	2.50E+06	4.50E+00
RS1	**Site Area Emergency**		
	Hand Calculation	1.57E+07	3.90E+01
	STAMPEDE	2.50E+07	4.50E+01
RG1	**General Emergency**		
	Hand Calculation	1.57E+08	3.90E+02
	STAMPEDE	2.50E+08	4.50E+02

3.0 METHOD OF ANALYSIS

Using the limiting dose at the site boundary, the release is back calculated using atmospheric dispersion models. The X/Q value used is calculated from Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power* Plants. Rather than using the most conservative meteorology, average meteorological conditions are used as inputs to most closely agree with STP emergency dose assessment methodology per the ODCM and STAMPEDE. Assumed nuclide inventories are taken from Reference 5.4. The dose conversion factors are taken from Reference 5.2. A release concentration is used to find an initial projected dose at the Site Boundary. Using the projected dose at the Site Boundary, the release concentration is scaled to find the limiting dose for each EAL.

4.0 INPUTS

- The Unit Vent flow rate is taken from the Offsite Dose Calculation Manual; Revision 17, March 2011 and is 9.44E+07 cc/sec.
- The main steam line pressure and PORV choke flow rate were taken from Reference 5.5 and are 1285 psig and 1.05E+06 lbm/hr respectively.
- The specific volume of saturated steam at this pressure is taken from the NIST steam tables and is 0.338 ft^3/lbm.
- The release concentration is varied to find the release concentration which correlates to each emergency action level dose. Emergency action level doses are taken from NEI 99-01 Revision 6 for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The limiting doses are listed in Table 4.1. NEI 990-01 Revision 6 states that these values are based on fractions of the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) and the General Emergency represents the protective action values recommended by the EPA.

Table 4.1 EAL Thresholds

	Alert	Site Area	General
TEDE	10 mrem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

- A release lasting one hour is selected per NEI 99-01 Revision 6 developer notes.
- Atmospheric dispersion factors are calculated per Regulatory Guide 1.145 (Reference 5.1). The reactor building dimensions used as inputs for this calculation are taken from Reference 5.13.
- Nuclide inventories are taken from TGX/THX 3-1, (Reference 5.4) which is the source document for the nuclide inventories used in STAMPEDE. The release inventories are a gap release and noble gases plus 0.2% iodine which are listed below. Each nuclide inventory was normalized to one so it could be scaled to various release activities.

Table 4.2 Gap Inventory

Nuclide	Activity (µCi/cc)	Normalized	Nuclide	Activity (µCi/cc)	Normalized
I-131	1.10E+05	1.12E-03	Xe-135	5.50E+06	5.62E-02
I-132	1.50E+05	1.53E-03	Xe-137	1.90E+07	1.94E-01
I-133	2.20E+05	2.25E-03	Xe-138	1.80E+07	1.84E-01
I-134	2.40E+05	2.45E-03	Cs-134	3.70E+01	3.78E-07
I-135	2.00E+05	2.05E-03	Cs-137	2.90E+01	2.97E-07
Kr-83m	1.30E+06	1.33E-02	Te132	4.80E+00	4.91E-08
Kr-85m	2.90E+06	2.97E-02	Mo99	1.22E+01	1.25E-07
Kr-85	3.70E+05	3.78E-03	Ru103	8.80E-03	9.00E-11
Kr-87	5.50E+06	5.62E-02	Ru106	2.90E-03	2.97E-11
Kr-88	7.80E+06	7.98E-02	Zr95	1.10E-02	1.12E-10
Kr-89	9.50E+06	9.72E-02	La140	1.90E-02	1.94E-10
Xe-131m	1.10E+05	1.12E-03	Ce144	7.40E-03	7.57E-11
Xe-133m	6.80E+05	6.95E-03	Ce-141	1.00E-02	1.02E-10
Xe-133	2.20E+07	2.25E-01	Sr89	6.40E-02	6.55E-10
Xe-135m	4.20E+06	4.30E-02	Sr90	3.20E-03	3.27E-11

Table 4.3 Noble Gases+0.2% Iodine Inventory

Nuclide	Inventory	Normalized
I-131	6.10E-02	2.26E-04
I-132	8.61E-02	3.19E-04
I-133	1.00E-01	3.72E-04
I-134	1.86E-02	6.92E-05
I-135	2.73E-01	1.01E-03
Xe-131m	2.80E+00	1.04E-02
Xe-133	2.40E+02	8.90E-01
Xe-133m	4.20E+00	1.56E-02
Xe-135	7.60E+00	2.82E-02
Xe-135m	4.00E-01	1.48E-03
Xe-137	1.60E-01	5.93E-04
Xe-138	5.80E-01	2.15E-03
Kr-83m	3.70E-01	1.37E-03
Kr-85	7.60E+00	2.82E-02
Kr-85m	1.50E+00	5.56E-03
Kr-87	9.80E-01	3.63E-03
Kr-88	2.80E+00	1.04E-02
Kr-89	8.40E-02	3.12E-04

- The dose conversion factors taken from EPA 400R92001 (Reference 5.2) are listed in Tables 4.4 and 4.5 below.

Table 4.4 TEDE Dose Conversion Factors

Nuclide	Dose Conversion Factor (rem per uCi*hr/cc)	Nuclide	Dose Conversion Factor (rem per uCi*hr/cc)
I-131	5.30E+04	Xe-135	1.40E+02
I-132	4.90E+03	Xe-137	1.10E+02
I-133	1.50E+04	Xe-138	7.20E+02
I-134	3.10E+03	Cs-134	6.30E+04
I-135	8.10E+03	Cs-137	4.10E+04
Kr-83m		Te132	1.20E+04
Kr-85m	9.30E+01	Mo99	5.20E+03
Kr-85	1.30E+00	Ru103	1.30E+04
Kr-87	5.10E+02	Ru106	5.70E+05
Kr-88	1.30E+03	Zr95	3.20E+04
Kr-89	1.20E+03	La140	1.10E+04
Xe-131m	4.9	Ce144	4.50E+05
Xe-133m	1.70E+01	Ce-141	1.10E+04
Xe-133	2.00E+01	Sr89	5.00E+04
Xe-135m	2.50E+02	Sr90	1.60E+06

Table 4.5 Thyroid CDE Dose Conversion Factors

Nuclide	Thyroid CDE DCF (rem per uCi*hr/cc)
I-131	1.30E+06
I-132	7.70E+03
I-133	2.20E+05
I-134	1.30E+03
I-135	3.80E+04

- The unit vent noble gas monitor energy efficiency by nuclide is taken from Offsite Dose Calculation Manual (Reference 5.3). The values are relative to Xe-133 efficiency since the monitor is calibrated to Xe-133. Table 4.6 displays the energy efficiency by nuclide relative to Xe-133.

Table 4.6 Energy Efficiency Relative to Xe-133

Nuclide	Efficiency Relative to Xe-133 (uCi/cc)$_{equivalent}$
Kr-83m	
Kr-85m	1.9
Kr-85	2.4
Kr-87	2.8
Kr-88	2.3
Kr-89	2.8
Xe-131m	0.015
Xe-133m	0.14
Xe-133	1
Xe-135m	0.042
Xe-135	2.5
Xe-137	2.8
Xe-138	2.8

*There is no relative efficiency available for Kr-83m. Assumption 6.4 further justifies the omission.

Table 4.7 Nuclide Half Lives

Nuclide	Half Life (hr)	Nuclide	Half Life (hr)
I-131	1.93E+02	Xe-135	9.08E+00
I-132	2.38E+00	Xe-137	6.38E-02
I-133	2.03E+01	Xe-138	2.36E-01
I-134	8.77E-01	Cs-134	1.80E+04
I-135	6.61E+00	Cs-137	2.60E+05
Kr-83m	1.83E+00	Te132	7.79E+01
Kr-85m	4.48E+00	Mo99	6.62E+01
Kr-85	9.40E+04	Ru103	9.44E+02
Kr-87	1.27E+00	Ru106	8.84E+03
Kr-88	2.84E+00	Zr95	1.55E+03
Kr-89	5.10E-02	La140	4.03E+01
Xe-131m	2.83E+02	Ce144	6.82E+03
Xe-133m	5.42E+01	Ce-141	7.77E+02
Xe-133	1.27E+02	Sr89	1.21E+03
Xe-135m	2.60E-01	Sr90	2.50E+05

- The half-lives are taken from Reference 5.15 which lists the input data used by STAMPEDE.

5.0 REFERENCES

5.1 Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.
5.2 EPA 400R92001, Manual of Protective Action Guides and Protective actions for Nuclear Incidents, Revision 1, May 1992.
5.3 Offsite Dose Calculation Manual, Revision 17, March 2011.
5.4 TGX/THX 3-1, Revision 5, Westinghouse Radiation Analysis Manual.
5.5 MC05591, Main Steam PORV Capacity Verification, Revision 1.
5.6 NIST Steam Tables, 2011.
5.7 0ERP01-ZV-IN01, Emergency Classification, Revision 10.
5.8 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21.
5.9 STP Calculation NC-9012, Process and Effluent Radiation Monitor Set Points, Revision 7
5.10 STP Calculation NC-9011, CRMS Rad Monitor Setpoints, Revision 2.
5.11 STAMPEDE Computer Program, Revision 7.0.3.3.
5.12 STAMPEDE User's Manual
5.13 STP Drawing 6C189N5007, General Arrangement Reactor Containment Building, Revision 6
5.14 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
5.15 ITWMS Call Number 1000010987 Design Document, Revision 0

6.0 ASSUMPTIONS

6.1 Release lasts for one hour

Per NEI 99-01 (Reference 5.14), IC AA1, AS1, AG1 developer notes, the release should be assumed to last one hour.

For this to be true for the main steam line, it is assumed that the PORV is open for one hour. To calculate the most limiting case, it is assumed that the maximum flow possible is being released from the PORV.

6.2 Nuclide mix

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) any unit vent release with increased RCS activity and no core melt should be calculated using a gap inventory. It is conservative to assume an increased RCS activity and not within the intended scope of the relevant initiating conditions to assume core melt. Therefore, a gap inventory is used for all unit vent releases.

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of 100 percent noble gases plus 0.2 percent iodine. Since a steam generator tube rupture releasing through the PORVs is the only steam generator tube rupture scenario which would create offsite doses large enough to meet or exceed the EALs, this assumption is made.

6.3 Atmospheric Dispersion

NEI 99-01 (Reference 5.14) developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8), when actual meteorology is not available, the default STAMPEDE values should be used. The default STAMPEDE values assume a stability class D for atmospheric dispersion and a windspeed of 13.2 mph. These values were used as inputs for the atmospheric dispersion calculation.

It is clear that STAMPEDE uses the same method for calculating atmospheric dispersion factor (X/Q) outlined in section 7.1.1 of this Attachment. However, STAMPEDE does not follow the same logic in selecting the appropriate result from the three calculations. The STAMPEDE value printed in the results found in attachment 3 is consistent with the largest of the three hand calculated X/Q values. This suggests that STAMPEDE simply selects the largest of the three X/Q values resulting in a much more conservative estimate. This calculation will deviate from the recommendations of Regulatory Guide 1.145 and conform to the methodology STAMPEDE uses.

The close proximity of all release points allows for a single atmospheric dispersion coefficient to be used. This assumption is also made by STAMPEDE.

6.4 Exposure Pathways

The dose conversion factors used in table 4.4 and 4.5 represent a summation of dose conversion factors for external plume exposure, inhalation from the plume, and external exposure from deposition. Because the dose estimations are used for implementing early phase protective actions, conversion factors using limited pathways are appropriate.

The EPA does not provide a dose conversion factor for Kr-83m. Because the PAGs are based on EPA dose calculations, it is appropriate to only use the nuclides for which dose conversion factors are provided. Additionally, Kr-83m represents only 1.33% of the nuclide inventory activity and its exclusion would not significantly affect the final dose.

6.5 The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

Decay is incorporated for one hour from reactor shutdown as well as migration time. Half-lives are taken from Reference 5.15. Migration time is assumed to be the reciprocal of the wind speed.

7.0 HAND CALCULATIONS

7.1 Unit Vent Monitor

7.1.1 X/Q

The atmospheric dispersion factor, X/Q, determines the change in concentration between the unit vent discharge and the dose reception site. This value is based on meteorological conditions and will vary with wind speed and stability class. The ODCM uses the highest annual average X/Q value at the site boundary which is 5.3E-06 sec/m^3. However, for an accident related release STAMPEDE is used rather than the ODCM. STAMPEDE uses real time, user entered, or default meteorological conditions to calculate the X/Q for a specific accident. Default values will be used as inputs into the Regulatory Guide 1.145 method for calculating X/Q as described below. Default values are identified in section 6.0, Atmospheric Dispersion.

For a neutral atmospheric stability class, which is the default in STAMPEDE, X/Q values can be determined through the following set of equations.

$$\frac{X}{Q} = \frac{1}{\overline{U}_{10}\left(\pi\sigma_y\sigma_z + \frac{A}{2}\right)}$$

Equation 7.1.1.1

$$\frac{X}{Q} = \frac{1}{\overline{U}_{10}(3\pi\sigma_y\sigma_z)}$$

Equation 7.1.1.2

$$\frac{X}{Q} = \frac{1}{\overline{U}_{10}\pi\Sigma_y\sigma_z}$$

Equation 7.1.1.3

Where

X/Q = relative concentration (sec/m^3)

π = 3.14159

$\overline{U}_{10}$ = windspeed at 10 meters above plant grade (m/s)

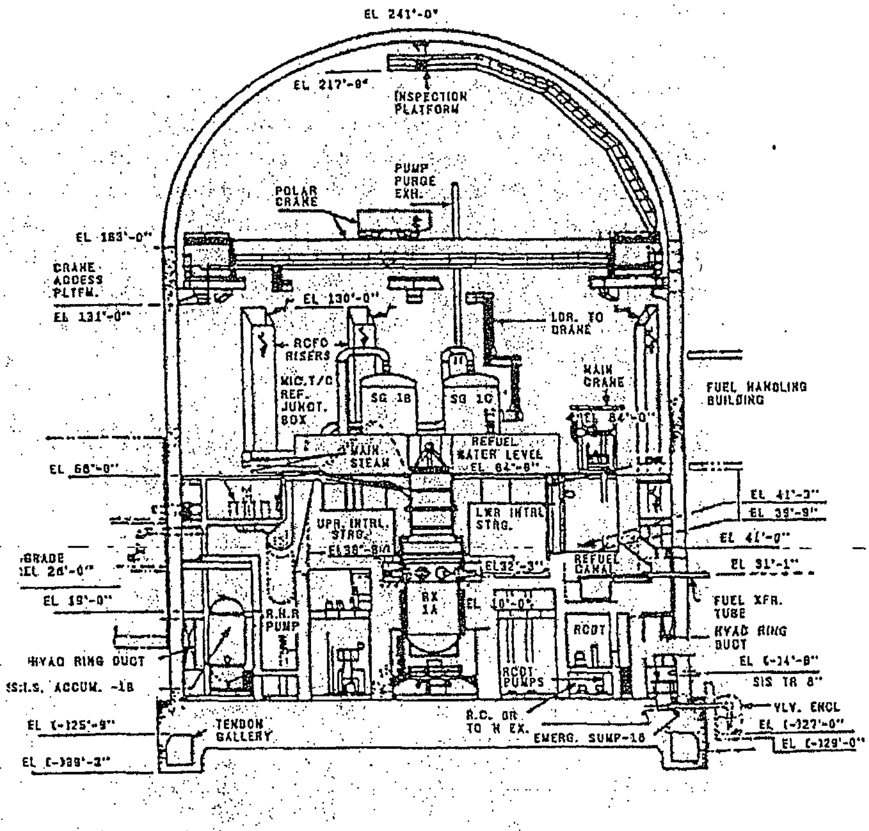
σ_y = lateral plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 1

σ_z = vertical plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 2

Σ_y = $(M-1)\sigma_{y800m} + \sigma_y$ = lateral plume spread with meander and building wake effects (m), a function of atmospheric stability, windspeed $\overline{U}_{10}$, and distance; M is determined from Regulatory Guide 1.145 Figure 3

A = the smallest vertical-plane cross-sectional area of the reactor building (m^2), taken from Reference 5.13 and shown below

Figure 7.1.1.1: Reactor Building Dimensions



Assuming the reactor building cross section to be a perfect rectangle and half sphere, the variables are defined as follows;

$\overline{U}_{10} = 13.2$ mph $= 5.9$ m/s

$\sigma_y \;= 1200$ m

$\sigma_z \;= 4.2$ m

$\Sigma_y \;= (M-1)\sigma_{y800m} + \sigma_y$; M=1 $\rightarrow \sigma_y = 1200$ m

A $= (135' * 158') + \left(\frac{\pi * 79^2}{2}\right) = 31128.37$

The three equations become;

$$\frac{X}{Q} = \frac{1}{5.9\left(\pi 1200 * 4.2 * \frac{31128.37}{2}\right)} = 5.398 * 10^{-6}$$

$$\frac{X}{Q} = \frac{1}{5.9(3\pi * 1200 * 4.2)} = 3.568 * 10^{-6}$$

$$\frac{X}{Q} = \frac{1}{5.9 * \pi * \left[(1-1)\sigma_{y800m} + 1200\right] * 4.2} = 1.07 * 10^{-5}$$

To select the appropriate X/Q value, the first two X/Q values should be compared and the higher value selected. This value is then compared with the third X/Q value and the lower of those two is the appropriate X/Q value. The appropriate X/Q is 5.39E-06 sec/m^3 for default meteorological conditions by the methodology recommended in Regulatory Guide 1.145.

This calculated value is very similar to the ODCM highest average value of 5.3E-06 sec/m^3 which was not selected for use. Additionally, the value shown in the STAMPEDE output file at one mile is 1.032E-05 sec/m^3. This suggests that STAMPEDE uses the same methodology and simply selects the largest atmospheric dispersion value to remain conservative. This methodology will be replicated and 1.07E-05 will be used as the X/Q.

7.1.2 Nuclide Inventory

As previously stated, a gap inventory is appropriate for this problem. The gap inventory is taken from TGX/THX 3-1 (Reference 5.4) which is used as the source term for STAMPEDE inventories. The concentrations were then normalized so they could be scaled to the varying emergency classifications. The values for the normalized inventory can be found in Table 4.2.

7.1.3 Dose Conversion Factors

As stated in NEI99-01 (Reference 5.14) developer notes, the purpose of dose projections is to check if the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) have been exceeded. The dose conversion factors provided by the EPA in EPA 400R92001 are used. These dose conversion factors account for external plume exposure, inhalation from the plume, and external exposure from deposition and are listed Tables 4.4 and 4.5, and taken from tables 5-1, 5-2 in EPA 400R92001 (Reference 5.2).

The EPA does not provide a dose conversion factor for Kr-83m. This nuclide contributes 1.33% of the inventory activity. The lack of this nuclide's contribution to the final dose will not significantly affect the outcome.

7.1.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.1.5 Dose Calculations

The dose rate at the site boundary is calculated using Equation 7.1.5.1.

$$\dot{D} = \frac{X}{Q} F \sum_{i}^{n} C_i * 0.5^{\frac{1.07575}{T_{1/2_i}}} * DCF_i$$

Equation 7.1.5.1

Where

$\dot{D}$ = dose rate per hour at the site boundary

$\frac{X}{Q}$ = atmospheric dispersion coefficient as calculated in section 7.1.1

F = unit vent flow rate

C_i= concentration of nuclide i at the time of shutdown

1.07575 = the total decay time of interest from section 7.1.4

$T_{1/2i}$ = the half-life of nuclide i

DCF_i = the dose conversion factor for nuclide i listed in tables 4.4 and 4.5

The total concentration of the nuclides is varied to find the dose rate of interest. Beginning with an arbitrary release concentration of 1 µCi/cc, the dose rate is calculated. Since the dose is linearly correlated to concentration, the release concentration may be scaled to find the dose rate of interest.

The Alert EAL is 10 mrem TEDE or 50 mrem Thyroid CDE. Using the above method to calculate TEDE with the appropriate conversion factors, a limiting release rate of 2.33E+06 µCi/sec from the unit vent results in 5.7 mrem TEDE. Using the calculated release rate to find Thyroid CDE with the appropriate conversion factors, the same release results in a 50 mrem Thyroid CDE at the site boundary. Thus, 2.33E+06 µCi/sec is the limiting release rate based on the 50 mrem Thyroid CDE EAL initiating condition.

The limiting release rate threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert release rate threshold value.

These calculations can be found in Attachment 2.

7.1.6 Monitor Response

The unit vent noble gas monitor is calibrated to Xe-133. Monitor efficiencies relative to Xe-133 by nuclide are listed in ODCM Table B3-2. To find the monitor reading associated with each limiting release, the noble gas concentrations must be multiplied by the monitor response and summed. Table 4.6 shows the indicated response of the unit vent noble gas monitor by nuclide and Equation 7.1.5.1 shows how the monitor response was calculated.

$$Monitor\ Response = \sum_{i}^{n} C_i * Re_i$$

Equation 7.1.5.1

Where

C_i = concentration of nuclide i (μCi/cc)

Re_i = monitor response to nuclide i $(\mu Ci/cc)_{Xe\text{-}133\ equivalent}$

In the case of an Alert, the 2.33E+06 μCi/sec release rate will read as 1.57E+06 μCi/sec on the monitor. Kr-83m does not have an indicated monitor response coefficient. Because Kr-83m is only 1.34% of the noble gases and does not contribute to the dose calculation, its exclusion is acceptable.

This again is a linear correlation and the SAE and GE scale by factors of 10 and 100 respectively.

These calculations can be found in Attachment 2.

7.2 Main Steam Line Monitors

7.2.1 X/Q

Since the atmospheric dispersion is independent of nuclide inventory or release rate and the close proximity of the releases, the X/Q value will be the same for a main steam line release as it is for a unit vent release. This assumption is also taken by STAMPEDE and outline in Assumption 6.3.

7.2.2 Nuclide Inventory

Per 0ERP01-ZV-TP01, if the release path is the main steam line with a steam generator tube rupture, the nuclide inventory should be 100% noble gas and 0.2% of the iodine from the reactor coolant.

The secondary steam concentration for noble gases and iodine after a steam generator tube rupture are taken from TGX/THX 3-1 (Reference 5.4). Values for the reactor coolant inventory are listed in table 4.3. All of the noble gases are used and the iodine concentration from the coolant inventory is scaled to total 0.2% of iodine in the total coolant inventory. These inventories are then normalized to one. These values are listed in Table 4.3.

7.2.3 Dose Conversion Factors

The dose conversion factors used are found in Tables 4.4 and 4.5, taken from tables 5-1, 5-2 in EPA 400R92001.

7.2.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.2.5 Dose Calculations

Equation 7.1.5.1 applies to the release from the main steam lines. The main steam line flow rate is used instead of the unit vent flow rate for the value *F*. The main steam line flow rate was calculated in Equation 7.1.2.2 of the STAMPEDE CALCULATIONS section of this document as 2.79E+06 cc/sec.

The Alert EAL threshold is 10 mrem TEDE or 50 mrem Thyroid CDE at the site boundary (Table 4.2). Using the method in Equation 7.1.5.1 to calculate TEDE with the appropriate conversion factors, a concentration at time of shutdown of 4.10 μCi/cc would result in 6.89 mrem TEDE at the site boundary if the steam line PORV was open for an hour. Using the same steam line concentration to calculate Thyroid CDE results in 50 mrem Thyroid CDE at the site boundary.

The steam line concentrations at the time of shutdown for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, values for the steam line concentration at time of shutdown are 41.0 and 410 μCi/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE.

These calculations can be found in Attachment 2.

7.2.6 Monitor Response

Because the main steam line monitor is adjacent to the main steam line, significant shielding takes place between the source and monitor. STP calculation NC-9011 Revision 2 calculates a conversion factor for the main steam lines for a noble gas inventory which is incorporated into the monitor readout. No monitor response needs to be calculated.

The concentration of the main steam line one hour after shutdown given a concentration of 4.10 μCi/cc at the time of shutdown is 3.90 μCi/cc. This calculation is also found in Attachment 2. Additionally, the monitor readings for the SAE and GE one hour after shutdown are 39.0 and 390 μCi/cc respectively. These values are the thresholds for the main steam line monitor.

Table A2-1: Unusual Event Emergency Calculations

Unit Vent Limiting Concentration (uCi/cc)	Flow Rate (cc/sec)	Release Rate (uCi/sec)
1.48E-03	9.44E+07	**1.40E+05**
MSL Limiting Release Rate (uCi/sec)	**Flow Rate (cc/sec)**	**Limiting Concentration (uCi/cc)**
1.40E+05	2.79E+06	**5.00E-02**

Table A2-2: Input Values for Calculations

X/Q (s/m^3)	duration (s)	Release Rate (cc/sec)	Release Constant (s*cc/m^3)	Unit Conversion for Release Constant (hr*m^3/s*cc)	Total Release Variable (uCi/cc)	Decay Time (hr)
5.40E-06	3600	9.44E+07	1.83E+06	5.10E-04	1.79E-02	1.07575

Table A2-3: Calculations for Boundary Concentrations and TEDE dose due to Unit Vent Release

Nuclide	Inventory (Ci/cc)	Normalized	Varied concentration (uCi/cc)	Release Constant (hr*cc/cc)	Dose Concentration @ Boundary (uCi*hr/cc)	Half Life (hr)	Decayed Concentration (uCi*hr/cc)	Dose Conversion Factor (rem-cc/uCi-hr)	Dose Contribution (rem)
I-131	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	1.93E+02	2.78E-08	5.30E+04	1.47E-03
I-132	1.50E+05	1.53E-03	3.77E-05	1.01E-03	3.81E-08	2.38E+00	2.79E-08	4.90E+03	1.37E-04
I-133	2.20E+05	2.25E-03	5.55E-05	1.01E-03	5.61E-08	2.03E+01	5.40E-08	1.50E+04	8.11E-04
I-134	2.40E+05	2.45E-03	6.04E-05	1.01E-03	6.11E-08	8.77E-01	2.61E-08	3.10E+03	8.09E-05
I-135	2.00E+05	2.05E-03	5.06E-05	1.01E-03	5.11E-08	6.61E+00	4.56E-08	8.10E+03	3.70E-04
Kr-83m	1.30E+06	1.33E-02	3.28E-04	1.01E-03	3.31E-07	1.83E+00	2.21E-07		0.00E+00
Kr-85m	2.90E+06	2.97E-02	7.33E-04	1.01E-03	7.40E-07	4.48E+00	6.27E-07	9.30E+01	5.83E-05
Kr-85	3.70E+05	3.78E-03	9.33E-05	1.01E-03	9.42E-08	9.40E+04	9.42E-08	1.30E+00	1.22E-07
Kr-87	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	1.27E+00	7.79E-07	5.10E+02	3.97E-04
Kr-88	7.80E+06	7.98E-02	1.97E-03	1.01E-03	1.99E-06	2.84E+00	1.53E-06	1.30E+03	1.99E-03
Kr-89	9.50E+06	9.72E-02	2.40E-03	1.01E-03	2.42E-06	5.10E-02	1.08E-12	1.20E+03	1.30E-09
Xe-131m	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	2.83E+02	2.78E-08	2.50E+02	1.36E-07
Xe-133m	6.80E+05	6.95E-03	1.71E-04	1.01E-03	1.73E-07	5.42E+01	1.71E-07	1.40E+02	2.90E-06
Xe-133	2.20E+07	2.25E-01	5.55E-03	1.01E-03	5.61E-06	1.27E+02	5.57E-06	1.10E+02	1.11E-04
Xe-135m	4.20E+06	4.30E-02	1.06E-03	1.01E-03	1.07E-06	2.60E-01	6.09E-08	7.20E+02	1.52E-05
Xe-135	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	9.08E+00	1.29E-06	5.30E+04	1.81E-04
Xe-137	1.90E+07	1.94E-01	4.79E-03	1.01E-03	4.83E-06	6.38E-02	4.06E-11	4.90E+03	4.47E-09
Xe-138	1.80E+07	1.84E-01	4.54E-03	1.01E-03	4.59E-06	2.36E-01	1.95E-07	1.50E+04	1.40E-04

Nuclide	Inventory	Normalized	Varied concentration	Release Constant	Concentration @ Boundary	Half Life	Decayed Concentration	Dose Conversion Factor	Dose Contribution
	Ci/cc	[illegible]	(uCi/cc)	(hr*cc/cc)	(uCi*hr/cc)	(hr)	(uCi*hr/cc)	rem per uCi*hr/cc	(rem)
Cs-134	3.70E+01	3.78E-07	9.33E-09	1.01E-03	9.42E-12	1.80E+04	9.42E-12	6.30E+04	5.93E-07
Cs-137	2.90E+01	2.97E-07	7.33E-09	1.01E-03	7.40E-12	2.60E+05	7.40E-12	4.10E+04	3.03E-07
Te132	4.80E+00	4.91E-08	1.21E-09	1.01E-03	1.22E-12	7.79E+01	1.21E-12	1.20E+04	1.45E-08
Mo99	1.22E+01	1.25E-07	3.08E-09	1.01E-03	3.11E-12	6.62E+01	3.08E-12	5.20E+03	1.60E-08
Ru103	8.80E-03	9.00E-11	2.22E-12	1.01E-03	2.24E-15	9.44E+02	2.24E-15	1.30E+04	2.91E-11
Ru106	2.90E-03	2.97E-11	7.33E-13	1.01E-03	7.40E-16	8.84E+03	7.40E-16	5.70E+05	4.22E-10
Zr95	1.10E-02	1.12E-10	2.76E-12	1.01E-03	2.79E-15	1.55E+03	2.79E-15	3.20E+04	8.93E-11
La140	1.90E-02	1.94E-10	4.79E-12	1.01E-03	4.83E-15	4.03E+01	4.75E-15	1.10E+04	5.22E-11
Ce144	7.40E-03	7.57E-11	1.87E-12	1.01E-03	1.89E-15	6.82E+03	1.89E-15	4.50E+05	8.49E-10
Ce-141	1.00E-02	1.02E-10	2.52E-12	1.01E-03	2.54E-15	7.77E+02	2.54E-15	1.10E+04	2.79E-11
Sr89	6.40E-02	6.55E-10	1.62E-11	1.01E-03	1.63E-14	1.21E+03	1.63E-14	5.00E+04	8.16E-10
Sr90	3.20E-03	3.27E-11	8.07E-13	1.01E-03	8.15E-16	2.50E+05	8.15E-16	1.60E+06	1.30E-09
								Total TEDE Dose	**5.77E-03**

Table A2-4: Thyroid Dose Calculation for Unit Vent Release

Nuclide	Decayed Concentration	Thyroid DCF	Thyroid Dose
	(uCi/cc)	(rem per uCi-hr/cc)	(rem)
I-131	2.78E-08	1.30E+06	3.61E-02
I-132	2.79E-08	7.70E+03	2.15E-04
I-133	5.40E-08	2.20E+05	1.19E-02
I-134	2.61E-08	1.30E+03	3.39E-05
I-135	4.56E-08	3.80E+04	1.73E-03
		Total Thyroid Dose	5.00E-02

Table A2-5: Unit Vent Monitor Response to Nuclide Inventory

Nuclide	Concentration	Half Life	Concentration After 1hr	Response Coeff	Response	
	(uCi/cc)	Hours	(uCi/cc)	(uCi/cc)	(uCi/cc)	
Kr-83m	3.28E-04	1.83E+00	2.25E-04		0.00E+00	
Kr-85m	7.33E-04	4.48E+00	6.28E-04	1.9	1.19E-03	
Kr-85	9.33E-05	9.40E+04	9.33E-05	2.4	2.24E-04	
Kr-87	1.39E-03	1.27E+00	8.03E-04	2.8	2.25E-03	
Kr-88	1.97E-03	2.84E+00	1.54E-03	2.3	3.55E-03	
Kr-89	2.40E-03	5.10E-02	3.00E-09	2.8	8.40E-09	
Xe-131m	2.76E-05	2.83E+02	2.76E-05	0.015	4.13E-07	
Xe-133m	1.71E-04	5.42E+01	1.69E-04	0.14	2.37E-05	
Xe-133	5.55E-03	1.27E+02	5.52E-03	1	5.52E-03	
Xe-135m	1.06E-03	2.60E-01	7.38E-05	0.042	3.10E-06	
Xe-135	1.39E-03	9.08E+00	1.28E-03	2.5	3.21E-03	
Xe-137	4.79E-03	6.38E-02	9.15E-08	2.8	2.56E-07	
Xe-138	4.54E-03	2.36E-01	2.41E-04	2.8	6.74E-04	
				Monitor Reading:	1.66E-02	1.257E+06
					(uCi/cc)	(uCi/sec)

Table A2-6: Input for Main Steam Line Release Calculation

Choke Flow	Limiting Release	Steam Specific Volume	Volumetric Release	Release Rate unit conversion	Variable concentration	Decay Time
(lbm/hr)	(μCi/sec)	(ft^3/lbm)	(ft^3/hr)	(cc/sec)	(μCi/cc)	(hr)
1.05E+06	3.37E+06	0.338	3.55E+05	2.79E+06	4.05	1.07575

Table A2-7: Calculations for Boundary Concentrations and TEDE dose due to Main Steam Line Release

Nuclide	Steam Inventory	Normalized	Varied Concentration	Release Constant	Concentration @ Boundary	Half life	Decayed Concentration	DCF	Dose Contribution
			(μCi/cc)	(hr-cc/cc)	(μCi-hr/cc)	(hr)	(μCi-hr/cc)	(rem per μCi-hr/cc)	(rem)
I-131	6.10E-02	2.26E-04	9.27E-04	2.9853E-05	2.77E-08	1.93E+02	2.76E-08	5.30E+04	1.46E-03
I-132	8.61E-02	3.19E-04	1.31E-03	2.9853E-05	3.90E-08	2.38E+00	2.85E-08	4.90E+03	1.40E-04
I-133	1.00E-01	3.72E-04	1.53E-03	2.9853E-05	4.55E-08	2.03E+01	4.39E-08	1.50E+04	6.58E-04
I-134	1.86E-02	6.92E-05	2.84E-04	2.9853E-05	8.47E-09	8.77E-01	3.62E-09	3.10E+03	1.12E-05
I-135	2.73E-01	1.01E-03	4.14E-03	2.9853E-05	1.24E-07	6.61E+00	1.10E-07	8.10E+03	8.95E-04
Xe-131m	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.83E+02	1.27E-06	4.90E+00	6.22E-06
Xe-133	2.40E+02	8.90E-01	3.65E+00	2.9853E-05	1.09E-04	5.42E+01	1.07E-04	2.00E+01	2.15E-03
Xe-133m	4.20E+00	1.56E-02	6.40E-02	2.9853E-05	1.91E-06	1.27E+02	1.90E-06	1.70E+01	3.23E-05
Xe-135	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	2.60E-01	1.96E-07	1.40E+02	2.75E-05
Xe-135m	4.00E-01	1.48E-03	6.07E-03	2.9853E-05	1.81E-07	9.08E+00	1.67E-07	2.50E+02	4.17E-05
Xe-137	1.60E-01	5.93E-04	2.43E-03	2.9853E-05	7.26E-08	6.38E-02	6.10E-13	1.40E+02	8.53E-11
Xe-138	5.80E-01	2.15E-03	8.82E-03	2.9853E-05	2.63E-07	2.36E-01	1.12E-08	7.20E+02	8.04E-06
Kr-83m	3.70E-01	1.37E-03	5.62E-03	2.9853E-05	1.68E-07	1.83E+00	1.12E-07		0.00E+00
Kr-85	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	4.48E+00	2.92E-06	1.30E+00	3.80E-06
Kr-85m	1.50E+00	5.56E-03	2.28E-02	2.9853E-05	6.81E-07	9.40E+04	6.81E-07	9.30E+01	6.33E-05
Kr-87	9.80E-01	3.63E-03	1.49E-02	2.9853E-05	4.44E-07	1.27E+00	2.47E-07	5.10E+02	1.26E-04
Kr-88	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.84E+00	9.79E-07	1.30E+03	1.27E-03
Kr-89	8.40E-02	3.12E-04	1.28E-03	2.9853E-05	3.82E-08	5.10E-02	1.71E-14	1.20E+03	2.05E-11
								Total Dose	**6.89E-03**

*Release Constant = X/Q * duration * release rate

Table A2-8: Main Steam Line Release Thyroid Dose Calculation

Nuclide	Concentration @ Boundary	Thyroid DCF	Thyroid Dose
	(µCi*hr/cc)	rem per µCi*hr/cc	(rem)
I-131	2.76E-08	1.30E+06	3.58E-02
I-132	2.85E-08	7.70E+03	2.20E-04
I-133	4.39E-08	2.20E+05	9.66E-03
I-134	3.62E-09	1.30E+03	4.71E-06
I-135	1.10E-07	3.80E+04	4.20E-03
			4.99E-02

Table A2-9: Main Steam Line Reading at Release

Nuclide	Concentration	Half Life	Concentration 1 Hour after Shutdown
	(µCi/cc)	Hours	(µCi/cc)
I-131	9.27E-04	1.93E+02	9.23E-04
I-132	1.31E-03	2.38E+00	9.77E-04
I-133	1.53E-03	2.03E+01	1.47E-03
I-134	2.84E-04	8.77E-01	1.29E-04
I-135	4.14E-03	6.61E+00	3.73E-03
Xe-131m	4.26E-02	2.83E+02	4.25E-02
Xe-133	3.65E+00	5.42E+01	3.60E+00
Xe-133m	6.40E-02	1.27E+02	6.36E-02
Xe-135	1.16E-01	2.60E-01	8.04E-03
Xe-135m	6.07E-03	9.08E+00	5.62E-03
Xe-137	2.43E-03	6.38E-02	4.65E-08
Xe-138	8.82E-03	2.36E-01	4.67E-04
Kr-83m	5.62E-03	1.83E+00	3.85E-03
Kr-85	1.16E-01	4.48E+00	9.90E-02
Kr-85m	2.28E-02	9.40E+04	2.28E-02
Kr-87	1.49E-02	1.27E+00	8.62E-03
Kr-88	4.26E-02	2.84E+00	3.34E-02
Kr-89	1.28E-03	5.10E-02	1.60E-09
		Total Activity	3.90E+00

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:24 User Name: Unit Vent Alert

Comments:

User Supplied Information

Meteorological Data Inputs:

Groundlevel wind velocity: 13.2 mi/hr

Groundlevel wind from: 189 degrees

User-selected Stability Class

Stability Class: "D - Neutral"

Monitored Unit Vent Release:

Unit Vent Release Rate entered: 2.50E+006 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:24

Release Start Date/Time: 12/17/2013 15:24

Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+006 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	2.53E+004	I-131:	3.12E+003	Cs-134:	1.05E+000
Kr-85:	1.05E+004	I-132:	3.18E+003	Cs-137:	8.25E-001
Kr-85M:	7.06E+004	I-133:	6.05E+003	Ce/Pr-144:	2.10E-004
Kr-87:	9.03E+004	I-134:	3.02E+003	Ce-141:	2.84E-004
Kr-88:	1.74E+005	I-135:	5.12E+003	La-140:	5.31E-004
Kr-89:	3.06E-001			Mo-99:	3.43E-001
Xe-131M:	3.12E+003			Ru/Rh-106:	8.25E-005
Xe-133:	6.22E+005			Ru-103:	2.50E-004
Xe-133M:	1.91E+004			Sr/Y-90:	9.10E-005
Xe-135:	1.45E+005			Sr-89:	1.82E-003
Xe-135M:	8.14E+003			Te-132:	1.36E-001
Xe-137:	9.53E+000			Zr-95:	3.13E-004
Xe-138:	2.66E+004				

12/17/2013 3:24:46 PM

DRILL **STAMPEDE Results Information** DRILL
Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:24 **User Name:** Unit Vent Alert
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

Measurable Dose Rates / **PAG Dose Rates**

Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.009	0.016	0.137
1.0	0.0033	0.006	0.051
2.0	0.001	0.002	0.018
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

Measurable Doses / **PAG Doses**

Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.009	0.016	0.137
1.0	0.003	0.006	0.051
2.0	0.001	0.002	0.018
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+006 uCi/sec

Offsite Dose Projection (rem):

	1 mile	3 miles	5 miles	10 miles
TEDE	0.006	0.002	0.001	0.000
CDE	0.051	0.018	0.004	0.001

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of ≥ 3 mrem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:24:44 PM

Name Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director Date/Time

12/17/2013 3:24:28 PM

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/18/2013 07:54 User Name: SteamLine Site Alert
Comments:

User Supplied Information

Meteorological Data Inputs:
- Ground level wind velocity: 13.2 mi/hr
- Ground level wind from: 130 degrees
- User-selected Stability Class
- Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+000 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/18/2013 06:54
Release Start Date/Time: 12/18/2013 07:54
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.19E+007 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+004	I-131:	3.05E+003	Cs-134:	0.00E+000
Kr-85:	3.43E+005	I-132:	3.22E+003	Cs-137:	0.00E+000
Kr-85M:	5.79E+004	I-133:	4.88E+003	Ce/Pr-144:	0.00E+000
Kr-87:	2.55E+004	I-134:	4.22E+002	Ce-141:	0.00E+000
Kr-88:	9.89E+004	I-135:	1.23E+004	La-140:	0.00E+000
Kr-89:	4.25E-003			Mo-99:	0.00E+000
Xe-131M:	1.26E+005			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+007			Ru-103:	0.00E+000
Xe-133M:	1.87E+005			Sr/Y-90:	0.00E+000
Xe-135:	3.18E+005			Sr-89:	0.00E+000
Xe-135M:	1.23E+003			Te-132:	0.00E+000
Xe-137:	1.27E-001			Zr-95:	0.00E+000
Xe-138:	1.36E+003				

12/18/2013 7:55:19 AM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/18/2013 07:54 User Name: SteamLine Site Alert

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.108E-008

Measurable Dose Rates / **PAG Dose Rates**

Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.011	0.019	0.135
1.0	0.0042	0.007	0.050
2.0	0.002	0.003	0.017
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

Measurable Doses / **PAG Doses**

Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.011	0.019	0.135
1.0	0.004	0.007	0.050
2.0	0.002	0.003	0.017
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

12/18/2013 7:54:42 AM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+007 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.007	0.003	0.001	0.000
CDE	0.050	0.017	0.004	0.001

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of ≥ 3 mrem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY: ______________________ 12/18/2013 7:55:14 AM

Name Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director Date/Time

12/18/2013 7:54:42 AM

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:25 User Name: Unit Vent Site Area

Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored Unit Vent Release:
Unit Vent Release Rate entered: 2.50E+007 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:25
Release Start Date/Time: 12/17/2013 15:25
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+007 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	2.52E+005	I-131:	3.12E+004	Cs-134:	1.05E+001
Kr-85:	1.05E+005	I-132:	3.18E+004	Cs-137:	8.24E+000
Kr-85M:	7.06E+005	I-133:	6.04E+004	Ce/Pr-144:	2.10E-003
Kr-87:	9.03E+005	I-134:	3.08E+004	Ce-141:	2.84E-003
Kr-88:	1.73E+006	I-135:	5.12E+004	La-140:	5.31E-003
Kr-89:	3.14E+000			Mo-99:	3.43E+000
Xe-131M:	3.12E+004			Ru/Rh-106:	8.24E-004
Xe-133:	6.22E+006			Ru-103:	2.50E-003
Xe-133M:	1.91E+005			Sr/Y-90:	9.10E-004
Xe-135:	1.45E+006			Sr-89:	1.82E-002
Xe-135M:	8.18E+004			Te-132:	1.35E+000
Xe-137:	9.74E+001			Zr-95:	3.13E-003
Xe-138:	2.67E+005				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.03.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:25 User Name: Unit Vent Site Area

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.416E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.088	0.160	1.364
1.0	0.033	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.014
20.0	0.000	0.001	0.005

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.088	0.160	1.364
1.0	0.033	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.014
20.0	0.000	0.001	0.005

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/29/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+007 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.060	0.021	0.005	0.002
CDE	0.510	0.176	0.041	0.014

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RS1 (SITE AREA EMERGENCY) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:25:28 PM

Name Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director Date/Time

12/17/2013 3:25:21 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:28 User Name: Steam Line Site Area
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+001 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/17/2013 14:28
Release Start Date/Time: 12/17/2013 15:28
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.20E+008 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+005	I-131:	3.07E+004	Cs-134:	0.00E+000
Kr-85:	3.45E+006	I-132:	3.23E+004	Cs-137:	0.00E+000
Kr-85M:	5.81E+005	I-133:	4.90E+004	Ce/Pr-144:	0.00E+000
Kr-87:	2.55E+005	I-134:	4.22E+003	Ce-141:	0.00E+000
Kr-88:	9.91E+005	I-135:	1.24E+005	La-140:	0.00E+000
Kr-89:	3.92E-002			Mo-99:	0.00E+000
Xe-131M:	1.27E+006			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+008			Ru-103:	0.00E+000
Xe-133M:	1.88E+006			Sr/Y-90:	0.00E+000
Xe-135:	3.19E+006			Sr-89:	0.00E+000
Xe-135M:	1.21E+004			Te-132:	0.00E+000
Xe-137:	1.19E+000			Zr-95:	0.00E+000
Xe-138:	1.34E+004				

DRILL STAMPEDE Results Information DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:28 User Name: Steam Line Site Area

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.430E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.102E-008

Measurable Dose Rates / PAG Dose Rates

Distance (miles)	Measurable Dose Rates: Immersion Whole Body noble gas gamma (rem/hr)	PAG Dose Rates: TEDE external + internal (rem/hr)	PAG Dose Rates: Iodine CDE Thyroid (rem/hr)
0.5	0.111	0.189	1.354
1.0	0.0423	0.072	0.506
2.0	0.015	0.025	0.175
5.0	0.004	0.006	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.013
20.0	0.001	0.001	0.005

Measurable Doses / PAG Doses

Distance (miles)	Measurable Doses: Immersion Whole Body noble gas gamma (rem)	PAG Doses: TEDE external + internal (rem)	PAG Doses: Iodine CDE Thyroid (rem)
0.5	0.111	0.189	1.354
1.0	0.042	0.072	0.506
2.0	0.015	0.025	0.175
5.0	0.004	0.006	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.013
20.0	0.001	0.001	0.005

12/17/2013 3:28:53 PM

DRILL **STAMPEDE Results Information** DRILL
Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE
Wind Velocity: 13.2 mi/hr
Wind Direction: 180
Release Rate: 1.20E+008 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.072	0.025	0.006	0.002
CDE	0.506	0.175	0.041	0.013

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): **1**

SHELTER IN PLACE ZONE(S): **2**

AFFECTED DOWNWIND SECTORS: **R, A, B**

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RS1 (SITE AREA EMERGENCY) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:29:00 PM
Name Date/Time

REVIEWED BY: ______________________ ______________________
Rad Manager/Radiological Director Date/Time

12/17/2013 3:28:53 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:26 User Name: Unit Vent General
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored Unit Vent Release:
Unit Vent Release Rate entered: 2.50E+008 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:26
Release Start Date/Time: 12/17/2013 15:26
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+008 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	2.52E+006	I-131:	3.12E+005	Cs-134:	1.05E+002
Kr-85:	1.05E+006	I-132:	3.18E+006	Cs-137:	8.25E+001
Kr-85M:	7.06E+006	I-133:	6.04E+005	Ce/Pr-144:	2.10E-002
Kr-87:	9.03E+006	I-134:	3.08E+005	Ce-141:	2.84E-002
Kr-88:	1.73E+007	I-135:	5.12E+005	La-140:	5.31E-002
Kr-89:	3.10E+001			Mo-99:	3.43E+001
Xe-131M:	3.12E+005			Ru/Rh-106:	8.25E-003
Xe-133:	6.22E+007			Ru-103:	2.50E-002
Xe-133M:	1.91E+006			Sr/Y-90:	9.10E-003
Xe-135:	1.45E+007			Sr-89:	1.82E-001
Xe-135M:	8.16E+005			Te-132:	1.35E+001
Xe-137:	9.64E+002			Zr-95:	3.13E-002
Xe-138:	2.66E+006				

12/17/2013 3:26:37 PM

DRILL STAMPEDE Results Information DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:26 User Name: Unit Vent General
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

Measurable Dose Rates / PAG Dose Rates

Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.879	1.598	13.646
1.0	0.3322	0.601	5.099
2.0	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.006	0.050

Measurable Doses / PAG Doses

Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.879	1.598	13.646
1.0	0.332	0.601	5.099
2.0	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.006	0.050

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DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+008 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.601	0.210	0.050	0.017
CDE	$\boxed{5.099}$	1.762	0.411	0.135

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1, 2

SHELTER IN PLACE ZONE(S): 6, 11

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

> Based on a Site Boundary (1 Mile) Dose Projection ≥ 1 rem TEDE and/or 5 rem Thyroid CDE the Emergency Classification Initiating Condition RG1 (GENERAL EMERGENCY) has been met

PERFORMED BY: ______________________ 12/17/2013 3:26:33 PM

Name Date/Time

REVIEWED BY: ______________________ ______________________

End Manager/Radiological Director Date/Time

12/17/2013 3:26:25 PM

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:30 User Name: SteamLine General
Comments:

User Supplied Information

Meteorological Data Inputs:
- Ground level wind velocity: 13.2 mi/hr
- Ground level wind from: 180 degrees
- User-selected Stability Class
- Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+002 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/17/2013 14:30
Release Start Date/Time: 12/17/2013 15:30
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.20E+009 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+006	I-131:	3.07E+005	Cs-134:	0.00E+000
Kr-85:	3.45E+007	I-132:	3.24E+005	Cs-137:	0.00E+000
Kr-85M:	5.82E+006	I-133:	4.90E+005	Ce/Pr-144:	0.00E+000
Kr-87:	2.56E+006	I-134:	4.24E+004	Ce-141:	0.00E+000
Kr-88:	9.92E+006	I-135:	1.24E+006	La-140:	0.00E+000
Kr-89:	4.13E-001			Mo-99:	0.00E+000
Xe-131M:	1.27E+007			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+009			Ru-103:	0.00E+000
Xe-133M:	1.38E+007			Sr/Y-90:	0.00E+000
Xe-135:	3.19E+007			Sr-89:	0.00E+000
Xe-135M:	1.23E+005			Te-132:	0.00E+000
Xe-137:	1.24E+001			Zr-95:	0.00E+000
Xe-138:	1.35E+005				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:30 User Name: SteamLine General

Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

Measurable Dose Rates / **PAG Dose Rates**

Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	1.108	1.895	13.537
1.0	0.4238	0.717	5.058
2.0	0.153	0.254	1.747
5.0	0.040	0.063	0.407
7.5	0.022	0.034	0.211
10.0	0.015	0.022	0.134
20.0	0.006	0.008	0.049

Measurable Doses / **PAG Doses**

Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	1.108	1.895	13.537
1.0	0.424	0.717	5.058
2.0	0.153	0.254	1.747
5.0	0.040	0.063	0.407
7.5	0.022	0.034	0.211
10.0	0.015	0.022	0.134
20.0	0.006	0.008	0.049

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DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.20E+009 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.727	0.254	0.063	0.022
CDE	5.068	1.747	0.407	0.134

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1, 2

SHELTER IN PLACE ZONE(S): 6, 11

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 1 rem TEDE and/or 5 rem Thyroid CDE the Emergency Classification Initiating Condition RG1 (GENERAL EMERGENCY) has been met

PERFORMED BY: ____________________ 12/17/2013 3:30:48 PM

Name Date/Time

REVIEWED BY: ____________________ ____________________

~~Rad Manager/Radiological Director~~ ~~Date/Time~~

12/17/2013 3:30:39 PM

CN-2883

The particulate channel is used as part of the Reactor Coolant Pressure Boundary (RCPB) leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the Containment, are sufficient to detect an increase in leakage rate of the equivalent of one gal/min within one hour. Elements of this monitor, including the indicator mounted in the RMS CR cabinet, are designed and qualified to remain functional following a Safe Shutdown Earthquake (SSE), in compliance with RG 1.45. Further information on the RCPB leakage detection system is presented in Section 5.2.5.

11.5.2.3.3 Unit Vent Monitor: The unit vent monitor samples the plant vent stack prior to discharge to the environment and monitor for particulates, iodine, and noble gases.

The unit vent particulate and iodine monitor draws representative air samples from the plant vent stack via isokinetic nozzles in the stack, and directs them through a moving filter paper monitored by a shielded beta-sensitive scintillation detector. The sample stream then passes through a charcoal collector, where collected iodine is monitored by a shielded gamma-sensitive scintillation detector. The sample is then returned to the vent stack.

A separate wide-range gas monitor is provided for the unit vent. The monitor has two isokinetic nozzles for sampling during both normal and accident conditions. The stack samples pass first through a sample conditioning unit which filters particulates and iodine and may be used to take grab samples. The samples then pass through the shielded detector assembly, which uses three detectors to cover the complete range required. The low range detector uses a beta-sensitive plastic scintillator-photomultiplier (PM) tube. The mid-range and high-range detectors use cadmium telluride (CdTe), chlorine-doped, solid-state sensors. This wide-range gas monitor satisfies the requirements of NUREG-0737, Item II.F.1 for provisions for sampling plant effluents for iodines and particulates and for noble gas effluents from the plant vent.

11.5.2.3.4 Control Room Electrical Auxiliary Building Ventilation Monitors: The CR/EAB ventilation monitors are Class 1E monitors which continuously assess the intake air to the CR for indication of abnormal airborne radioactivity concentration. Each monitor assembly is powered from a separate electrical power source. In the event of high radiation CR emergency ventilation operation is initiated (Section 7.3.2). Failure of a monitor is alarmed in the CR.

Each monitor assembly is comprised of a recirculation pump, beta-sensitive scintillation detector, four-pi lead shielding, check source, stainless steel sample gas receiving chamber, and associated electronics.

11.5.2.3.5 Condenser Vacuum Pump Monitor: Gaseous samples are drawn through an off-~~line system by a pump from the discharge of the vacuum pump exhaust header of the condenser.~~ This channel monitors the gaseous sample for radioactivity which would be indicative of an SG tube leak, allowing reactor coolant to enter the secondary side fluid; this monitor complements the SGBD monitors in indication of a SG tube leak. The gaseous radioactivity levels are monitored by a single detector in a manner similar to the unit vent wide range gas monitor.

11.5.2.3.6 Spent Fuel Pool Exhaust Monitors: The SFPE monitors are Class 1E and are identical to the CR/EAB ventilation monitors described in Section 11.5.2.3.4 except that they sample the exhaust from the FHB. In the event of high radiation the monitors initiate emergency operation

11.5.2.5.1 Gaseous Waste Processing System Inlet Monitor: The GWPS inlet monitor employs a gamma (NaI crystal) scintillator/photomultiplier tube combination to measure the radioactivity level of the waste gases entering the GWPS. The monitor is used in conjunction with the GWPS discharge monitor to measure overall effectiveness of the GWPS.

11.5.2.5.2 GWPS Discharge Monitor: This monitor is similar to the GWPS inlet monitor and is installed upstream of the GWPS discharge valve. Upon detection of high radioactivity or monitor failure, the GWPS discharge valve, FV-4671, is automatically closed.

11.5.2.5.3 Main Steam Line Monitors: Each MS line is monitored by an ATL monitor consisting of a Geiger Mueller (GM) tube detector and an ion chamber detector with overlapping ranges. The detectors are shielded by 3 in. of lead.

The monitors are designed to monitor gross gamma activity in the steam line and provide a basis for determining possible atmospheric releases from the MS power-operated relief valve (PORV), SG safety valves, and/or auxiliary feedwater pump turbine.

The monitors provide a dose rate range equivalent to 10^{-1} to 10^{3} μCi/cm^{3} xenon-133. Based upon core inventory, the ratio of xenon-133 to other nuclides in the fuel can be detemined. In order to obtain the above concentrations of xenon-133 in the main steam line, a large primary-to-secondary leak must be present coincident with a large amount of fuel failure. The presence of xenon-133 indicates other radioactive isotopes are present.

Using the relative ratios of isotopes present in the MS line, a computer model for determination of dose rates from these isotopes, detector response curves, the thickness of the MS line, and the geometry of the MS line relative to the detector, the dose rate equivalent to MS line concentration is obtained. The quantity of radioactive effluents released is obtained by multiplying the xenon-133 equivalent MS line concentrations by the isotope ratio times the steam release rate.

These detectors are safety-related Class 1E and meet the requirements of RG 1.97 and NUREG-0737.

11.5.2.5.4 Steam Generator Blowdown Monitors: These monitors are identical to the MS line monitors and are adjacent to the SG blowdown lines in the Isolation Valve Cubicle (IVC).

The monitors are used as an aid in determining the source of SG blowdown radioactivity due to SG tube rupture or a large primary-to-secondary leak.

~~These detectors are safety-related Class 1E and meet the requirements of RG 1.97.~~

11.5.2.5.5 Main Steam Line High Energy Gamma (N-16) Monitors: Each main steam line is monitored by an ATL NaI scintillation detector. These detectors were installed to monitor the status of steam generator primary to secondary tube leaks and to provide a diagnostic tool for all individuals concerned with steam generator condition. These detectors are designed to detect high energy gamma activity in the 6 to 7.2 MEV energy range. High energy gamma activity in the main steam lines indicates the presence of N-16. The level of N-16 in the main steam lines is used to

RG2

Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 Facilities Description. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum

NRC ORDER EA-12-051 (SFP LEVELS)

REQUIREMENTS FOR RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION AT OPERATING REACTOR SITES AND CONSTRUCTION PERMIT HOLDERS

All licensees identified in Attachment 1 to this Order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

1. The spent fuel pool level instrumentation shall include the following design features:

    1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.

    1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

    1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

    1.4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

    1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.

    1.6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

1.7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

1.8 Testing: The instrument channel design shall provide for routine testing and calibration.

1.9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.

2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

The three critical levels that must be monitored in a spent fuel pool are discussed below. It should be noted that continuous indication from a single instrument over the entire span from level 1 to level 3 is not required but could be used. If more than one instrument is used to monitor the entire span, that set of instruments constitutes a single channel satisfying either the primary or backup instrument channel requirement (refer to Figure 1 below).

A visual representation of monitoring levels 1, 2 and 3 and the associated requirements for monitoring between the points are presented in Figure 1. The minimum requirements apply to the separation distance between level indications and support development of appropriate response procedures. These requirements are separate from the instrument channel design accuracy discussed in section 3, which apply to either discrete or to continuous instruments.

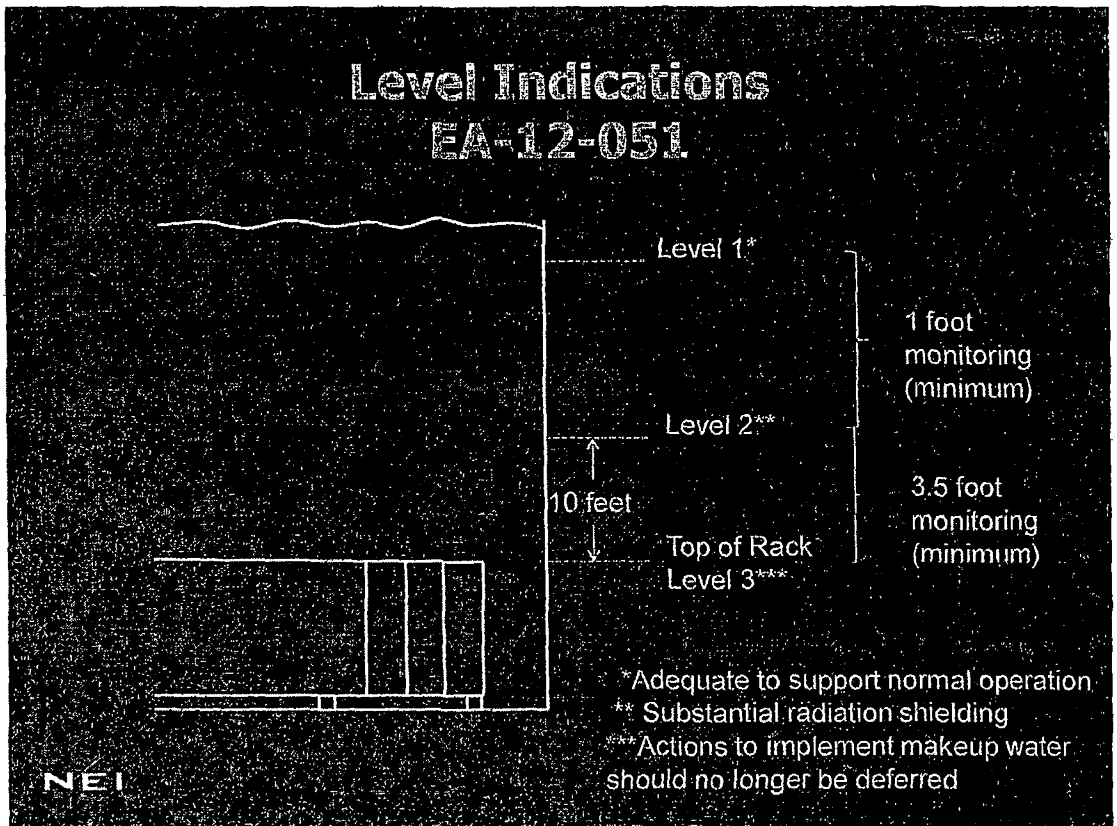

Figure 1

2.3.1. Level 1 – level that is adequate to support operation of the normal fuel pool cooling system

A typical fuel pool cooling system design includes a combination of weirs and/or vacuum breakers that prevent siphoning of the pool water level, below a minimum level, in the event of a piping rupture that can affect the SFP level. Level 1 represents the HIGHER of the following two points:

- The level at which reliable suction loss occurs due to uncovering of the coolant inlet pipe, weir or vacuum breaker (depending on the design), or
- The level at which the water height, assuming saturated conditions, above the centerline of the cooling pump suction provides the required net positive suction head specified by the pump manufacturer or engineering analysis.

This level will vary from plant to plant and the instrument designer will need to consult plant-specific design information to determine the actual point that supports adequate cooling system performance.

2.3.2. Level 2 – level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck

Level 2 represents the range of water level where any necessary operations in the vicinity of the spent fuel pool can be completed without significant dose consequences from direct gamma radiation from the stored spent fuel. Level 2 is based on either of the following:

- 10 feet (+/- 1 foot) above the highest point of any fuel rack seated in the spent fuel pools, or
- a designated level that provides adequate radiation shielding to maintain personnel radiological dose levels within acceptable limits while performing local operations in the vicinity of the pool. This level shall be based on either plant-specific or appropriate generic shielding calculations, considering the emergency conditions that may apply at the time and the scope of necessary local operations, including installation of portable SFP instrument channel components. Additional guidance can be found in EPA-400 (Reference 4), USNRC Regulatory Guide 1.13 (Reference 5) and ANSI/ANS-57.2-1983 (Reference 6).

Designation of this level should not be interpreted to imply that actions to initiate water make-up must be delayed until SFP water levels have reached or are lower than this point.

2.3.3. Level 3 – level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

Level 3 corresponds nominally (i.e., +/- 1 foot) to the highest point of any fuel rack seated in the spent fuel pool. Level 3 is defined in this manner to provide the maximum range of information to operators, decision makers and emergency response personnel. Designation of this level should not be interpreted to imply that actions to initiate water make-up must or should be delayed until this level is reached.

Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 28, 2013
NOC-AE-13002959
10 CFR 50.4
10 CFR 2.202

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1&2
Docket Nos. STN 50-498, STN 50-499
Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to
Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

References:
1. Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051)
2. NRC Interim Staff Guidance JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, August 29, 2012
3. Letter D. W. Rencurrel to NRC, "Initial Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)", dated October 24, 2012

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued an order (Reference 1) to STP Nuclear Operating Company (STPNOC). Reference 1 directs STP Nuclear Operating Company to provide a reliable indication of the water level in associated spent fuel storage pools. Specific requirements are outlined in Attachment 2 of Reference 1.

Reference 1 required submission of an overall integrated plan, including how compliance will be achieved. The final interim staff guidance (Reference 2) was issued August 29, 2012 providing licensees an acceptable approach for complying with the order. The purpose of this letter is to provide the overall integrated plan, including a description of how compliance will be achieved pursuant to Section IV, Condition C.1.a, of Reference 1 in accordance with the guidance in Attachment 2 to Reference 1 and the guidance in Reference 2. See the Enclosure for STPNOC's response to the requested information.

There are no new commitments in this letter.

33650640

If there are any questions regarding this letter, please contact Robyn Savage at (361) 972-7438.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2/28/13

Dennis L. Koehl
President and CEO/CNO

Enclosure: South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051

cc:
(paper copy)

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South Texas Project (STP)
Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation
to Meet NRC Order EA-12-051

1.0 OVERALL INTEGRATED PLAN INTRODUCTION

This document provides the overall Integrated Plan (the "Plan") which the STP Nuclear Operating Company ("STPNOC") will implement for Units 1 and 2 to comply with the requirements of NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Ref.2), (the "ORDER"), NRC Interim Staff Guidance JLD-ISG-2012-003 [Rev.0] (Ref.3), (the "ISG"), and NEI Report 12-02 [Rev.1] ("NEI 12-02").

This Plan follows the format and provides all of the information on the STP 1 & 2 Integrated Plan that is required in NEI 12-02 [Rev.1] (Ref.1), Section A-2-2. Throughout this Plan, any reference to NEI 12-02 and the ISG will be based on the revisions above. Any reference to NEI 12-02 will include compliance to the clarifications and exceptions to NEI 12-02 required by the Interim Staff Guidance, Rev. 0.

In response to the NRC requirements, STPNOC will provide two channels of independent, permanently-installed, wide-range spent fuel pool level instrumentation ("SFPLI"), for the spent fuel pool ("SFP") of each unit. The spent fuel pool for each unit is independent and not interconnected in any way. For each SFP, the instrumentation provided for each channel will utilize the same technology, as permitted by the NEI 12-02 [Rev.1]. The spent fuel pool level instrumentation will provide continuous level indication for each SFP on both the Primary and Backup Channels.

Both the Primary and Backup Channel/Instrument location and display of the SFP level will be independently mounted in each unit's Radwaste Control Room in the Mechanical Electrical Auxiliary Building (MEAB), which is an accessible post-event location. Other locations are still being considered.

Both the Primary and Backup Channel remote, non-safety related indication of the SFP level will also be provided in each unit's Control Room via input to the Plant Computer.

The instrumentation systems will not be safety-related, but will meet the requirements for augmented quality in accordance with NEI 12-02 [Rev.1] and the ISG as described below.

Since all of the potential suppliers have not completed development, the information in this Plan is based on the overall strategy and on information which, based on current information from potential suppliers, is thought to envelope the systems being developed for this application.

If there are any changes to the requirements in NRC JLD-ISG-2012-003 [Rev.0] and NEI 12-02 [Rev.1], relief from the requirements and schedule documented in this Plan may be required, in accordance with Section 12.0. Any required changes to this Plan will be defined in the periodic status reports submitted to the NRC.

2.0 APPLICABILITY:

This Plan applies to the spent fuel pools for South Texas Project Unit 1 and Unit 2.

3.0 SCHEDULE:

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 1 is scheduled for completion prior to 10/28/2015, which is the end of the second refueling outage (1RE19) following submittal of this Plan.

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 2 is scheduled for completion prior to 4/29/2015, which is the end of the second refueling outage (2RE17) following submittal of this Plan.

Unit 1 Milestones are as follows:

- Design/Engineering – September of 2014
- Purchase of instruments & equipment – February of 2015
- Receipt of equipment – June of 2015
- Unit 1 Installation & Functional Testing – October of 2015

Unit 2 Milestones are as follows:

- Design/Engineering – December of 2013
- Purchase of instruments & equipment – August of 2014
- Receipt of equipment – November of 2014
- Installation & Functional Testing – April of 2015

Consistent with the requirements of the ORDER and the guidance from NEI 12-02 [Rev.1], status reports will be generated in six (6) month intervals from the submittal of this Plan.

4.0 IDENTIFICATION OF SPENT FUEL POOL WATER LEVELS:

The STP Unit 1 and 2 spent fuel pools are essentially identical. The following SFP elevations are identified:

- The bottom of the pool is at Plant El. 21 ft. 11 in.
- The top of the SFP racks is approximately at Plant El. 39 ft. 10 in.
- The minimum Limiting Condition for Operation SFP level is Plant El. 62 ft.
- Normal SFP water level is at Plant El. 66 ft. 6 in.
- Non-safety related level switch alarms are activated at Plant El. 67 ft. on high level and Plant El. 66 ft. on low level.
- The spent fuel pool deck is at Plant El. 68 ft.

The required key SFP water levels, per guidance of NEI 12-02 [Rev.1] and ISG (with clarifications and exceptions), are as follows:

4.1 LEVEL 1: Level adequate to support operation of the normal fuel pool cooling system.

LEVEL 1 represents the HIGHER of either the level at which reliable suction loss to the cooling pump occurs, or, the required NPSH (Nominal Pump Suction Head) of the cooling pump.

Loss of reliable suction to SFP cooling pumps. For the purposes of this Plan, this level will conservatively be placed at Plant El. 64 ft. 2 in. This allows for just over 1 ft. of SFP water level above the top of the suction inlet flange (SFP Cooling Pump 14 in. suction line with centerline of suction inlet flange at Plant El. 62 ft. 6 in.) which will be sufficient for NPSH. (Ref. 9)

Therefore, considering the top of SFP fuel storage rack is at Plant El. 39 ft. 10 in., the indicated level on either the Primary or Backup Instrument Channel of greater than 24 ft. 4 in. above the top of the SFP fuel storage racks based on the design accuracy for the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, is adequate for normal SFP cooling system operation.

LEVEL 1 = **Plant El. 64 ft. 2 in or 24 ft. 4 in. water level above the top of the SFP fuel storage rack**

4.2 LEVEL 2: Level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck.

Indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of SFP stored fuel assemblies based on current guidance in NRC RG 1.13 [Rev.2] (Ref. 4) will achieve substantial radiation shielding. Requirements on substantial SFP radiation shielding is also given in ANSI/ANS-57.2-1983 (Ref. 5), and states that radiation shall not exceed 2.5 mRem/hr, but the minimum water depth to achieve this is undefined. NRC RG 1.13 [Rev.2] took exception to using dose rates as design input for minimum SFP water level, and instead defined the minimum level as 10 ft. above the stored fuel assemblies.

STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. Therefore, indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of the SFP fuel storage rack, based on the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, ensures there is adequate water level to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck.

LEVEL 2 = Plant El 49 ft. 10 in. or 10 ft. water level above the top of the SFP fuel storage rack.

4.3 LEVEL 3: Level where the fuel remains covered.

As stated above, STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. The installation of the SFPLI sensor will be such that it will measure as close as possible to the top of the SFP fuel rack. Indicated level on either the Primary or Backup Instrument Channel of greater than ½ ft. above the top of SFP fuel storage racks based upon the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, satisfies the NEI 12-02 [Rev.2] requirement of ± 1 ft. from the top of the fuel rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the SFP fuel storage rack.

LEVEL 3 = Plant El 40 ft. 4 in. or 6 in. water level above the top of the SFP fuel storage rack.

5.0 INSTRUMENTS:

Both the Primary and Backup Instrument Channels will utilize permanently-installed instruments. The design of the primary and backup instruments will be consistent with the requirements by NEI 12-02 [Rev.1], the ISG, and this Plan.

The current plan is for both channels to utilize Guided Wave Radar, which functions according to the principle of Time Domain Reflectometry (TDR). A generated pulse of electromagnetic energy travels down the probe. Upon reaching the liquid surface the pulse is reflected and based upon reflection times level is inferred. The measured range will be continuous from the high pool level elevation (67') to the top of the spent fuel racks (Ref. 8). However, STP is still evaluating other designs for this application. Any changes to the current plan will be reported in the 6 month update letter.

The supplier for the SFP instrumentation will be chosen by a competitive bidding process completed after submittal of this Plan, so the information in this Plan is based on the overall strategy and on available information from potential supplier's information on systems being developed for this application.

5.1 Primary (fixed) Instrument Channel

The Primary Instrument Channel level sensing components will be located in the northeast corner of the Spent Fuel Pool, as shown in Attachment 1. The primary instrument channel will provide continuous level indication over a range from Plant El. 40 ft. 4 in. (LEVEL 3) to Plant El. 67 ft. (SFP high level alarm) or a range of 26 ft. 8 in. In addition, the capability for discrete level indications at LEVEL1, LEVEL 2 and LEVEL 3, as described in Section 4.0, will be available.

Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 Facilities Description. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum

CU1

STEP DESCRIPTION FOR 0POP04-RC-0003 STEP 3.0

STEP: CHECK Trends For Any Of The Following Indications Of RCS Leakage:

- Rad Monitor RT8011 Particulate – Rising
- Reactor Coolant Drain Tank Level – Rising
- Pressurizer Relief Tank Level – Rising
- RCB Normal Sump Level – Rising

PURPOSE: To determine if leakage is from RCS and not CVCS.

BASIS: Indication of RT8011, RCDT, PRT or RCB Normal Sump levels rising will confirm that the leakage is from RCS and not CVCS which is normally tied to the RCS.

ACTIONS: Monitor trends from RT8011, RCDT, PRT or RCB Normal Sump.

INSTRUMENTATION: Level indications located on CP004 and various plant computer monitors located in control room. Radiation Monitor Computer RM-11.

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A

3.57 Minimize the time at lowered RCS inventory (fuel in the reactor with level at or below the reactor vessel flange). Controls for Infrequently Performed Evolution per 0PGP03-ZA-0506, Tests or Evolutions Requiring Additional Controls, and 0PGP03-ZO-0049, Conduct of Tests or Evolutions Requiring Additional Controls, SHALL be in place prior to lowering RCS level below 0% Pressurizer Cold Calibration Level elevation (elevation 52 ft 2 in) at Step 9.30.

3.58 WHEN Steam Generator (SG) temperature is lowered, THEN SG narrow range level indication will indicate higher than actual level.

3.59 Addendum 7 contains a list of conditions that should be met prior to taking credit for using the Steam Generators as a decay heat removal means while in Mode 5.

3.60 During plant cooldown, all SGs will normally be connected to the steam header to assure a uniform cooldown of the RCS. (UFSAR 5.2.2.11.3)

3.61 The Main Steam lines upstream of the MSIVs may require periodic blowdown for moisture control. This can be accomplished by performing Addendum 13. MONITOR the following "MAIN STEAM OUTLET DRIP LEG LEVEL SWITCH" Plant Computer points for indications of moisture buildup in the Main Steam Lines:

- LD7900, S/G 1A(2A) MS LN DRN FROM MS-2001
- LD7901, S/G 1B(2B) MS LN DRN FROM MS-2002
- LD7902, S/G 1C(2C) MS LN DRN FROM MS-2003
- LD7903, S/G 1D(2D) MS LN DRN FROM MS-2004

3.62 Deaerator Storage Tank Level SHALL be maintained in normal band of 60% to 80% when condenser vacuum is established. Going below 60% level may affect condenser vacuum. (Ref. 2.111)

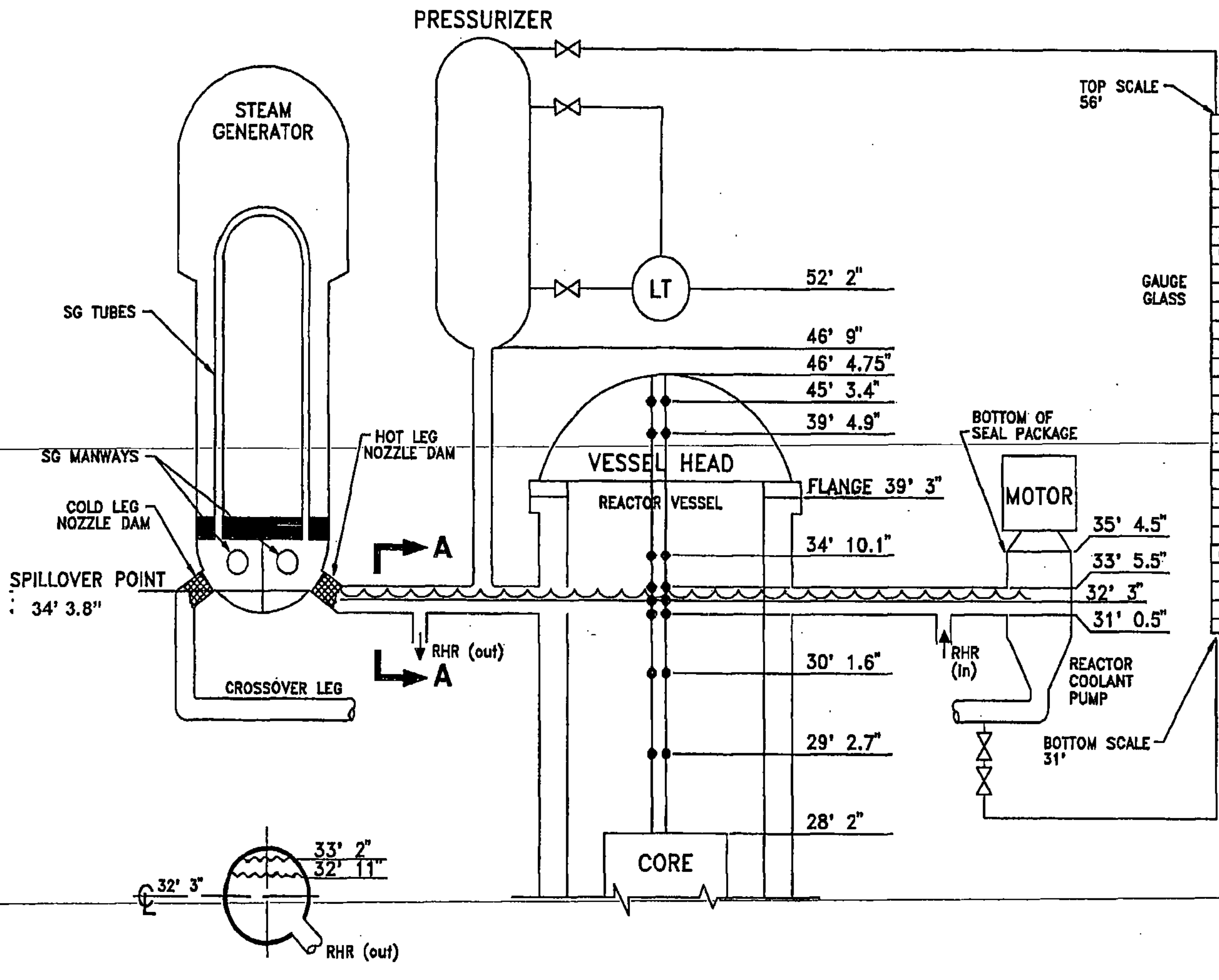
3.63 The principles of 0PGP03-ZO-0042, Reactivity Management Program, are in effect at all times during Operations in this procedure.

3.64 Shutdown margin SHALL be verified adequate based on the RCS boron concentration.

3.65 IF planned to place the RCS in MODE 5 with reactor coolant loops **NOT** filled or MODE 6 AND planned to swap the CVCS Bed Demineralizers in service during RCS in MODE 5 with reactor coolant loops **NOT** filled or MODE 6 THEN FLUSH the oncoming Demineralizers per 0POP02-CV-0004, Chemical and Volume Control System Subsystem PRIOR TO entering RCS in MODE 5 with reactor coolant loops **NOT** filled and MODE 6 conditions. (Ref 2.57)

3.66 IF Personnel Air Lock (PAL) doors are open in Mode 5, THEN Addendum 21, Closure of Personnel Air Lock Doors, is available to establish containment closure.

REACTOR COOLANT SYSTEM



SECTION A-A
HOT LEG

STP D-0794
Rev 2

NOTE

- Top of Core is elevation 28 ft 2 inches.
- SG spillover is elevation 34 ft 3.8 inches.

SENSOR UNCOVERED	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
All Covered	100	100	46' 4.75"	Upper Head Full
1	64	100	45' 3.4"	Upper Head Partially Drained
2	0	100	39' 4.9"	Plenum Full
3	0	85	34' 10.1"	Plenum **NOT** Full (Enter Reduced Inventory)
4	0	66	33' 5.5"	Top of Hot Leg Nozzle
5	0	48	32' 3"	Hot Leg Centerline
6	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
7	0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
8	0	0	29' 2.7"	Upper Core Plate

CU2

PROCEDURE PURPOSE

This procedure provides guidelines for the initial response and stabilization of the plant in the event of a loss of any single or all 13.8 KV bus(es) or 4.16 KV Bus(es). This includes all 13.8 KV Auxiliary and Standby buses, and 4.16 KV buses with the exception of Buses 1K(2K), 1L(2L) and 1M. Loss of a 4.16 KV ESF bus is addressed as it indicates at least a partial loss of offsite and onsite AC power (ESF bus power can only be completely lost if both offsite and onsite power sources to the specific bus are lost).

MAJOR ACTION CATEGORIES

- Provide interface with Emergency Operating Procedures and provide the instructions to establish the minimum equipment required to safely stabilize the unit.
- Identify actions associated with commitments to perform the action within a specified time period after the initiating event.

DISCUSSION:

The electrical distribution system at STP has by design, a high degree of flexibility and ability to withstand casualties, especially the Class 1E alternating current systems. However throughout the nuclear industry Loss Of Offsite Power (LOOP) events have occurred as well as Station Blackout (loss of all offsite and onsite AC power) events.

When dealing with a loss of offsite AC power, both complete and partial, with the Unit in Modes 1 or 2, the loss of an Auxiliary power bus will result in the loss of a Reactor Coolant Pump requiring a reactor trip because STP is not analyzed for operation with only three Reactor Coolant Pumps. In the event that no ESF bus is available the indication is that all offsite and onsite AC power has been lost requiring transition to the Emergency Operating Procedures. Under these same conditions STP has committed to shed the Channel I Load Sequencer from its power supply within the first 30 minutes after the initiating event, and if the associated battery bank has a jumpered cell then all the loads on DP 1201 and DP 1204 will be shed except for QDPS and SG PORVs.

The initial response provided by this procedure is directed to the stabilization of critical plant parameters and then analyzes the extent of the loss of power. While this procedure does not identify the specific combination of buses that have been lost, it does identify the specific area of the power loss so that a procedure that is more specific to the method for power restoration can be referred to.

STEP DESCRIPTION FOR 0POP04-AE-0001 STEP 3.0

STEP: CHECK 4.16 KV ESF Bus Status:

a. ANY 4.16 KV ESF Bus NOT energized from offsite power (VERIFY the voltage on all three phases of each ESF Bus).

b. VERIFY Applicable STBY DG(s) running

c. VERIFY Applicable STBY DG(s) output breaker(s) closed to the associated 4.16 KV ESF bus

PURPOSE: To determine the status of the 4.16 KV ESF buses and performs any corrective actions that can be performed under the current conditions.

BASIS: ~~This step attempts to start SDG for a de-energized bus. Also this step ensures output~~ breaker is closed and if not determines the cause of the failure and provides steps to correct and energize the bus.

If "4KV BUS O/C LOCKOUT" indicating lamp on applicable BSMP {CP003} is illuminated the bus cannot be energized until corrective maintenance is complete.

ACTIONS: Determine if the SDG is available to be started by checking for O/C lockout and other fault protection. If available then perform the steps to start SDG and close output breaker.

If the SDG is already running at this step, then determine the need to close the affected SDG breaker to energize the associated bus and close the breaker in the event that no faults exist. If a fault does exist, then the cause of the fault would have to be corrected before protective actuation device can be reset and the bus energized.

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: If the SDG has a 4.16 KV ESF Bus overcurrent lockout, SDG generator differential lockout or an SDG overspeed lockout then these faults will need to be corrected and reset to energize the bus.

PROCEDURE PURPOSE

The purpose of this procedure is to restore power to any ESF bus which is not energized. In the case where only one ESF bus is energized by a DG, and another one cannot be energized by the associated DG or offsite power, then steps are taken to operate breakers and disconnects to use the one running DG to supply key loads on another bus.

MAJOR ACTION CATEGORIES

- Tie the operating DG to another bus via the emergency switchgear bus 1L(2L).
- Energize at least one ESF bus from the Emergency Transformer.
- Control and load essential equipment on to the available ESF buses.

DISCUSSION:

STP has committed under specific conditions related to loss of offsite and onsite power to energize at least two ESF buses from a running DG in order to energize specific loads needed to extend station battery life or provide availability of ESF equipment that is electrically powered from one of two specific ESF buses.

NOTE

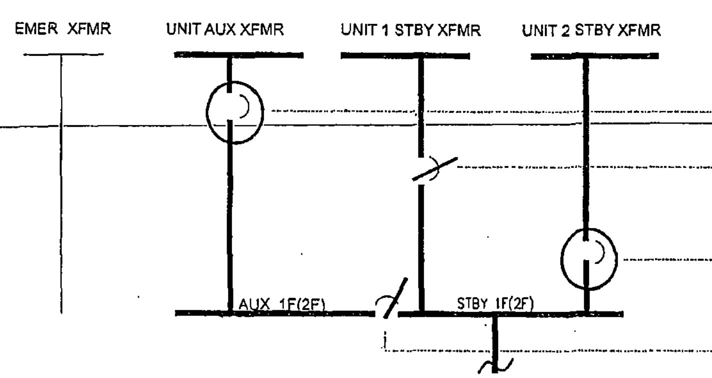
There are 5 possible lineups on Data Sheet 2, 3, and 4 for 13.8 KV XFMRS in the **DESIGNATED** Class 1E 4160 VAC Bus Power Source Table that meet Technical Specification requirements for being a power source for the 4.16 KV Buses:

- (1) UAT supplying the AUX BUS and STBY BUS
- (2) UAT supplying the AUX BUS and UNIT 1 STBY XFMR supplying the STBY BUS
- (3) UNIT 1 STBY XFMR supplying the STBY BUS and the AUX BUS
- (4) UAT supplying the AUX BUS and UNIT 2 STBY XFMR supplying the STBY BUS
- (5) UNIT 2 STBY XFMR supplying the STBY BUS and the AUX BUS

5.2 COMPLETE Required ESF Power Train Data Sheet 2 through 4 by performing the following steps.

5.2.1 RECORD actual breaker/disconnect positions for the 13.8 KV XFMRs, AUX BUS, STBY BUSES and from the 13.8 KV STBY BUS to the 480 VAC BUSES.

- RECORD "CLOSED" breaker/disconnect positions by drawing a line at an angle through the breaker.
- RECORD "OPEN" breaker/disconnect positions by drawing a CIRCLE around the breaker.



This example illustrates:
(3) UNIT 1 STBY XFMR supplying the STBY BUS and the AUX BUS

6.0 Acceptance Criteria

NOTE

- Addendum 2, Two Physically Independent Circuits, provides a drawing of rights of way and offsite circuits to aide in the definition of "two physically independent circuits".
- Loss of one 13.8 KV Standby Bus to 4.16 KV ESF bus line constitutes loss of one required offsite source. (Reference 8.2)
- Loss of two 13.8 KV Standby busses to 4.16 KV ESF bus lines constitutes loss of two required offsite sources. (Reference 8.2)
- The preceding notes also apply when the 4.16 KV ESF bus is not energized by the 13.8 KV XFMR.
- Step 6.1 applies during standby diesel inoperability.
- Step 6.2 applies during offsite independent circuits inoperability.
- Note and Precaution 3.28 should be referred to for additional clarification regarding allowable indication to be utilized when obtaining 345 KV switchyard voltage.

6.1 Two physically independent circuits exist between the offsite transmission network and onsite Class 1E Distribution System as determined from Data Sheet 1, 2, 3, 4, and 9. (Technical Specifications 3.8.1.1.b, 3.8.1.1.f, and 4.8.1.1.1.a)

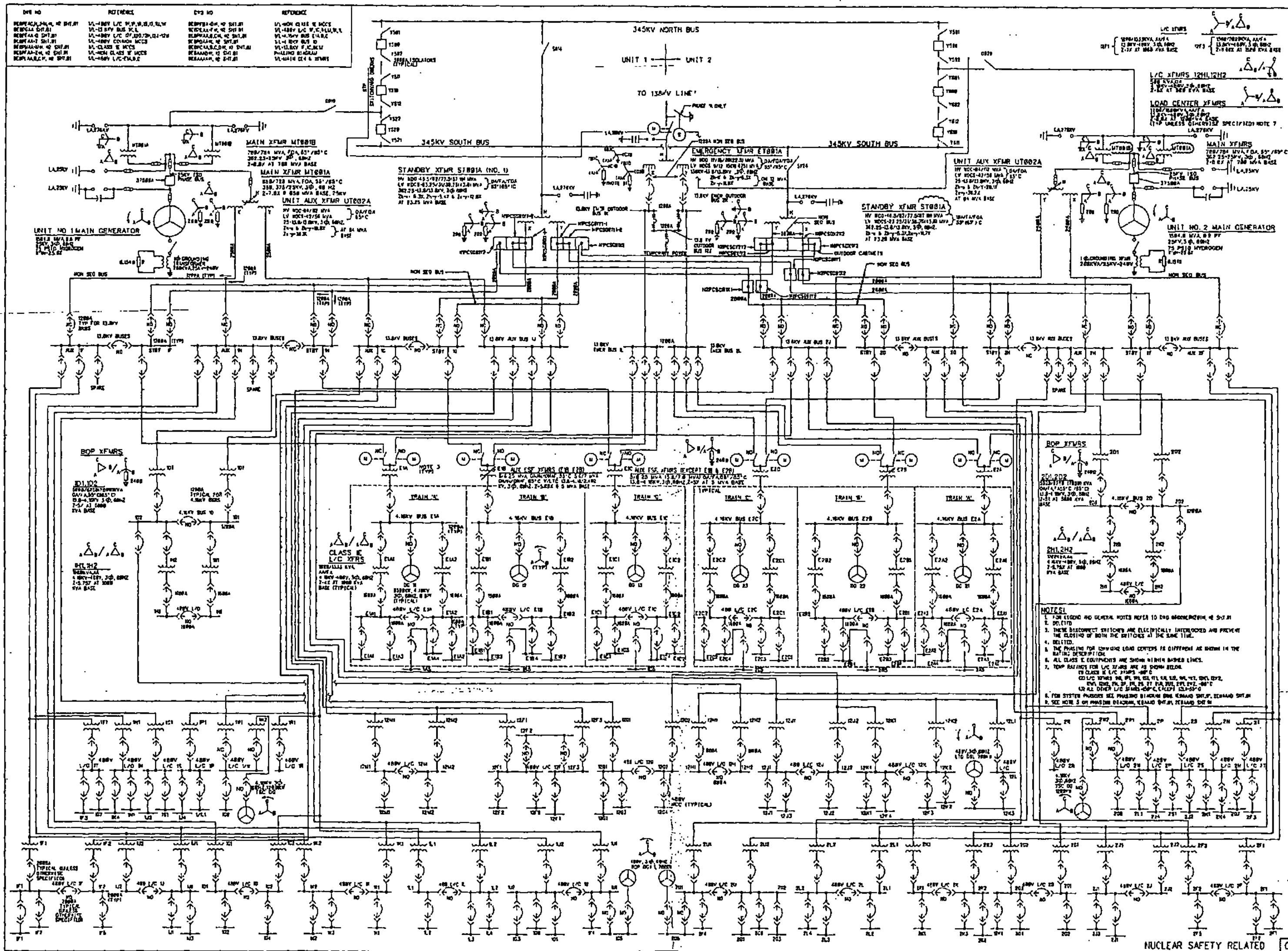
- North and South Bus in service with bus voltage:
  - ≥ 340 KV" for NORMAL LINEUP

    OR

  - ≥ 356 KV for NORMAL LINEUP with UAT or Train B ESF LTC in "MANUAL"

    OR

  - ≥ 358 KV for all ALTERNATE LINEUPs **OR** voltage specified in the "Minimum Voltage for Various Alternate 13.8 KV Bus Alignments" Addendum of 0POP02-AE-0002, Transformer Normal Breaker and Switch Lineup.
- Two of the following Rights of Way with a 345 KV line are available:
  - NW Right of Way 1 (White Point 39)
  - NW Right of Way 2 (Elm Creek 27 OR WA Parish 39 OR Elm Creek 18)
  - Eastern Right of Way (Dow Velasco 27 OR Dow Velasco 18)
- Two of the following 13.8 KV XFMRs are available:
  - Unit Aux XFMR
  - Unit 1 Stby XFMR
  - Unit 2 Stby XFMR
- Three 13.8 KV Standby Buses energizing the 4.16 KV ESF bus lines.



No.	Issue Date	Revision
22	4-6-14	INCORP. DCN 1888872.
21	9-30-09	INCORP. DCN 0331876 AND REVISED PER CR 09-117-9.
20	9-5-07	INCORP. DCN 0700049.
19	6-23-05	INCORP. DCN 0500726.
18	10-16-02	REVISED PER CR 02-13827-1
0-12	8-20-83	ISSUED FOR CONSTRUCTION.

SOUTH TEXAS PROJECT
NUCLEAR OPERATING COMPANY

SINGLE LINE DIAGRAM
MAIN ONE LINE DIAGRAM
UNIT NO. 1 & 2

Priority	CADD File No.	Dwg. No.	STI No.	Scale	Dwg. No.	Sht	Rev.
1	E0187.241	D070928	33518968	NONE	00000E0AAAA	01	24

CU3

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

CU4

33742632

	0PGP04-ZA-0307	Rev. 6	
	Preparation of Calculations		
Form 1	Calculation Cover Sheet		

CALCULATION COVER SHEET

Page 1

Calculation No.: 13-DJ-006 **Unit:** 9 **Bldg/Area/Sys:** VARIOUS

Quality Class: A **Priority Code:** 2

[ ] Design Calculation [X] Engineering Calculation **Cog. Org.:** ELECTRICAL

Title: 125 VDC BATTERY FOUR HOUR COPING ANALYSIS

Additional Review: **Dept:** N/A **Signature:** **Date:**

Additional Review: **Dept:** N/A **Signature:** **Date:**

RPE Certification Required: [ ] Yes [X] No

RPE Signature: N/A **Date:** **Registration No.:**

RPE Seal:

☐ This calculation revision contains a change in the methodology as described in UFSAR Section ________ Rev____.

ORIGINAL

CR actions tracking documents impacted by this revision to the calculation:

N/A

	Approval Signature PRINT/SIGN	Date	Rev	Revision Description
Originator (ESP Cert 9569)	HECTOR LEON / Hector E. Leon	9/12/13	0	Initial Issue
Checker (ESP Cert 9569)	Santos Rosales / Santos Rosales	9/12/13	0	
SE	C.H. Georgeson / C.H. Georgeson	9/12/13	0	

5.0 ACCEPTANCE CRITERIA

5.1 Battery Size

The required battery size, as calculated using the IEEE Standard 485-1978 (Ref. 6.3.2) methodology, must be less than or equal to the installed battery size, including the impact of minimum temperature and aging factors. This is determined by comparing the number of positive plates calculated to the actual number of positive plates for the installed battery.

STP's defense-in-depth strategy requires the four Class 1E DC channels to be "AC-Independent" for a minimum of four (4) hours, to facilitate coping with a postulated loss of AC power event. The results of this calculation show the following:

5.1.1. With no battery cells jumpered out (i.e. 59 cell operation), the Class 1E DC Channel I can operate for a period of four (4) hours without battery charging support by manually de-energizing ESF Load Sequencer "A" within 30 minutes following the loss of Channel I battery charging capability. Class 1E DC Channel II can operate for a period of eight (8) hours without battery charging support and without shedding of any connected loads. Class 1E DC Channels III and IV can operate for a period of four (4) hours without battery charging support and without shedding of any connected loads.

5.1.2. With one battery cell jumpered out (i.e. 58 cell operation), the Class 1E DC Channel I can operate for a period of four (4) hours without battery charging support by manually de-energizing the ESF Load Sequencer "A" and shedding all but three loads on Panel 1201 within 30 minutes following the loss of Channel I battery charging capability. Class 1E DC Channel II can operate for a period of eight (8) hours without battery charging support and without shedding of any connected loads. Class 1E DC Channel III can operate for a period of four (4) hours without battery charging support and without shedding of any connected loads. Class 1E DC Channel IV can operate for a period of four (4) hours without battery charging support by manually shedding all but three loads on Panel 1204 within 30 minutes following the loss of Channel IV battery charging capability. The loads on both 1201 and 1204 are breakers 13, 15, and 17 which are QDPS APC A1 (C1) at 7 amps, QDPS APC A2 (C2) at 10 amps and Steam Generator 1A PORV Servo Amplifier at 2 amps, according to EC-5008. This is a total of 19 amps which are then converted to power at 120 AC resulting in a power of 2280 watts. The efficiency losses per EC-5008 are 2511 w. Summing these results in a power of 4791 W which are then converted back to DC amps at 125 VDC resulting in the loads on the respective batteries EIV1201 and EIV1204 of 38.328 amps after 30 minutes.

The minimum battery voltage that was used in this calculation for all safety batteries to calculate the margin above was 106 volts. As an input to operations Emergency Operating Procedure 0POP05-EO-EC00 'Loss of All AC Power', the minimum bus voltage that any safety train can operate to is 105.5 VDC. Below 105.5 VDC it may be possible that some loads will have inadequate voltage to operate properly.

6.0 REFERENCES

6.1 Regulatory

6.1.1 South Texas Project Technical Specifications and Bases, Amendment 198 for Unit 1 and Amendment 186 for Unit 2

6.1.2 South Texas Project Updated Final Safety Analysis Report (STP UFSAR) Chapter 8, Revision 16

6.1.3 Letter from T. H. Cloninger, STPEGS, to the NRC Document Control Desk, Revised Position of 10CFR50. 63, "Loss of All Alternating Current Power," dated March 1, 199. (ST-HL-AE-5010)

6.1.4 Letter from T. H. Cloninger, STPEGS, to the NRC Document Control Desk, Supplemental Information to Revised Position of 10CFR50.63, "Loss of All Alternating Current Power," dated June 14, 1995 (ST-HLAE-5103)

6.1.5 Letter from the Thomas W. Alexion, NRC, to Mr. William T. Cottle, STPEGS, Revised Station Blackout (SBO) Position, South Texas Project, Units 1 and 2 (STP), dated July 24, 1995 (TAC Nos. M90061 and M90062) (ST-AE-HL-94257).

6.1.6 10CFR 50.59 Screen # 10-17753-5 Revision 0, "Revise Station Blackout Position to delete the need for a Coping Analysis"

6.2 Technical

6.2.1 Class 1E 125 VDC Design Criteria Document, 4E520EQ0100, Rev 6

6.2.2 VTD-A363-002 Rev 6, "Instruction and Operating Manual 10 KVA Inverter"

6.2.3 VTD-A363-0045, Rev 1, "Vendor Technical Manual for Ametek Solidstate Controls 25KVA Inverter / Rectifier"

6.2.4 EC05036 Rev 8 "DC Cable Sizing"

6.2.5 EC05037 Rev 5 "Maximum Allowable Length of AC Power Cables"

6.2.6 EC06038 Rev 9 "Power Cable Sizing Verification"

6.2.7 VTD-S637-0009 Rev 1, ESF Load Sequencer for South Texas Project Electric Generating Station

6.2.8 DCN 9602493, dated 4/29/96

ADDENDUM 4
VITAL DC BUS MONITORING

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Do NOT allow battery voltages to drop to LESS THAN 105 VDC for plant equipment protection.

NOTE

Train A, B, and C bus voltages should be monitored for the duration of the event, and their respective battery output breakers opened if bus voltages lowers to LESS THAN OR EQUAL TO 105.5 VDC in order to conserve the battery should a STBY DG become available.

___4 **MONITOR Class 1E 125 VDC system Train A, B, & C bus voltage.**

___a. Train A AND B bus voltages - GREATER THAN 105.5 VDC

a. PERFORM the following:

1) DISPATCH operator to perform ADDENDUM 3, FAILING AIR TO MSIVs AND MSIBs for all MSIV(s) and MSIB(s).

2) WHEN ADDENDUM 3, FAILING AIR TO MSIVs AND MSIBs is complete, THEN GO TO Step 4.b of this Addendum.

___b. Train A, B, OR C bus voltages - GREATER THAN 105.5 VDC.

b. DISPATCH operator to open the associated battery output breaker:

- "BTRY E1A11(E2A11) MAIN BKR" E1A11(E2A11) BKR 1B (EAB 10')
- "BTRY E1B11(E2B11) MAIN BKR" E1B11(E2B11) BKR 1B (EAB 35')
- "BTRY E1C11(E2C11) MAIN BKR" E1C11(E2C11) BKR 1B (EAB 60')

___5 **CHECK Sequencer(s) ready for restoration following bus energization**

RETURN TO procedure step in effect.

CU5

STI 32563193	0PGP05-ZV-0011	Rev. 7	Page 1 of 23
Emergency Communications			
Quality	Non Safety-Related	Usage: **Available**	Effective Date: 12/03/09

S. Korenek	N/A	N/A	Emergency Response Division
PREPARER	TECHNICAL	USER	COGNIZANT ORGANIZATION

Table of Contents Page

1. Purpose and Scope

   1.1 This procedure provides guidance in the use of emergency communications systems when responding to an emergency at the South Texas Project Electric Generating Station (STPEGS).

2. Definitions

   2.1 FTS 2001 System: A federal telephone system used by the Nuclear Regulatory Commission (NRC) and nuclear utilities for emergency communications.

   2.2 RINGDOWN LINE: A telephone line that does NOT require the operator or caller to dial a number to activate the circuit.

   2.3 UNIT OVERRIDE: A circuit select switch (CSS) found on selected communications consoles, which when selected, activates prioritization circuitry for public address announcements. Additionally, when activated, this button directs announcements to ALL public address zones.

3. Responsibilities

   3.1 The Emergency Director, or designee, is responsible for activating the Emergency Notification System (ENS) to notify the NRC of a declared emergency, and to maintain communications with the NRC Operations Center.

   3.2 The Emergency Director, or designee, is responsible for activating the State/County ringdown line to notify State/County officials of a declared emergency.

   3.3 The Radiological Manager or Radiological Director is responsible for activating the Health Physics Network (HPN) if requested by the NRC, to inform the Health Physics Section of the NRC of the emergency radiological environmental conditions and to coordinate health physics information and response during a declared emergency at STP.

   3.4 The Manager, Information Technology or designee is responsible for the installation, testing, maintenance, and modifications of the emergency communications systems.

4. Emergency Communications System

> **NOTE**
>
> Refer to Addendum 2, Notification Methods to Offsite Agencies, for alternate telephone numbers and notification methods to be used throughout this procedure.

4.1 Emergency Telephone Circuits

4.1.1 Emergency Notification System (ENS)

- The ENS is a telephone circuit provided by the NRC and is terminated on an FTS 2001 telephone. The principal method of communications with the NRC is the ENS. The circuit may also be activated by the NRC. The ENS is activated to notify the NRC of declared emergency and to maintain communications with the NRC Operations Center.

- IF the ENS is determined to be out of service and upon subsequent return to service, THEN notify the NRC Operations Center.

- ACTIVATE the ENS by lifting the handset on the telephone and dialing the appropriate number.

4.1.2 State and County Ringdown Line

- The State/County ringdown line is provided to notify State and County officials of a declared emergency. The State/County ringdown line is an automatic ringdown telephone circuit terminated on a communications console OR an ORANGE telephone.

- ACTIVATE the State/County ringdown line by:
  - LIFTING the HANDSET on the ORANGE telephone

    OR

  - UTILIZING the communication console in accordance with Step 4.8, Communications Console System.

4.1.3 Health Physics Network (HPN)

- The Health Physics Network (HPN) is a telephone circuit provided by the NRC and is terminated on an FTS 2001 telephone. It is to be used only at the request of the NRC. The HPN telephone is designed to provide communications with the NRC Health Physics Section and/or other nuclear power plants during a declared emergency. STP health physics personnel MAY request a conference call with other nuclear power plants on the HPN by asking the NRC to connect the desired plant(s).
- IF the HPN telephone line is determined to be out of service and upon subsequent return to service, THEN notify the NRC Operations Center. (IEN 89-19)
- ACTIVATE the HPN by lifting the handset on the telephone and dialing the appropriate number.

4.1.4 STP Coordinator Ringdown Line

- The STP Coordinator ringdown line is an automatic ringdown between the Qualified Scheduling Entity (QSE) and STP communications consoles.
- Utilize the communications console in accordance with Step 4.8, Communications Console System.

4.2 Telephone System

4.2.1 The STP Telephone System consists of company owned and maintained telephone switching equipment and cable. The onsite system is connected to regular telephone services via an onsite demarcation point. The offsite services are provided by Verizon and Southwestern Bell Telephone. Offsite commercial telephone services are augmented by a Center Point Energy owned and operated microwave system. The microwave system provides telephone and data services via tie lines into the Houston corporate offices. The corporate office telephone system interconnects into the local telephone system in Houston. The combined microwave and corporate office telephone systems provide augmentation to the normal local onsite - offsite telephone services at STP.

4.2.2 Calling in (from offsite) may be accomplished in one of two ways:

- Direct inward dialing (DID), OR
- Calling the site number of (361) 972-3611 and using the automated attendant. Direct inward dialing extensions begin with a 4, 7 or 8. All others must go through the automated attendant.

4.2.3 Calling offsite (from onsite) may be accomplished in one of two ways:

- DIAL 9-1-AREA CODE - telephone number, OR
- DIAL 32-0 to Center Point Energy and have the Operator complete the call.

4.2.4 Onsite calling is accomplished by dialing the desired extension number.

~~4.2.5 Two (2) mobile cellular telephones are provided to Offsite Field Teams as a~~ back up to radio communications.

4.3 Portable Satellite Telephone

NOTE

Portable, independent satellite telephones are provided to the Station as a backup to all company owned and commercial telephone equipment/services. These telephones can be utilized for worldwide access via satellite.

4.3.1 Need clear view of the sky, outdoors, away from buildings and tall structures.

~~4.3.2 Turn the phones power on/off: Press and hold the Power Button for 1 to 2~~ seconds see Addendum 5.

4.3.3 Rotate and pull extend antenna into vertical position.

4.3.4 To dial, press and hold the 0+ button until the display shows a + sign (the + sign is an international calling code), then proceed to dial just like any other long distance call (1 + area code + phone number).

4.3.5 When you finish dialing, press OK to make the call.

4.3.6 Talk with antenna above your head and vertical to the ground.

4.3.7 When you complete the call press OK again to hang up.

4.3.8 Each portable satellite telephone is labeled with number and required codes for an outside caller to call back.

4.3.9 To retrieve Voice Mail messages perform the following:

- Dial the satellite telephone number.
- During the voice greeting, enter *.
- When prompted for your password, enter 1111.
- Follow the voice prompts to:

  a. Play your messages.

  b. Record a Message.

  c. Change your greeting.

  d. Access personal options.

  e. Make a call.

4.4 Desktop Satellite Telephone

> NOTE
>
> Independent desktop satellite telephones are provided to the Station as a backup to all company owned and commercial telephone equipment/services. These telephones can be utilized for worldwide access via satellite. A desktop satellite telephone is maintained in both control rooms, both Technical Support Centers, and the Emergency Operations Facility.

4.4.1 Although similar in many respects to a normal telephone, the desktop Satellite Telephone has some differences:

- Pick up the telephone handset and listen; you should hear the normal steady state dial tone. The Satellite Terminal Box call status indicator should also shine green continuously. A continuous orange indicator signifies acceptable but marginal signal strength. The Satellite Terminal Box is located in the EOF Communications Room, and in the Unit 1 and 2 TSC Copy Room's on the communications rack.

a. If you hear nothing, there is possibly something wrong with your telephone or cable. Refer to Addendum 6, Desktop Satellite Telephone Troubleshooting.

b. If you hear a single tone interrupted every few seconds by silence, check that the signal strength indicator is orange or green. If not, there may be a problem with your SIM card. Refer to Addendum 6, Desktop Satellite Telephone Troubleshooting.

- Dial 001 + area code + phone number. Once you have entered the phone number you will hear progress pips from the Iridium network. It can take up to 30 seconds for the Iridium network to connect a call, so a pause at this stage is not unusual.

- Eventually you will hear the other end ringing, or hear a busy tone and voice message indicating why your call was unsuccessful. When the other party answers the call status indicator will change from steady orange to flashing orange, indicating a call in progress.

- To terminate the call just hang up the handset. The call status light turns off.

4.4.2 Each desktop satellite telephone is labeled with number and required codes for an outside caller to call back.

4.4.3 To retrieve Voice Mail messages perform the following:

- Dial the satellite telephone number.

- During the voice greeting, enter *.

- When prompted for your password, enter 1111.

- Follow the voice prompts to:

  a. Play your messages.

  b. Record a Message.

  c. Change your greeting.

  d. Access personal options.

  e. Make a call.

4.5 Radio Communications

4.5.1 The Radio Communications System consists of repeaters, mobile, handheld, and base two-way FM transceivers licensed to STP Nuclear Operating Company by the Federal Communications Commission. The radio repeaters are installed in a radio communications building at the base of the radio antenna tower onsite. The repeaters are supplied normal power from the plant power and emergency power from an automatic starting engine driven generator. The generator is supplied fuel from a local fuel tank. The handheld, mobile and base stations are programmed to operate through the repeaters or direct.

4.5.2 Radio communications with the Matagorda County Emergency Operations Center is accomplished by the use of a radio transmitter/receiver in the Security Central and Secondary Alarm Stations, and a transmitter/receiver at the Matagorda County Sheriff's Office tuned to an STP radio frequency.

4.5.3 Offsite Field Team radio communications are accomplished on STP Nuclear Operating Company licensed radio channels. The repeaters provide coverage of the ten mile Emergency Planning Zone from one handheld radio to another handheld radio or to a base station.

CAUTION

Handheld radios SHALL NOT be used to transmit from inside the ESF Switch Gear Room, Control Room, Technical Support Center, Emergency Operations Facility, Auxiliary Shut Down Panel Rooms, Computer Rooms, nor within ten (10) feet of an open instrument cabinet, computer or computer terminals. The only exceptions to the above restrictions are emergencies where a threat exists to the plant OR human safety and no other means of emergency communications are available.

4.5.4 PERFORM the following to use a radio for communication:

- ALIGN the assigned radio channel on the handheld by selecting the appropriate channel number and Modes A and B for repeater or Mode C for direct communication.
- PRESS the microphone button and talk, keeping the microphone about 2 inches in front of the mouth, and
- RELEASE the microphone button to receive, AND ADJUST the volume by turning the knob marked VOL.

- ADJUST the squelch by turning the knob marked SQUELCH until noise is heard, then back until the speaker is quiet. This setting is for the maximum sensitivity, only on mobile radios.

- Communicate with other portable, mobile or base radio stations.

4.6 Glenayre Paging System

4.6.1 The Glenayre Paging System is a tone system that may be activated from plant telephones or from an offsite touch-tone telephone. The system has a range of over a 60-mile radius from the site. The system transmitters are connected to emergency power generators with automatic starting equipment.

4.6.2 Instructions for activating the Glenayre Paging System are contained in 0ERP01-ZV-IN03, Emergency Response Organization Notification.

4.7 Maintenance Jack Communications System

4.7.1 A maintenance jack amplified and sound-powered telephone system is available for onsite communication between certain areas. Refer to Addendum 4, Related Maintenance Jacks. The system is powered by amplifiers on pre-designed circuits. Each circuit may be activated or combined with another circuit by the proper selections on the system control panels located in each Control Room. The system has the capability to be voice activated. The voice-activated circuit is one loop that interconnects each of the maintenance jack terminals into one circuit.

4.7.2 IF it is desired to have amplified voice communications, THEN PERFORM the following:

- SELECT the desired zones on the selection panel in the Control Room.

  - INSERT a headset plug into one of the jack stations marked 1 or 2 at the area.

  - INSERT a headset plug into the jack marked plant for voice-powered communications at the desired jack station.

4.8 Communications Console System

4.8.1 The communications console is an integrated communications panel and switching system which is subdivided into seven groups: direct line (ringdown), telephone, radio (RF), public address (PA), alarm system, conference, and voice direct line (VDL). Refer to Addendum 1, Communications Console Panel, for locations of the console controls. Each communications group is composed of several two-position switches. These positions are:

- MONITOR - Top position (amber light will glow)
- TALK/LISTEN - Down position (green light will glow)

4.8.2 When the TALK/LISTEN switch is activated (green light) for the State/County Ringdown line this locks out all other communications consoles. When the call is completed, deactivate by depressing the TALK/LISTEN switch a second time to clear the green light.

4.8.3 These Consoles are installed in the Control Rooms, Auxiliary Shutdown Panel Rooms, Operations Support Centers, Technical Support Centers, Emergency Operations Facility, Security Force Supervisor's Office, Central and Secondary Alarm Stations, Simulator and in the Maintenance Office Facility. During Refueling Outages, console(s) may be installed on the applicable units One Stop Shop.

NOTE

Many circuits may be monitored simultaneously. These circuits are heard through the left ear if using the headset. The volume for the monitor position is controlled by the MONITOR VOLUME control located in the Handset/Headset Control Group.

Usually the communicator operating the console will be talking (TALK/LISTEN switch is activated) on only one circuit at a time. These conversations will be heard through the right ear if using the headset. The volume control for the TALK/LISTEN position is controlled with the RECEIVE VOLUME control also located in the Handset/Headset Control Group.

The communicator may actively communicate with all circuits simultaneously. It is important to note that all circuits with the TALK/LISTEN switch activated will hear the communicators conversation, which may not be desirable. To deactivate, depress the TALK/LISTEN switch a second time to clear the green light.

4.8.4 Direct Line (Ringdown) Group Operation

CAUTION

Activating the circuit select switch (CSS) in the MONITOR (top position) will activate an Idle Circuit and cause the ringdown line to ring. The position switch SHALL be in the TALK/LISTEN (bottom position) before speaking.

a. WHEN it is desired to place a call, THEN perform the following:

NOTE

The next step will ringdown the other phone.

- Activate the appropriate circuit select switch in the TALK/LISTEN position.

- WHEN the phone is answered, THEN PRESS the push-to-talk button when speaking.

- WHEN communication is terminated, THEN DEACTIVATE the bottom TALK/LISTEN position switch.

NOTE

An audible signal will be heard through the speaker and the CSS red lamp will flash when a party is calling.

b. WHEN a call is received, THEN perform the following:

- ACTIVATE the bottom TALK/LISTEN position switch.

- WHEN it is desired to talk, THEN press the push-to-talk button when speaking.

- WHEN communication is terminated THEN deactivate the bottom TALK/LISTEN position switch.

4.8.5 Telephone Group Operation

> NOTE
>
> All normal site phone functions are available through the console.

a. IF it is desired to make a call, THEN PERFORM the following:

- ACTIVATE the circuit select switch for selected extension in the TALK/LISTEN (bottom) position AND WAIT until a dial tone is received on the headset or handset.
- DIAL the number using the telephone keypad.
- WHEN the number called answers, THEN PRESS the push-to-talks button while speaking.
- WHEN communication is terminated, THEN DEACTIVATE the TALK/LISTEN switch.

b. WHEN a call is received, THEN PERFORM the following:

> NOTE
>
> An audible signal will be heard through the speaker and the CSS red light will flash when another party is calling.

- ACTIVATE the circuit select switch (CSS) in the TALK/LISTEN (bottom position.
- WHEN it is desired to talk, THEN PRESS the push-to-talk button while speaking.
- WHEN communication is terminated THEN DEACTIVATE the two position TALK/LISTEN switch.
- IF it is desired to place a call on hold, THEN ACTIVATE the MONITOR switch.

4.8.6 Radio Group Operation

NOTE

Radio channels may be monitored by moving the circuit select switch (CSS) to the MONITOR (top) position.

a. IF it is desired to transmit a message on a radio frequency, THEN activate the circuit select switch to the TALK/LISTEN (bottom) position.

b. PRESS the push-to-talk button when speaking.

c. WHEN communication is terminated THEN deactivate the bottom TALK/LISTEN position switch.

4.8.7 Plant Public Address and Alarm System

NOTE

Emergency alarm and public address override switch capabilities are found on the communications console panels in the following locations: all panels in each Unit's Control Room, and Technical Support Center, the Emergency Operations Facility, Central Alarm Station, Secondary Alarm Station, and the Simulator.

a. IF it is desired to make a public address announcement, THEN perform the following:

- SELECT the two position switch corresponding to the desired zone (listed on Addendum 3) that is to receive the announcement.
- Activate the two position switch(es) to the TALK/LISTEN (bottom) position in the appropriate zone(s).
- PRESS the push-to-talk button when speaking.
- Deactivate the bottom TALK/LISTEN position switch at the conclusion of the announcement.

b. Emergency Public Address Alarms and Announcement

> NOTE
>
> There are three public address emergency alarms: Assembly, Fire, and RCB Evacuation Alarm.
>
> Alarms will be broadcast as directed over the PA system. Alarm switches actuate for 8 seconds, then disconnect unless the PUSH-TO-TALK button on the handset is depressed.

- WHEN directed, THEN select the appropriate alarm.
- WHEN the alarm is completed, THEN DEACTIVATE the alarm switch, activate the Unit override switch, AND make the appropriate emergency announcement over the PA system as directed.
- WHEN the alarm/announcement is completed, THEN deactivate all switches.

4.8.8 Conference Network

> NOTE
>
> Loops may be monitored for informational purposes by selecting the MONITOR circuit select switch.

a. PERFORM the following to establish group conference:

- VERIFY that all conferring parties are on the same loop.
- VERIFY that all conferring parties on the loop have the circuit select switch (CSS) in the TALK/LISTEN (bottom) position.
- WHEN it is desired to talk, THEN press the push-to-talk button when speaking.
- WHEN communication is terminated, THEN deactivate the bottom TALK/LISTEN position switch.

4.8.9 Voice Direct Line (VDL)

a. The Voice Direct Line (VDL) is a direct line from Quintron communication console to console.

- Lift the handset on the appropriate console.
- Activate the appropriate circuit selector switch on the communication to the TALK/LISTEN position.

5. Maintenance

5.1 Information Technology personnel SHALL maintain the emergency communications systems.

5.2 Maintenance SHALL be done as required to keep the system in good operating condition and as committed to in license documents.

6. References

6.1 NUREG-0654/FEMA-REP-1, Criteria For the Development and Evaluation of Emergency Preparedness in Support of Nuclear Power Plants

6.2 South Texas Project Electric Generating Station Emergency Plan

6.3 0PGP07-ZA-0011, Communications Systems

6.4 0ERP01-ZV-IN03, Emergency Response Organization Notification

6.5 IEN 89-19, Health Physics Network

7. Support Documents

7.1 Addendum 1, Communications Console Panel

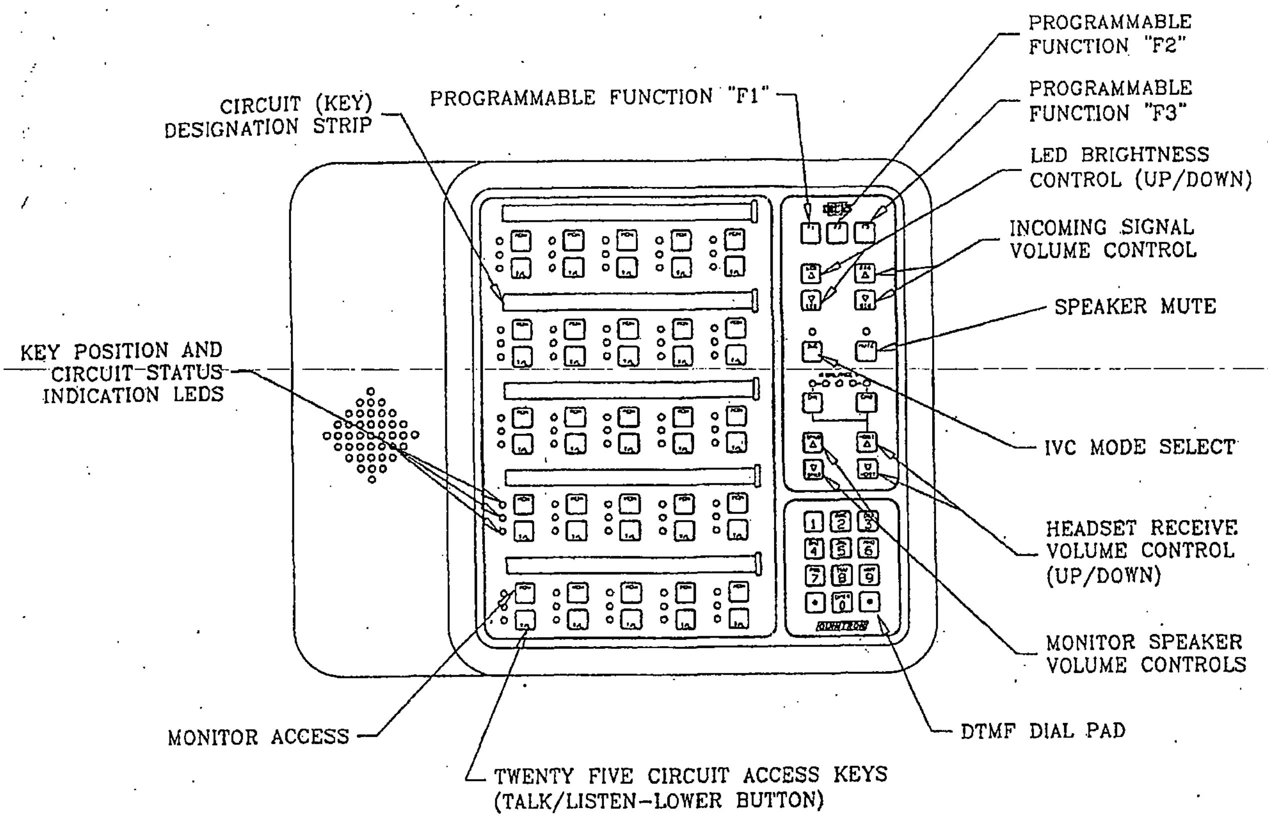
7.2 Addendum 2, Notification Methods to Offsite Agencies

7.3 Addendum 3, Station Public Address Selections

7.4 Addendum 4, Related Maintenance Jacks

7.5 Addendum 5, Portable Satellite Telephone

7.6 Addendum 6, Desktop Satellite Telephone Troubleshooting



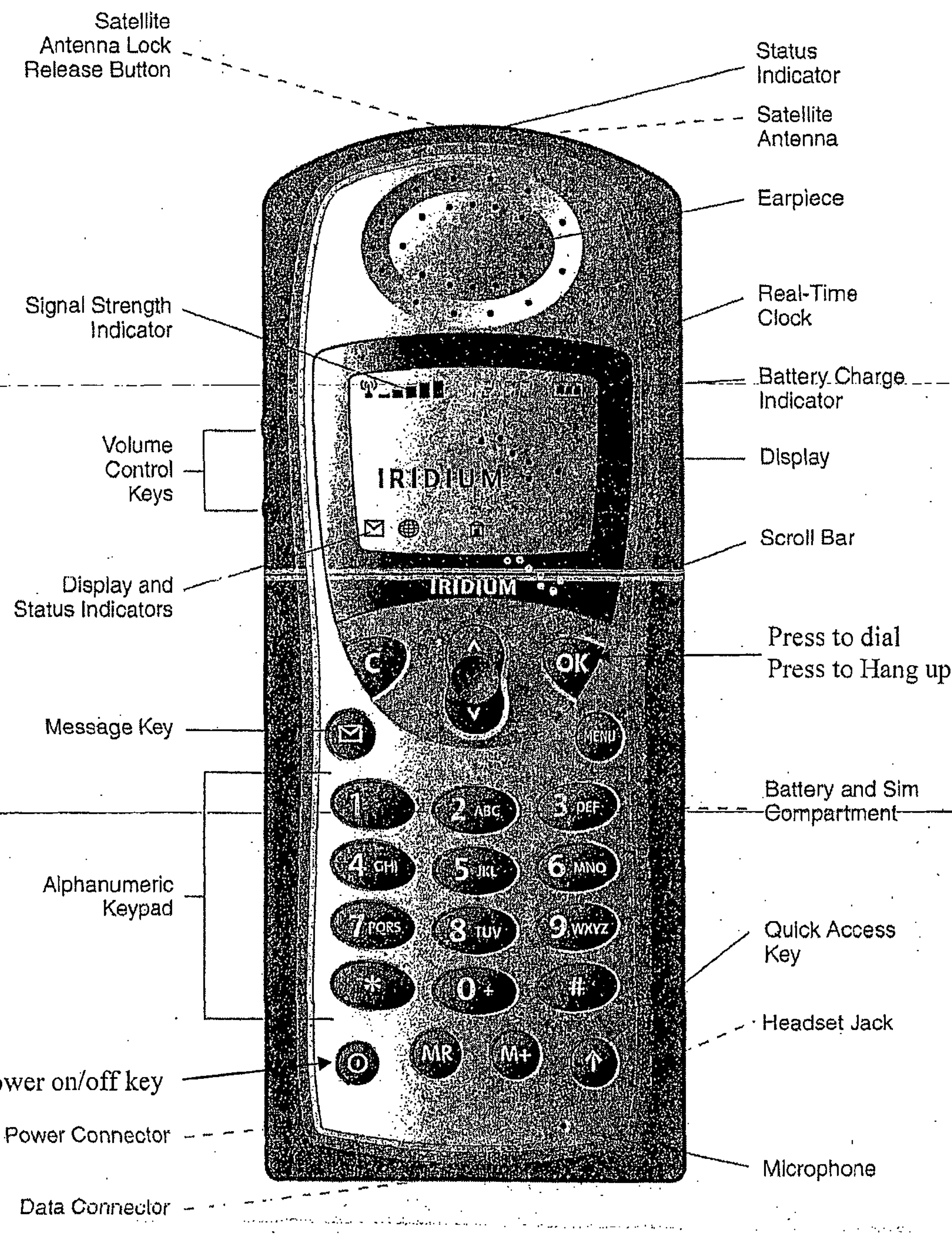
25 KEY DESKTOP

ENS	STATE (DPS, PIERCE)	COUNTY (SHERIFF)	HPN
1-301-816-5100	1-979-541-4595	1-979-241-3205	1-301-816-5100
1-301-951-0550	N/A	1-979-244-1178 (ONLY when EOC is Activated)	1-301-951-0550

	NRC	State/County
ENS	X	
Ringdown Line to the DPS, Disaster District Sub 2C (State of Texas) and the Matagorda County Sheriff's Office (Matagorda County).		X
Outside Telephone Lines.	X	X
Satellite Telephone.	X	X
Unit 1 Control Room Direct Line to Bay City.	X	X
Microwave Line to Center Point Energy and call forwarded to the appropriate number.	X	X
~~Ringdown Line to the STP Coordinator (QSE) and request~~ the call be forwarded to the appropriate number.	~~X~~	~~X~~
Security Radio communication to the Matagorda County Sheriff's Office and request the call information be passed onto the appropriate number.	X	X

Unit 1 ALL	Unit 2 ALL	Units 1 & 2 ALL	Unit Override	50 Telephone	51 Telephone
Zone 1	**Zone 2**	**Zones 1,2, & 3**	**Zones 1 - 4**	**Zone 3**	**Zone 4**
Electrical Auxiliary Building (EAB) Mechanical Auxiliary Building (MAB) Isolation Valve Cubicle (IVC) Reactor Containment Building (RCB) Fuel Handling Building (FHB) Diesel Generator Building (DGB) Turbine Generator Building (TGB)	Electrical Auxiliary Building (EAB) Mechanical Auxiliary Building (MAB) Isolation Valve Cubicle (IVC) Reactor Containment Building (RCB) Fuel Handling Building (FHB) Diesel Generator Building (DGB) Turbine Generator Building (TGB)	Unit 1 & 2 Yard	All Zones simultaneously with activated prioritization circuitry	Essential Cooling Water Intake Structure (ECWIS) Circulating Water Intake Structure (CWIS) Lighting Diesel Generator Building (LD) Load Center Buildings 12J, 12K, 12L, 12M and the Electrical Load Center Building (EL) Hypochlorination Make Up Demineralizer (MUD) South/East Load Center Building Fire Pump House North, East and West Gate Houses Units 1 and 2 Main and Standby Transformer Emergency Transformer Fuel Storage Building Low Level Waste Building CWS Load Center Warehouse and Machine Shop Units 1 & 2	Nuclear Support Center (NSC), Nuclear Training Facility (NTF) Owner Controlled Area

		UNIT 1	UNIT 2
TRANSFER SWITCH PANEL	TRAIN A	ESF1	ESF1
TRANSFER SWITCH PANEL	TRAIN A	ESF2	ESF2
TRANSFER SWITCH PANEL	TRAIN B	ESF8	ESF3
TRANSFER SWITCH PANEL	TRAIN B	ESF9	ESF9
TRANSFER SWITCH PANEL	TRAIN C	ESF10	ESF10
TRANSFER SWITCH PANEL	TRAIN C	ESF11	ESF11
STANDBY DIESEL GENERATOR CONTROL PANEL	TRAIN A	1SDG3	2SDG3
STANDBY DIESEL GENERATOR CONTROL PANEL	TRAIN B	1SDG2	2SDG2
STANDBY DIESEL GENERATOR CONTROL PANEL	TRAIN C	1SDG1	2SDG1
CHILLER CONTROL PANEL, COLUMN 18V		TGI-17	TGI-17
BORIC ACID TANK ROOM ELE. 29' MAB, ROOM 076		RW-16	RW-16
CCW SURGE TANK ROOM ELE. 60' MAB		MA-18	MA-18
ESSENTIAL CHILLED WATER INTAKE STRUCTURE	TRAIN A	1YD5	2YD8
ESSENTIAL CHILLED WATER INTAKE STRUCTURE	TRAIN B	1YD6	2YD9
ESSENTIAL CHILLED WATER INTAKE STRUCTURE	TRAIN C	1YD7	2YD10
AUXILIARY FEEDWATER STORAGE TANK AREA, COLUMN 19Q		TGI-12	TGI-12

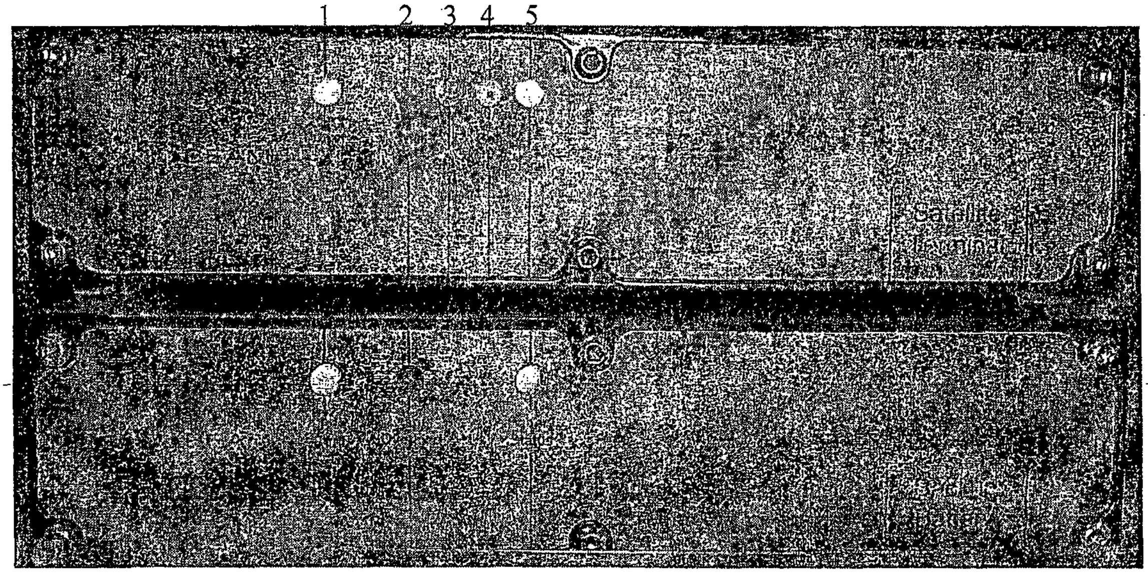


Satellite Antenna Lock Release Button
Status Indicator
Satellite Antenna
Earpiece
Signal Strength Indicator
Real-Time Clock
Battery Charge Indicator
Volume Control Keys
Display
IRIDIUM
Scroll Bar
Display and Status Indicators
Press to dial
Press to Hang up
Message Key
Battery and Sim Compartment
Alphanumeric Keypad
Quick Access Key
Headset Jack
Power on/off key
Power Connector
Microphone
Data Connector

REMOTE SATELLITE TERMINAL (RST-100)

Front Panel

1 2 3 4 5



Status LEDS are located on the front panel, they show the RST-100 status.

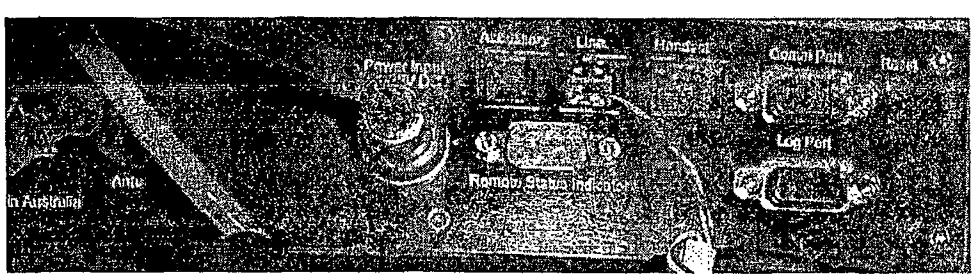
1. Power 2. Voicemail waiting 3. SMS waiting 4. Call status 5. Signal strength

Once powered up, the RST-100 attempts to register with the Iridium network. The signal level LED uses color to indicate how strong the Iridium signal is at your location.

Indicator color	Signal strength
Green	Strong
Orange	Acceptable
Red	No signal, problem with installation

In most cases the indicator will show green after a short period of approximately 15 seconds - orange indicates an acceptable but marginal signal strength. If the indicator remains red, there is a problem with the installation.

Rear Panel



This table provides information to help you troubleshoot problems encountered while using the Desktop Satellite Telephone (RST-100). If the problem continues contact IT Communications at extension 7000 or if the ERO/Storm Crew is activated the Communications Systems Supervisor.

QUESTION	ANSWER
No lights on the front panel of the Satellite Terminal Box.	Check power is connected.
RST-100 fails to register with the Iridium service after 30 seconds.	Press reset button located on the rear of the Satellite Terminal Box.
No dial tone.	Check if a data call is in progress and power is connected and equipment is in a normal state.
Cannot make call, two tone signal heard.	Phone requires a PIN. See step 1 below.
You can't make calls.	Check that the antenna is properly mounted. Did you enter the number in international format? Check the signal strength. If the signal is weak, wait a few minutes for thick cloud cover to move. Has a new SIM card been inserted?
You can't receive calls.	Check the antenna. Is it properly mounted? Check the signal strength. If the signal is weak, wait a few minutes for thick cloud cover to move. Check the telephone ringer setting to see if it is off.
The Voicemail indicator keeps flashing.	There is not enough memory available to store another message. Or there is a message waiting. Delete messages and free up some space.

1. Personal Identification Number (PIN):

   Your RST-100 may require a PIN, this will be indicated by the Signal light flashing Red and a distinctive dial tone consisting of two alternating tones.

   If a distinctive, two-tone dial tone is heard, one of two access codes is required - the SIM PIN or the PIN Unlock Code (PUK).

   - The SIM PIN is required if the two tones are of equal length. If so simply enter the four digit PIN (1111) and await a change of tone (up to ten seconds), then hang up. If the PIN was correct the phone will register and you may proceed with normal use as described below.
   - The PUK is required if the PIN has been incorrectly entered three times and is indicated when the high tone is longer than the low tone. Contact communications at extension 7000 or if the ERO/Storm Crew is activated the Communications Systems Supervisor

2. Voicemail

   - If a Voicemail message has been left for you, the RST-100 flashes the Voicemail Waiting indicator. The indicator is cleared whenever the user connects to the Voicemail retrieval number.

3. Facsimile Support

   - The Iridium network does not support facsimile transmission.

CA1

NOTE

Step 3.0 will determine if leakage is actual RCS leakage.

____ **3.0 CHECK Trends For Any Of The Following Indications Of RCS Leakage:**

Go TO Step 5.0.

- Rad Monitor RT8011 Particulate – Rising
- Reactor Coolant Drain Tank Level – Rising
- Pressurizer Relief Tank Level – Rising
- RCB Normal Sump Level – Rising

____ **4.0 PERFORM One Of The Following To Determine The RCS Leak Rate:**

- 0PSP03-RC-0006, Reactor Coolant Inventory

  OR

- DETERMINE the RCS leak rate using pressurizer level, VCT level, and comparing charging and letdown flows

STEP DESCRIPTION FOR 0POP04-RC-0003 STEP 3.0

STEP: CHECK Trends For Any Of The Following Indications Of RCS Leakage:

- Rad Monitor RT8011 Particulate – Rising
- Reactor Coolant Drain Tank Level – Rising
- Pressurizer Relief Tank Level – Rising
- RCB Normal Sump Level – Rising

PURPOSE: To determine if leakage is from RCS and not CVCS.

BASIS: Indication of RT8011, RCDT, PRT or RCB Normal Sump levels rising will confirm that the leakage is from RCS and not CVCS which is normally tied to the RCS.

ACTIONS: Monitor trends from RT8011, RCDT, PRT or RCB Normal Sump.

INSTRUMENTATION: Level indications located on CP004 and various plant computer monitors located in control room. Radiation Monitor Computer RM-11.

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A

7.43 MONITOR Plant Computer group RH-12 (8412) OR TREND the following points for the applicable pump:

- "RHR PUMP 1A(2A)" RHFE0867 RHIA0880
- "RHR PUMP 1B(2B)" RHFE0868 RHIA0881
- "RHR PUMP 1C(2C)" RHFE0869 RHIA0882

CAUTION

- DO **NOT** start an RHR pump with vessel level below 32 ft 9 inch. (6 inches above hot leg centerline)
- WHEN the DG is being paralleled OR operated in parallel with offsite power, THEN the associated Trains "RHR PUMP" **SHALL NOT** be started or operated: (CR 05-4915)

7.44 ENSURE the associated train's Emergency Diesel Generator for the pump to be started in the next step is **NOT** being paralleled OR operated in parallel with offsite power. (CR 05-4915)

7.45 START the desired RHR pump:

- "RHR PUMP 1A(2A)"
- "RHR PUMP 1B(2B)"
- "RHR PUMP 1C(2C)"

7.43 MONITOR Plant Computer group RH-12 (8412) OR TREND the following points for the applicable pump: ____

- "RHR PUMP 1A(2A)" RHFE0867 RHIA0880
- "RHR PUMP 1B(2B)" RHFE0868 RHIA0881
- "RHR PUMP 1C(2C)" RHFE0869 RHIA0882

CAUTION

- DO **NOT** start an RHR pump with vessel level below 32 ft 9 inch. (6 inches above hot leg centerline)
- WHEN the DG is being paralleled OR operated in parallel with offsite power, THEN the associated Trains "RHR PUMP" **SHALL NOT** be started or operated: (CR 05-4915)

7.44 ENSURE the associated train's Emergency Diesel Generator for the pump to be started in the next step is **NOT** being paralleled OR operated in parallel with offsite power. (CR 05-4915) ____

7.45 START the desired RHR pump:

- "RHR PUMP 1A(2A)" ____
- "RHR PUMP 1B(2B)" ____
- "RHR PUMP 1C(2C)" ____

CA2

PROCEDURE PURPOSE

This procedure provides guidelines for the initial response and stabilization of the plant in the event of a loss of any single or all 13.8 KV bus(es) or 4.16 KV Bus(es). This includes all 13.8 KV Auxiliary and Standby buses, and 4.16 KV buses with the exception of Buses 1K(2K), 1L(2L) and 1M. Loss of a 4.16 KV ESF bus is addressed as it indicates at least a partial loss of offsite and onsite AC power (ESF bus power can only be completely lost if both offsite and onsite power sources to the specific bus are lost).

MAJOR ACTION CATEGORIES

- Provide interface with Emergency Operating Procedures and provide the instructions to establish the minimum equipment required to safely stabilize the unit.
- ~~Identify actions associated with commitments to perform the action within a specified time period~~ after the initiating event.

DISCUSSION:

The electrical distribution system at STP has by design, a high degree of flexibility and ability to withstand casualties, especially the Class 1E alternating current systems. However throughout the nuclear industry Loss Of Offsite Power (LOOP) events have occurred as well as Station Blackout (loss of all offsite and onsite AC power) events.

When dealing with a loss of offsite AC power, both complete and partial, with the Unit in Modes 1 or 2, the loss of an Auxiliary power bus will result in the loss of a Reactor Coolant Pump requiring a reactor trip because STP is not analyzed for operation with only three Reactor Coolant Pumps. In the event that no ESF bus is available the indication is that all offsite and onsite AC power has been lost requiring ~~transition to the Emergency Operating Procedures.~~ Under these same conditions STP has committed to shed the Channel I Load Sequencer from its power supply within the first 30 minutes after the initiating event, and if the associated battery bank has a jumpered cell then all the loads on DP 1201 and DP 1204 will be shed except for QDPS and SG PORVs.

The initial response provided by this procedure is directed to the stabilization of critical plant parameters and then analyzes the extent of the loss of power. While this procedure does not identify the specific combination of buses that have been lost, it does identify the specific area of the power loss so that a procedure that is more specific to the method for power restoration can be referred to.

STEP DESCRIPTION FOR 0POP04-AE-0001 STEP 3.0

STEP: CHECK 4.16 KV ESF Bus Status:

a. ANY 4.16 KV ESF Bus NOT energized from offsite power (VERIFY the voltage on all three phases of each ESF Bus).

b. VERIFY Applicable STBY DG(s) running

c. VERIFY Applicable STBY DG(s) output breaker(s) closed to the associated 4.16 KV ESF bus

PURPOSE: To determine the status of the 4.16 KV ESF buses and performs any corrective actions that can be performed under the current conditions.

BASIS: This step attempts to start SDG for a de-energized bus. Also this step ensures output breaker is closed and if not determines the cause of the failure and provides steps to correct and energize the bus.

If "4KV BUS O/C LOCKOUT" indicating lamp on applicable BSMP {CP003} is illuminated the bus cannot be energized until corrective maintenance is complete.

ACTIONS: Determine if the SDG is available to be started by checking for O/C lockout and other fault protection. If available then perform the steps to start SDG and close output breaker.

If the SDG is already running at this step, then determine the need to close the affected SDG breaker to energize the associated bus and close the breaker in the event that no faults exist. If a fault does exist, then the cause of the fault would have to be corrected before protective actuation device can be reset and the bus energized.

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: If the SDG has a 4.16 KV ESF Bus overcurrent lockout, SDG generator differential lockout or an SDG overspeed lockout then these faults will need to be corrected and reset to energize the bus.

PROCEDURE PURPOSE

The purpose of this procedure is to restore power to any ESF bus which is not energized. In the case where only one ESF bus is energized by a DG, and another one cannot be energized by the associated DG or offsite power, then steps are taken to operate breakers and disconnects to use the one running DG to supply key loads on another bus.

MAJOR ACTION CATEGORIES

- Tie the operating DG to another bus via the emergency switchgear bus 1L(2L).
- Energize at least one ESF bus from the Emergency Transformer.
- Control and load essential equipment on to the available ESF buses.

DISCUSSION:

STP has committed under specific conditions related to loss of offsite and onsite power to energize at least two ESF buses from a running DG in order to energize specific loads needed to extend station battery life or provide availability of ESF equipment that is electrically powered from one of two specific ESF buses.

NOTE

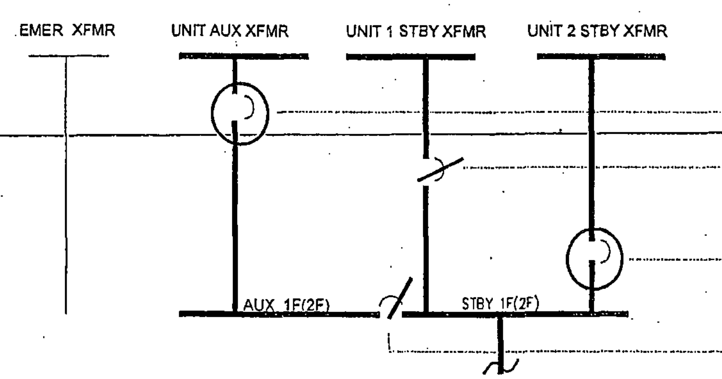
There are 5 possible lineups on Data Sheet 2, 3, and 4 for 13.8 KV XFMRS in the **DESIGNATED** Class 1E 4160 VAC Bus Power Source Table that meet Technical Specification requirements for being a power source for the 4.16 KV Buses:

- (1) UAT supplying the AUX BUS and STBY BUS
- (2) UAT supplying the AUX BUS and UNIT 1 STBY XFMR supplying the STBY BUS
- (3) UNIT 1 STBY XFMR supplying the STBY BUS and the AUX BUS
- (4) UAT supplying the AUX BUS and UNIT 2 STBY XFMR supplying the STBY BUS
- (5) UNIT 2 STBY XFMR supplying the STBY BUS and the AUX BUS

5.2 COMPLETE Required ESF Power Train Data Sheet 2 through 4 by performing the following steps.

5.2.1 RECORD actual breaker/disconnect positions for the 13.8 KV XFMRs, AUX BUS, STBY BUSES and from the 13.8 KV STBY BUS to the 480 VAC BUSES.

- RECORD "CLOSED" breaker/disconnect positions by drawing a line at an angle through the breaker.
- RECORD "OPEN" breaker/disconnect positions by drawing a CIRCLE around the breaker.



This example illustrates:
(3) UNIT 1 STBY XFMR supplying the STBY BUS and the AUX BUS

6.0 Acceptance Criteria

NOTE

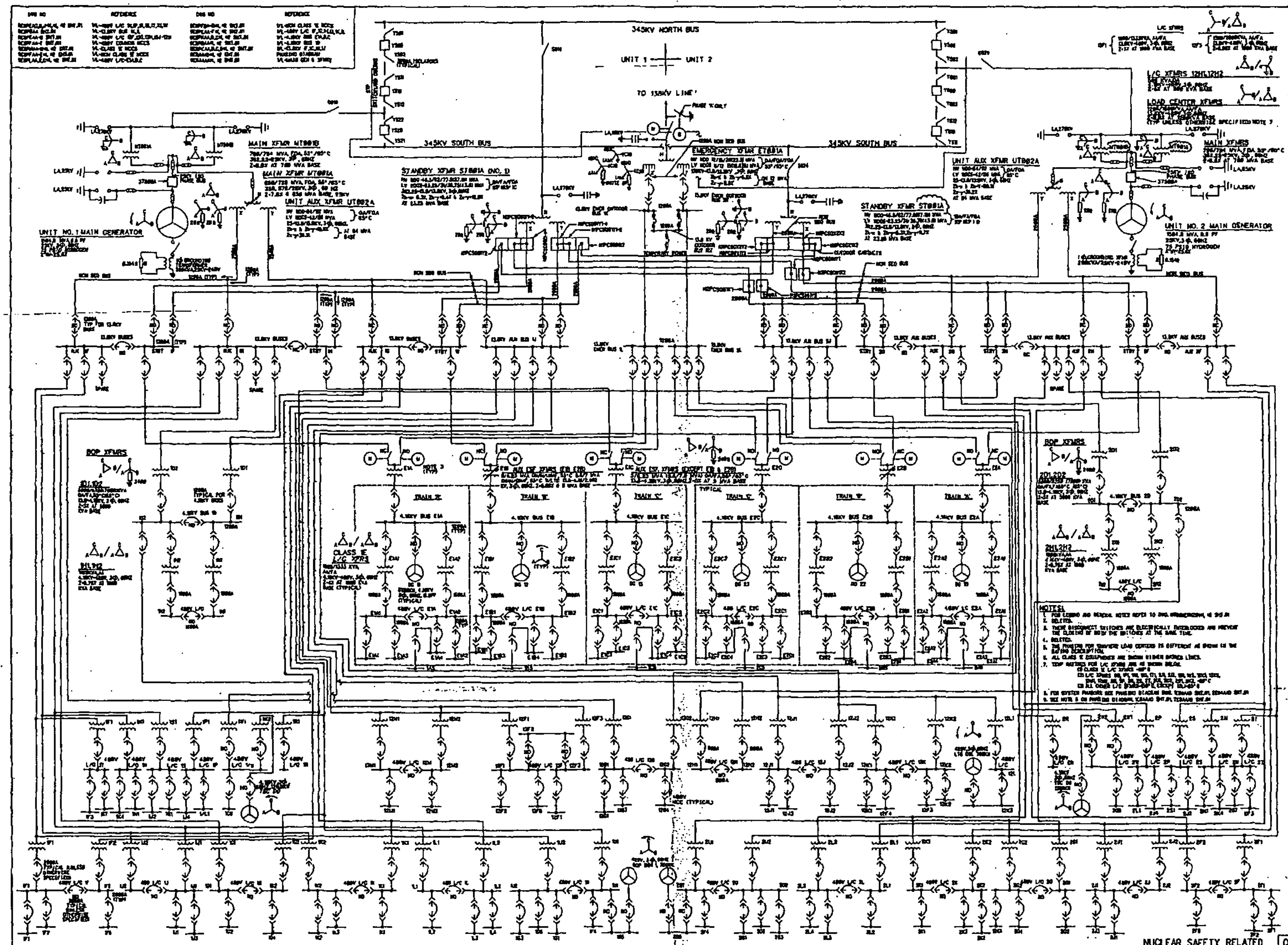
- Addendum 2, Two Physically Independent Circuits, provides a drawing of rights of way and offsite circuits to aide in the definition of "two physically independent circuits".
- Loss of one 13.8 KV Standby Bus to 4.16 KV ESF bus line constitutes loss of one required offsite source. (Reference 8.2)
- Loss of two 13.8 KV Standby busses to 4.16 KV ESF bus lines constitutes loss of two required offsite sources. (Reference 8.2)
- The preceding notes also apply when the 4.16 KV ESF bus is not energized by the 13.8 KV XFMR.
- Step 6.1 applies during standby diesel inoperability.
- Step 6.2 applies during offsite independent circuits inoperability.
- Note and Precaution 3.28 should be referred to for additional clarification regarding allowable indication to be utilized when obtaining 345 KV switchyard voltage.

6.1 Two physically independent circuits exist between the offsite transmission network and onsite Class 1E Distribution System as determined from Data Sheet 1, 2, 3, 4, and 9. (Technical Specifications 3.8.1.1.b, 3.8.1.1.f, and 4.8.1.1.1.a)

- North and South Bus in service with bus voltage:
  - ≥ 340 KV" for NORMAL LINEUP

    OR
  - ≥ 356 KV for NORMAL LINEUP with UAT or Train B ESF LTC in "MANUAL"

    OR
  - ≥ 358 KV for all ALTERNATE LINEUPs **OR** voltage specified in the "Minimum Voltage for Various Alternate 13.8 KV Bus Alignments" Addendum of 0POP02-AE-0002, Transformer Normal Breaker and Switch Lineup.
- Two of the following Rights of Way with a 345 KV line are available:
  - NW Right of Way 1 (White Point 39)
  - NW Right of Way 2 (Elm Creek 27 OR WA Parish 39 OR Elm Creek 18)
  - Eastern Right of Way (Dow Velasco 27 OR Dow Velasco 18)
- Two of the following 13.8 KV XFMRs are available:
  - Unit Aux XFMR
  - Unit 1 Stby XFMR
  - Unit 2 Stby XFMR
- Three 13.8 KV Standby Buses energizing the 4.16 KV ESF bus lines.



SOUTH TEXAS PROJECT NUCLEAR OPERATING COMPANY		SINGLE LINE DIAGRAM MAIN ONE LINE DIAGRAM UNIT NO. 1 & 2
33518968	NONE	00000E0AAAA
		01 24

NUCLEAR SAFETY RELATED 0

CA3

TABLE 1.2

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2.	STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3.	HOT STANDBY	< 0.99	0	$\geq$ 350°F
4.	HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5.	COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6.	REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

CA6

3.4 WATER LEVEL (FLOOD) DESIGN

The methods of analysis used to determine the design basis flood are discussed in Section 2.4. These methods are consistent with the requirements of Regulatory Guide (RG) 1.59.

The protection measures used to accommodate static and dynamic flood loads on Category I structures generally fall under the category of "incorporated barriers" as specified in regulatory position C.1 of RG 1.102.

3.4.1 Flood Protection

3.4.1.1 External Flood Protection Measures for Seismic Category I Structures. The flooding due to a postulated Main Cooling Reservoir (MCR) embankment breach produces the maximum water level around the power block structures as well as the controlling water elevations for buoyancy calculations. This is also the controlling phenomena in determining the maximum water level at the Essential Cooling Water Intake Structure (ECWIS). Studies and analyses on the MCR embankment have demonstrated that an adequate margin of safety can be maintained for all credible failure ~~mechanisms~~ (Section 2.5.6). Accordingly, mechanistic effects (such as scour and ~~erosion) associated with a postulated failure of the MCR embankment need not be evaluated.~~

The maximum water level on a vertical face at the south end of the plant structures is El. 50.8 ft mean sea level (MSL), which is El. 22.8 ft above plant grade. This maximum elevation occurs during a quasi-steady-state condition after a breach of the MCR embankment and is based on an instantaneous removal of approximately 2,000 ft of the embankment opposite the power block structures. This maximum elevation occurs on the south face of the Fuel-Handling Building (FHB) of Unit 1. The selection of postulated embankment breach widths and the assumptions made in determining the maximum flood elevations are described in Section 2.4.4.

Total inundation of the Essential Cooling Pond (ECP) occurs only under the condition of MCR embankment breach and does not affect the safe shutdown capability of the plant. The maximum water level calculated to occur at the ECWIS is El. 40.8 ft.

Safety-related structures, systems and components listed in Table 3.2.A-1 are protected against the effects of external flooding by:

1. Being designed to withstand the maximum flood level and associated effects and remain functional (such as seismic Category I structures and the Category I auxiliary feedwater storage tank), or

2. Being housed within seismic Category I structures which are designed as in item 1, above.

Flood protection of safety-related structures, systems, and components is provided for postulated flood levels and conditions described in Section 2.4.

Seismic Category I structures are designed to withstand the maximum flood levels by:

1. Having external walls and slabs of structures designed to resist the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady-state water level.

2. Ensuring the overall stability of the total structure against overturning and sliding due to the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady state water level, and

3. Ensuring that the total structure will not float due to buoyancy forces.

Figure 3.4-1 shows a general section through the plant. Figure 3.4-2 shows the seismic Category I Building maximum steady-state water surface profile, and the corresponding relationship of sill elevations for entrances to seismic Category I buildings.

Table 3.4-1 shows the results of hydraulic loading and buoyancy calculations which were done for the various safety-related facilities. The water depths shown on this table were developed from the maximum water surface elevations presented in Table 2.4.4-3.

An investigation of seismic Category I structures has been made for the flood levels and associated effects as previously described. The design for gross effects upon the structure incorporates safety factors greater than 1.1. All exterior seismic Category I building openings are located above the maximum steady-state flood level or are equipped with watertight doors when located below this profile, except as stated below.

Exceptions to the above-stated design basis for exterior building openings in seismic Category I structures are: (1) the opening for the truck bay in the radwaste loading area of the Mechanical-Electrical Auxiliaries Building (MEAB) and (2) the opening for the rail car access in the spent fuel cask loading area of the FHB. These areas are not protected from flooding because they do not have any safety-related systems and components located near or below the maximum flood level which is required to perform any essential function. In addition, the two areas are separated from the remainder of the building by walls which do not contain openings below the maximum water surface elevation corresponding to their location. The Tendon Gallery Access Shaftoover (TGAS) is provided with a watertight cover to prevent flood waters from entering the MEAB.

The safety-related equipment in the ECWIS is protected from the effects of the design basis flood. The personnel access doors on the west wall are provided with watertight doors; all other doors and openings are above the flood level. The dividing walls and doors between the ECWIS compartments minimize the potential for the propagation of flooding from one compartment to another.

The three maintenance knockout panels in the exterior walls of the Diesel-Generator Building (DGB), which are located below the maximum water surface elevation of 45.0 ft MSL, are watertight and designed for the hydrostatic forces. Each knockout panel allows access to only one of the three separate compartments within the structure, and only one panel may be removed at one time. The dividing walls between the compartments preclude propagation of flooding from one compartment to another.

The maintenance knockout panels in the exterior wall of the room, housing the component cooling water heat exchangers in the MEAB are located below the maximum steady-state water level shown on Figure 3.4-2. These panels are watertight. Since mechanistic effects from the MCR breach need not be evaluated, there is adequate time to replace the knockout panels for the remaining flood events of concern.

All exterior seismic Category I building wall and slab surfaces below grade are waterproofed. This conservatively protects the substructure of seismic Category I buildings from groundwater, which is expected to stabilize between El. 17 ft and 26 ft (1 to 10 ft below grade) after decommissioning of the dewatering system. No waterproofing is provided on exterior wall or slab surfaces above grade to protect against the effects of surge-wave run-up because of its short duration. All construction joints in exterior walls and slabs (except for localized areas of blockouts) are provided with waterstops to the maximum flood level for that location and can withstand hydrostatic and hydrodynamic effects.

All seismic joints between Category I structures contain dual 9-in. water stops capable of withstanding potential seismic and hydrostatic effects. Cracks in concrete are minimized by imposing strict QA and QC procedures on the quality of concrete and construction techniques.

Drains are provided with check valves such that the external flooding would not result in internal flooding through the inadvertent introduction of water through these drains into seismic Category I structures.

The duct banks are sealed so as to prevent backflow into safety-related areas. The cable in the duct banks is designed/specified for submerged installations.

Leakage from groundwater into the FHB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater inleakage occur, it is handled by the pumps in the FHB sump, the three-train compartment sumps, and the transfer cart area sump. For Unit 1 only, accumulated groundwater inleakage to the 64 degree tendon buttress area drains through a penetration in the RCB tendon gallery outer wall and is collected in the tendon gallery sump.

Leakage of groundwater into the MEAB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater leakage occur, it will be collected in sumps. Discharge from non-radioactive sumps are routed to the reservoir via a circulating water discharge line. Potentially radioactive discharge is pumped to the Liquid Waste Processing System (LWPS).

3.4.2 Analysis Procedures

3.4.2.1 Phenomena Considered in Design Load Calculations. For external flooding, the design basis events considered in design load calculations are as described in Section 3.4.1.

3.4.2.2 Flood-Force Application. The design flood conditions and elevations have been determined from an analysis of the phenomena discussed in Section 3.4.1.1.

In order to establish the controlling load conditions resulting from the embankment breach, both instantaneous surge wave runup as well as the longer term, quasi-steady-state conditions were analyzed. The wave runup condition conservatively assumes that the maximum total force perpendicular to the south face of the plant structures includes a dynamic component in addition to the associated hydrostatic forces. The quasi-steady state condition assumes that only the hydrostatic component contributes to the development of the total force for this case. The latter condition resulted in higher water surface elevations and greater hydraulic loads on power block structures.

The vertical buoyant loading condition is the force equal to the weight of water displaced by a structure. The discussion of lateral and vertical loadings is presented in the following subsections. Table 3.4-1 shows a summary of different lateral loadings at various locations around plant and ECP structures, caused by their respective controlling flood conditions. Procedures used to determine flood loadings are identified in Sections 3.4.2.2.1 and 3.4.2.2.2.

3.4.2.2.1 Lateral Loading:

3.4.2.2.1.1 Lateral Loading on the Power Block Structures – The analysis of the lateral force on the power block structures considered both the instantaneous wave runup and the quasi-steady state conditions. This analysis determined that the maximum total lateral force on the power block structures occurs when the maximum water level is reached during the quasi-steady state condition. Table 3.4-1 shows the controlling lateral forces (hydrostatic) exerted on different power block structures. These lateral forces are treated as triangular loadings on a vertical surface, varying at a rate of 62.4 lb/ft2/ft of structure depth. The procedures used to determine the dynamic and hydrostatic loadings for the above analysis conditions are discussed below:

1. Dynamic Force

The dynamic force on the south side of the power block structures is determined by application of linear momentum principles. The flow from the MCR is assumed to be normal to the south side of the power block structures. Therefore, the dynamic force exerted on the structures can be expressed by the following momentum equation (Ref. 3.4-2):

$$F = p\,Q\,V_o$$

where:

F = dynamic force normal to plant structure
p = density of flow
Q = flow rate
V_o = velocity of flow

The maximum value of pQv_o during surge formation is calculated. This is the contribution of momentum flux to the dynamic force. The contribution of the unsteadiness of momentum field is insignificant.

2. Hydrostatic Force

The lateral hydrostatic force is determined by the following equation (Ref. 3.4-2):

$$F_{Hyd} = \frac{1}{2} \gamma_w h^2$$

where:

F_{Hyd} = hydrostatic force, lb/ft of width

h = water depth, ft

γ_w = unit weight of water, lb/ft^3

3.4.2.2.1.2 Lateral Loading on the ECWIS and the South ECP Embankment – The determination of the maximum lateral force on the ECWIS considered both instantaneous and quasi-steady-state conditions. The maximum total force on the ECWIS is a result of the MCR embankment breach discussed in Section 2.4.4.2.2. This force is the result of a water elevation of 41.0 ft mean sea level during the quasi-steady state condition.

Since the south ECP embankment crest elevation is 34.0 ft MSL, it would be overtopped by the flood wave resulting from the MCR embankment breach. The south ECP embankment is designed to withstand the lateral force based on the maximum water elevation resulting from MCR embankment breach.

3.4.2.2.2 Vertical Loading: The roofs of seismic Category I structures are designed to withstand the weight of the accumulated PMP, assuming completely clogged drains (Section 2.4.2.3).

Table 3.4-1 shows the elevations of maximum water surface used for buoyancy calculations. The maximum buoyant force is calculated by assuming that the granular backfill around the structures is completely saturated so that the buoyant force will occur as soon as water arrives at the plant area.

~~3.4.3 Internal Flood Protection~~

3.4.3.1 Protection Features. Safety-related systems, components and structures are protected such that the plant can achieve and maintain a safe shutdown condition and prevent unacceptable radiological releases to the environment.

In general, the plant layout arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, there is minimal effect on other systems or components which are required for safe shutdown of the plant or to mitigate the consequence of internal flooding.

Where separation is not feasible, other protection features are employed. These protection features include the following:

CS1

ENERCON Excellence—Every project. Every day	CALCULATION COVER SHEET	CALC. NO. STPNOC013-CALC-006 REV. 1 PAGE NO. 1 of 42
Title:	Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds	**Client:** STP **Project:** STPNOC013

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions that require confirmation? (If **YES**, Identify the assumptions) ______	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, Identify the design verified calculation.) **Design Verified Calculation No.**______	☐	☒
3	Does this calculation Supersede an existing Calculation? (If **YES**, identify the superseded calculation.) **Superseded Calculation No.**______	☐	☒

Scope of Revision:

Added reference for reactor vessel head thickness, and updated calculations with new value (7.19 in). Removed any detector specific calculations so results can be applied to any detector at these locations. Made several editorial changes.

Revision Impact on Results:

The dose rates for the cases with reactor vessel head attached are higher due to the reduction in head thickness.

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☒ **Non-Safety Related** ☐

(Print Name and Sign)

Originator: Andrew Blackwell	Andrew Blackwell	**Date:** 3/21/14
Design Verifier: Curt Lindner		**Date:** 3/21/14
Approver: Marvin Morris	Marvin Morris	**Date:**

ENERCON Excellence—Every project. Every day	CALCULATION REVISION STATUS SHEET	CALC. NO. STPNOC13-CALC-006
		REV. 1
		PAGE NO. 2 of 42

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	02/07/2014	
1	03/21/2014	Updated containment dimensions including reactor vessel head thickness. Added more detail to calculations section. Made editorial changes.

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	0		
5-16,18,19,21-26,28,30,31,34-42	1		

APPENDIX REVISION STATUS

APPENDIX NO.	PAGE NO.	REVISION NO.	APPENDIX NO.	PAGE NO.	REVISION NO.
A	All	0			
B	All	1			

Calculation Design Verification Plan:

The calculation will be reviewed for correctness of inputs, design criteria, analyzed methods, and acceptance criteria.

The stated objectives and conclusions will be confirmed to be reasonable and valid.

Assumptions will be reviewed and confirmed to be appropriate and verified to be valid based on sound engineering principles and practices.

(Print Name and Sign for Approval – mark "N/A" if not required)

Approver: Marvin Morris Marvin Morris	Date:

Calculation Design Verification Summary:

The calculation has been designated as **Safety Related** as noted in the cover sheet.

The calculation has been verified to be correct and performed using appropriate design inputs, assumptions, analytical methods, design criteria, and acceptance criteria.

The conclusions developed in the calculation are reasonable, valid, and consistent with the purpose and scope.

The assumptions are appropriate and valid.

Based On The Above Summary, The Calculation Is Determined To Be Acceptable.

(Print Name and Sign)

Design Verifier: Curt Lindner	Date: 3/21/14
Others:	Date:

Item	CHECKLIST ITEMS	Yes	No	N/A
1	**Design Inputs** - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis, and incorporated in the calculation?	X		
2	**Assumptions** - Were the assumptions reasonable and adequately described, justified and/or verified, and documented?	X		
3	**Quality Assurance** - Were the appropriate QA classification and requirements assigned to the calculation?	X		
4	**Codes, Standards, and Regulatory Requirements** - Were the applicable codes, standards, and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied?	X		
5	**Construction and Operating Experience** - Have applicable construction and operating experience been considered?		X	
6	**Interfaces** - Have the design-interface requirements been satisfied, including interactions with other calculations?	X		
7	**Methods** - Was the calculation methodology appropriate and properly applied to satisfy the calculation objective?	X		
8	**Design Outputs** - Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives, and are the results reasonable compared to the inputs?	X		
9	**Radiation Exposure** - Has the calculation properly considered radiation exposure to the public and plant personnel?		X	
10	**Acceptance Criteria** - Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	X		
11	**Computer Software** - Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?	X		

COMMENTS:

In accordance with CSP 3.02, MCNP5 and SCALE6.0 have been verified for use on ENERCON computers.

	(Print Name and Sign)	
Design Verifier:	Curt Lindner [signature]	**Date:** 3/21/14
Others:		**Date:**

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1. Purpose and Scope

The purpose of this calculation is to evaluate dose rates as a function of water height in the reactor vessel during refueling operations in order to set Emergency Action Level (EAL) thresholds for core uncovery. The dose rates are calculated at the locations of the containment monitors RE-8055 and RE-8099 so that dose rate measurements by these devices can be used to estimate water level in the core, upon failure of other water level detection systems. This evaluation will calculate the dose rate at full core uncovery, as well as maximum water levels with a detectable dose rate response. Since the scope of this calculation concerns uncovering the reactor core, the effects of future fuel element storage in the nearby Fuel Storage Pit are not analyzed, since it's effects are negligible in comparison. The containment building, components within the building, and the reactor vessel and contents are modeled simplistically because only order of magnitude results are needed. As such, the dose rate results should be considered as reasonably representative of the magnitude of the actual dose rate only.

2. Summary of Results and Conclusion

The dose rate results for the configuration without the reactor vessel head and with the reactor vessel head are provided in Section 7.7.1 and Section 7.7.2, respectively. The dose rate with the core uncovered (i.e. water at the top of the active length) is 2.23E+04 mrem/h with the head in place and 9.30E+06 mrem/h with the head removed. Detailed results of the dose rate as a function of water height are provided in Figure 7-13 with the head removed and Figure 7-14 with the head attached.

3. References

1. "Standard Composition Library," ORNL/NUREG/CSD-2/V1/R6, Volume 3, Section M8, March 2000.
2. Calculation NC-6510. "Core Radionuclide Inventory for Chapter 15 Accident Analysis."
3. RSICC Code Package CCC-750, "SCALE 6.0: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Jan. 2009.
4. "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms", I.C. Gauld, O.W. Hermann, & R. M. Westfall. Jan. 2009.
5. STP001-CPC-001. Computer Program Certification MCNP5 Version 1.4 and SCALE 6.0.
6. ENERCON email from Paul Sudnak, dated December 9, 2013. (Appendix A).
7. Drawing 6C-18-N-5006, Rev. 9. "General Arrangement Reactor Containment Building Plan at El. 68' 0" Area G."
8. Drawing 6C-18-9-N-5007, Rev. 6. "General Arrangement Reactor Containment Building Section A-A Area G."
9. Drawing 6C-18-9-N-5008, Rev. 8. "General Arrangement Reactor Containment Building Section B-B Area G."
10. RSICC Code Package CCC-730, "MCNP/MCNPX Monte Carlo N-Particle Transport Code System 12 Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," January 2006.
11. ANSI/ANS 6.1.1-1977, Neutron and Gamma Flux-To-Dose Conversion Factors.
12. ENERCON email from Paul Sudnak, dated February 3, 2014 (Appendix A).
13. Drawing L5-01EM101, Rev. 1. "Closure Head General Assembly."
14. Drawing 1142E24. "Model 4XLR Reactor 173 in. I.D. Vessel."
15. Drawing 2C26-9-S-1004, Rev. 4. "Steel Reactor Containment Building Cylindrical Shell Liner Sects. And Dets. Unit N° 1 & 2."
16. Drawing 1211E6. "4 Loop Rapid XL Reactor General Assembly."

4. Assumptions

The following assumptions are used in the core uncovery dose rate calculation:

1. The core is homogenized based on the typical Vantage 5 fuel assembly dimensions, taking into account the fuel rods and space between. Any small variations in fuel parameters will have a negligible effect on containment dose rates.
2. Any non-fuel hardware is ignored since the primary self-shielding occurs in the fuel itself, and there may be some unknown streaming effects through the non-fuel hardware. This homogenization takes into account the water level when calculating the isotopic weight fraction and homogenized density.
3. The source term for this evaluation is based on the fission product inventory at the time of shutdown. Because there is no cooling time, the fuel gamma source term will predominate and the N-gamma and hardware activation can be neglected.
4. The compositions of the containment structure and components are based on the values in the SCALE standard composition library [1].
5. The RE-8055 and RE-8099 monitors are assumed to be 5 feet above the 68 foot level in order to take into account the mounting device.
6. The containment outer concrete thickness is modeled as 3 feet thick. Because the backscattering off the containment walls is due to the steel liner, this dimension has a negligible impact on dose rates near the reactor vessel.

5. Design Inputs

5.1 Fuel Assembly Parameters

The following fuel assembly parameters are used in the core homogenization in the MCNP model. They are based on typical fuel assembly values for Westinghouse Vantage 5 fuel.

Table 5-1 Design Input Fuel Assembly Parameters for Westinghouse Vantage 5 Fuel

Parameters	Value	Unit	Reference
Fuel Type	Westinghouse Vantage +		Assumption 1
# Fuel Rods per Assy	264		Assumption 1
Assembly Array	17x17		Assumption 1
Enrichment	4	wt %	Assumption 1
Density (% of theoretical)	0.95		Assumption 1
Fuel Pellet OD	0.3225	[in]	Assumption 1
Fuel Rod Pitch	0.496	[in]	Assumption 1

Parameters	Value	Unit	Reference
Fuel Rod OD	0.374	[in]	Assumption 1
Clad Thickness	0.0225	[in]	Assumption 1
Guide Tube OD	0.482	[in]	Assumption 1
Guide Tube Thickness	0.020	[in]	Assumption 1
# Guide Tubes	24		Assumption 1
Instrument Tube OD	0.482	[in]	Assumption 1
Instrument Tube Thickness	0.020	[in]	Assumption 1
# Instrument Tubes	1		Assumption 1
Active Length	14	[ft]	Assumption 1

5.2 Containment Dimensions

The following dimensions are based on drawings of the STP containment building and equipment. Some parameters are estimated using scaling when the drawings do not detail the exact dimension. These estimations are only applied to dimensions that have a negligible effect on the core uncovery dose rate analysis.

Table 5-2 Design Input Containment Dimensions

Dimension:	ft.	in	cm	reference
Reactor Pressure Vessel				
Elevation at top of active fuel	28	2	858.52	[6]
Elevation at head level platform	38	6.5	1174.75	[8]
Elevation at full water level in refueling cavity	66	6	2026.92	[8]
Closure head thickness	0	7.19	18.2626	[13]
Reactor pressure vessel inside diameter at shell	0	173	439.42	[14]
Height of reactor vessel from bottom of fuel to head level			742.95	Calculated
Steam Generator				
Elevation at bottom of SG	38	4	1168.4	[9]
Elevation at top of SG	105	9.875	3225.4825	[9]
Total SG height			2057.0825	Calculated
SG outer diameter			500	[7] Scaled

Dimension:	ft.	in	cm	reference
Active Fuel				
Active fuel bottom elevation	12	1	368.3	[9]
Active fuel height	14	0	426.72	[14]
Concrete Wall				
Lower Height	38	6.5	1174.75	[9]
Upper Height	85	0	2590.8	[9]
Overall Height			1416.05	Calculated
Thickness	2	0	106	[7] Scaled
Width			874.776	[7] Scaled
Length			2499.36	[7] Scaled
Steam Generators				
Lower Modeled Height	85	0	2590.8	[9]
Upper Modeled Height	105	9.875	3225.4825	[9]
Overall Modeled Height			634.6825	Calculated
Diameter			500	[7] Scaled
Containment				
Upper modeled height	153	0	4663.44	[8]
Lower modeled height	68	0	2072.64	[8]
Net Height			2590.8	Calculated
Inner Diameter	149	$11^1/_4$	4570	[15]
Liner Thickness	0	0.375	0.9525	[15]
Dome Inner Radius	74	$11^5/_8$	2285	[15]
Concrete Thickness	3	0	91.44	Assumption 6

5.3 Core Isotopic Inventory

Core isotopic activities are provided in Table 11 of [2]. The isotope specific activities are given in terms of Ci/MWt, which is converted to curies based on the total core thermal power of 4,100 MWt [2]. These calculations are performed in EXCEL spreadsheet *STP.xlsx*. A table of the input values is shown in Table 5-3, below.

Table 5-3 Design Basis Core Shutdown Source Term[1]

Isotope	Ci/MWt	Ci	Isotope	Ci/MWt	Ci
Kr83m	3.41E+03	1.40E+07	Ru106	1.34E+04	5.49E+07
Kr85m	7.07E+03	2.90E+07	Rh105	3.05E+04	1.25E+08
Kr85	2.93E+02	1.20E+06	Zr95	4.39E+04	1.80E+08
Kr87	1.34E+04	5.49E+07	Zr97	4.39E+04	1.80E+08
Kr88	1.90E+04	7.79E+07	Nb95	4.32E+04	1.77E+08
Kr89	2.32E+04	9.51E+07	La140	4.63E+04	1.90E+08
Xe131m	2.68E+02	1.10E+06	La141	4.62E+04	1.89E+08
Xe133m	1.66E+03	6.81E+06	La142	4.15E+04	1.70E+08
Xe133	5.37E+04	2.20E+08	Pr143	3.90E+04	1.60E+08
Xe135m	1.02E+04	4.18E+07	Nd147	1.73E+04	7.09E+07
Xe135	1.34E+04	5.49E+07	Am241	2.75E+00	1.13E+04
Xe137	4.63E+04	1.90E+08	Cm242	1.05E+03	4.31E+06
Xe138	4.39E+04	1.80E+08	Cm244	6.17E+01	2.53E+05
I131	2.59E+04	1.06E+08	Ce141	4.39E+04	1.80E+08
I132	3.71E+04	1.52E+08	Ce143	4.15E+04	1.70E+08
I133	5.37E+04	2.20E+08	Ce144	3.41E+04	1.40E+08
I134	5.85E+04	2.40E+08	Np239	5.12E+05	2.10E+09
I135	4.88E+04	2.00E+08	Pu238	8.71E+01	3.57E+05
Sb127	3.05E+03	1.25E+07	Pu239	1.96E+01	8.04E+04
Sb129	8.29E+03	3.40E+07	Pu240	2.48E+01	1.02E+05
Te127m	4.32E+02	1.77E+06	Pu241	4.17E+03	1.71E+07
Te127	3.05E+03	1.25E+07	Rb86	9.92E+01	4.07E+05
Te129m	1.22E+03	5.00E+06	Cs134	5.37E+03	2.20E+07
Te129	8.05E+03	3.30E+07	Cs136	1.54E+03	6.31E+06
Te131m	3.66E+03	1.50E+07	Cs137	3.17E+03	1.30E+07
Te132	3.82E+04	1.57E+08	Y90	3.56E+03	1.46E+07
Ba137m	2.93E+03	1.20E+07	Y91	3.41E+04	1.40E+08
Ba139	4.98E+04	2.04E+08	Y92	3.41E+04	1.40E+08
Ba140	4.63E+04	1.90E+08	Y93	3.90E+04	1.60E+08
Mo99	4.83E+04	1.98E+08	Sr89	2.68E+04	1.10E+08
Tc99m	4.07E+04	1.67E+08	Sr90	2.37E+03	9.72E+06
Ru103	3.90E+04	1.60E+08	Sr91	3.17E+04	1.30E+08
Ru105	2.68E+04	1.10E+08	Sr92	3.41E+04	1.40E+08

[1] Ci = Ci/MWt × 4,100 MWt

5.4 Material Compositions

The following compositions used in the MCNP model are taken from the SCALE standard composition library [1] and are shown in Table 5-4.

Table 5-4 SCALE Standard Compositions used in MCNP Model

Material	Isotope	Weight Fraction	Reference
Zry- 4	Zr	0.9823	[1]
(6.56 g/cm^3)	Sn	0.0145	
	Cr	0.0010	
	Fe	0.0021	
	Hf	0.0001	
UO_2	U-235	0.0353	[1]
(10.412 g/cm^3)[2]	U-238	0.8461	
	O	0.1186	
Air	C	0.0001	[1]
(1.21E-03 g/cm^3)	N	0.7651	
	O	0.2348	
Water	H	0.1111	[1]
(0.9982 g/cm^3)	O	0.8889	
SS-304	Fe	0.6838	[1]
(7.94 g/cm^3)	Cr	0.1900	
	Ni	0.0950	
	Mn	0.0200	
	Si	0.0100	
	C	0.0008	
	P	0.0004	
Concrete	O	0.5320	[1]
(2.30 g/cm^3)	Si	0.3370	
	Ca	0.0440	
	Al	0.0340	
	Na	0.0290	
	Fe	0.0140	
	H	0.0100	
Carbon Steel	C	0.0100	[1]
(7.82 g/cm^3)	Fe	0.9900	

[2] Based on 95% of theoretical density, Assumption 1.

6. Methodology

The reactor source terms are computed with ORIGEN-S of the SCALE 6.0 code package [3, 4]. The ORIGEN-S decay sequence is used to bin design input isotope specific activities into energy dependent photon bins. These energy specific photon emission bins are used as input for the energy distribution described by the MCNP source definitions.

The ORIGEN-S sequence in the SCALE6.0 program package is verified for use in safety related calculations [5]. The program certification form is maintained in the project file.

MCNP5, release 1.40 [10], Monte Carlo transport is used to determine the dose rates. The ENDF/B-VI neutron cross section library, ENDF60, and the ENDF/B-VI Release 8 Photo-atomic Data gamma cross section library, MCPLIB04 are utilized in the transport computations. This software has been verified for use in safety related calculations [5].

The detailed engineering drawings are converted into MCNP surface and cell cards in the proper dimensions. The radiation monitors of interest are modeled as point detectors to determine the expected dose rate for those detectors. The dose rates are calculated as a function of water height for two reactor refueling conditions:

1. With Head – the reactor is modeled with an 7.19 inch carbon steel plate as indicated in Table 5-2, which is additional attenuation between source and detector.

2. Without head – the reactor is modeled with nothing between the active fuel zone and containment.

For low water levels, variance reduction is accomplished with a geometric importance map that is imposed on the homogenized core. Without significant amounts of water present, this is enough to calculate statistically sound dose rate results. Once the water depth reaches a height where the variance of the results reaches an unacceptable level, a superimposed weight windows mesh is utilized to improve the variance reduction of the simple geometric scheme. The weight windows are iteratively generated using the MCNP weight windows generator card with a mesh over the existing geometry. All final dose rates presented in this calculation include weight windows variance reduction.

7. Calculations

7.1 Source Terms

In order to convert the isotope specific activity into an energy spectrum, ORIGEN-S of the SCALE6.0 code package is used to initiate a decay and bin into 19 photon energy groups. The energy groups along with their associated activities are used in the MCNP source definition to model the anticipated radiation emission following shutdown.

The ORIGEN-S input deck, *STPEAL.inp*, is provided below in Figure 7-1. This input has a simple decay case where the inputted isotopic composition in curies is decayed. The isotope is specified in the 73$$ card using the special identifier described in Section F7.6.2 of the ORIGEN-S manual, and the activity in curies is specified in the 74** card. The time steps for the decay are given on the 60** card in years. Although multiple time steps are calculated, the source term with zero decay time is used in this calculation to model the core immediately after shutdown. The output of the decay is given in terms of photons/s/Energy-Group, which is automatically normalized in the MCNP input.

Figure 7-1 ORIGEN-S Input Deck for MCNP Source Term Calculation

```
=origens                                                           ←Call Origen-S Sequence
0$$ a11 71 e t                                                     ←Logical Unit Assignments
                                                                          -Binary Photon Library (71)
PWR Source Term STP ELA Analysis                                   ←Case Title
3$$  21 1 1 a4 27 a16 4 a33 19 e t                                 ←Library Integer Constants
                                                                          -Units 83** Card Ci (4)
                                                                          -Gamma Energy Groups (19)
35$$ 0 t                                                           ←Not Used
54$$ a8 0 a11 2 e                                                  ←Special Calculation Options
                                                                          -Cutoff Value (Default)
                                                                          -(α,n) Composition Dependent
56$$ 0 6 a6 1 a10 0 a13 66 5 3 0 2 0 e                             ←Subcase Control Constants
                                                                          -Decay Only Subcase (0)
                                                                          -Number of Time Intervals (6)
                                                                          -Number of Nuclides (66)
                                                                          -Unit of Time in Years (5)
57** 0  a3 1-16 e                                                  ←Not Used
95$$ 0 t                                                           ←Not Used
STPEAL                                                             ←Subcase Title
Ci Source Terms                                                    ←Subcase Basis
60** 0 0.1 0.2 0.3 0.4 0.5                                         ←Time (years)
61** 5r1-8 1+6 1+4                                                 ←Cutoff Values
65$$                                                               ←Decay Period Print Triggers
'GRAM-ATOMS   GRAMS     CURIES     WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0     1 0 0      1 0 0         3Z            6Z
     3Z      1  1  1     1 0 1      1 1 1         3Z            6Z
     3Z      1  1  1     1 1 1      1 1 1         3Z            6Z
81$$ 2 0 26 1 e                                                    ←Gamma Source Constants
82$$ f2                                                            ←Produces Gamma Source Spectrum
83** 1.10E+07 1.00E+07 8.00E+06 6.50E+06 5.00E+06 4.00E+06 3.00E+06 ←Gamma Energy Groups
     2.50E+06 2.00E+06 1.66E+06 1.33E+06 1.00E+06 8.00E+05 6.00E+05
     4.00E+05 3.00E+05 2.00E+05 1.00E+05 5.00E+04 1.00E+04 e
84** 2.00E+07 6.43E+06 3.00E+06 1.85E+06 1.40E+06 9.00E+05 4.00E+05 ←Neutron Energy Groups
     1.00E+05 1.70E+04 3.00E+03 5.50E+02 1.00E+02 3.00E+01 1.00E+01   (Not Used)
     3.05E+00 1.77E+00 1.30E+00 1.13E+00 1.00E+00 8.00E-01 4.00E-01
     3.25E-01 2.25E-01 1.00E-01 5.00E-02 3.00E-02 1.00E-02 1.00E-05 e
73$$ 360831 360851 360850 360870 360880 360890 541311 541331 ←Nuclide Identifiers
    541330 541351 541350 541370 541380 531310 531320 531330
    531340 531350 511270 511290 521271 521270 521291 521290
    521311 521320 561371 561390 561400 420990 430991 441030
    441050 441060 451050 400950 450970 410950 571400 571410
    571420 591430 601470 952410 962420 962440 581410 581430
    581440 932390 942380 942390 942400 942410 370860 551340
    551360 551370 390900 390910 390920 390930 380890 380900
    380910 380920
74** 1.40E+07 2.90E+07 1.20E+06 5.49E+07 7.79E+07 9.51E+07 1.10E+06 ←Nuclide Concentrations (Ci)
    6.81E+06 2.20E+08 4.18E+07 5.49E+07 1.90E+08 1.80E+08 1.06E+08
    1.52E+08 2.20E+08 2.40E+08 2.00E+08 1.25E+07 3.40E+07 1.77E+06
    1.25E+07 5.00E+06 3.30E+07 1.50E+07 1.57E+08 1.20E+07 2.04E+08
    1.90E+08 1.98E+08 1.67E+08 1.60E+08 1.10E+08 5.49E+07 1.25E+08
    1.80E+08 1.80E+08 1.77E+08 1.90E+08 1.89E+08 1.70E+08 1.60E+08
    7.09E+07 1.13E+04 4.31E+06 2.53E+05 1.80E+08 1.70E+08 1.40E+08
    2.10E+09 3.57E+05 8.04E+04 1.02E+05 1.71E+07 4.07E+05 2.20E+07
    6.31E+06 1.30E+07 1.46E+07 1.40E+08 1.40E+08 1.60E+08 1.10E+08
    9.72E+06 1.30E+08 1.40E+08
75$$ 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 ←Library Kind
                                                              2-Actinide 3-Fission Product
    3 3 3 3 3 3 3 3 3 3 3 2 2 2 3 3 3 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3
    3 3
t
56$$ f0 t
End
```

The results of this calculation are summarized below in Table 7-1. These values will be used in the MCNP input source definition.

Table 7-1 Binned Total Core Source Term

Energy Group	Energy Boundaries (MeV)	Photons/sec
1	0.01-0.05	9.29E+19
2	0.05-0.1	2.93E+19
3	0.1-0.2	6.54E+19
4	0.2-0.3	4.28E+19
5	0.3-0.4	1.52E+19
6	0.4-0.6	3.58E+19
7	0.6-0.8	4.35E+19
8	0.8-1	2.66E+19
9	1-1.33	1.29E+19
10	1.33-1.66	1.65E+19
11	1.66-2	5.57E+18
12	2-2.5	5.53E+18
13	2.5-3	1.98E+18
14	3-4	7.81E+17
15	4-5	3.48E+16
16	5-6.5	3.95E+11
17	6.5-8	1.75E+08
18	8-10	3.71E+07
19	10-11	2.01E+06
totals		3.95E+20

7.2 MCNP Model Core Homogenization

Because the source term is given for the entire core, the self-shielding from the assemblies is an important part of the dose rate response in regions above the core. Particles born in the lower section of the core are very unlikely to penetrate through the core itself, and make it to the radiation monitors. For simplicity, the core is modeled as a 3 dimensional cylinder with a uniformly distributed spatial particle distribution. The calculations for the homogenization are done in the worksheet *Compositions* of the EXCEL workbook *STP.xlsx*. A density and isotopic composition is calculated with the water level above the top of the fuel. A summary of the calculations for the core composition and density is shown below. The inputs are based on the dimensions in Table 5-1 and the compositions in Table 5-4.

$$Rod\ Volume = \pi(Pellet\ Radius)^2 \times Active\ Length = (3.14)(0.16125\ in)^2(168\ in) = 13.7\ in^3$$

$$Rod\ Mass_{UO_2} = \rho \times V = \left(10.96\frac{g}{cc}\right)(0.95)(13.72\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 2341.5\ g$$

$$Assembly\ Mass_{UO_2} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (2341.5\ g)(264) = 618.2\ kg$$

$$Clad\ Volume = \pi\left(\frac{OD^2}{4} - \frac{ID^2}{4}\right) \times Active\ Length = (3.14)\left[\frac{(0.374\ in)^2}{4} - \frac{(0.329\ in)^2}{4}\right](168\ in)$$
$$= 4.17\ in^3$$

$$Rod\ Mass_{Zry-4} = \rho \times V = \left(6.56\ \frac{g}{cc}\right)(4.17\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 448.7\ g$$

$$Assembly\ Mass_{Zry-4} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (448.7\ g)(264) = 118.5\ kg$$

$$Assembly\ \ H_2O\ Volume = [(Assembly\ Width)^2 - \pi(Rod\ Radius)^2 \times 264] \times Active\ Length$$
$$= [(8.404\ in)^2 - (3.14)(0.187\ in)^2(264)](168\ in) = 6993\ in^3$$

$$Assembly\ Mass_{H_2O} = \rho \times V = \left(0.9982\frac{g}{cc}\right)(6993\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 114.4\ kg$$

$$Assembly\ Volume = Active\ Length\ \times (Assembly\ Width)^2 = (168\ in)(8.404\ in)^2 = 11865.4\ in^3$$

$$Density = \frac{Total\ Mass}{Volume} = \frac{1000(618.2 + 118.5\ kg + 114.4)\ kg}{11865.4\ in^3\left(2.54\frac{cm}{in}\right)^3} = 4.38\ g/cc$$

7.3 MCNP Model Geometry

The following MCNP model geometry is based on the containment dimensions summarized in Table 5-2. The model only focuses on the primary systems and components that provide shielding or reflection from the core to the radiation monitors. These components include the reactor vessel, concrete in reactor pit, containment walls (reflection), and steam generators (reflection). VISED plots of the model geometry are provided in Figure 7-2, Figure 7-3, and Figure 7-4. The MCNP surface cards with the model dimensions (cm) are shown in Figure 7-5, and the cell cards are shown in Figure 7-6 for the cases with no reactor head. A VISED plot of the model with the reactor head is shown in Figure 7-7. The surface and cell cards for the cases with the reactor head are shown in and Figure 7-8, respectively. Areas that are not of interest are given an importance of zero (white areas) so MCNP will not track particles in locations that will not contribute to the detector response. A summary of surfaces used in constructing this geometry is shown in Table 7-2, including a description of macrobody dimensions.

Figure 7-2 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (No Head)

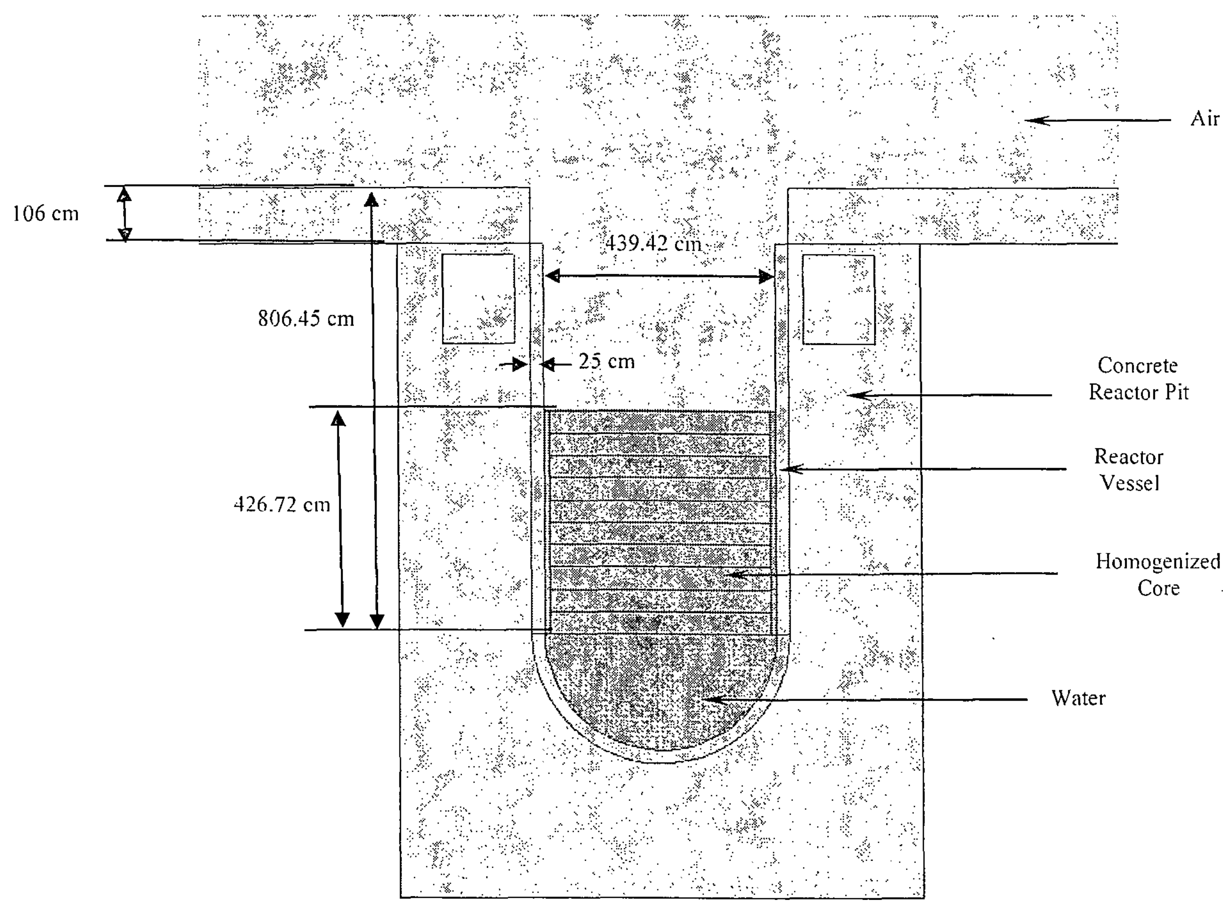
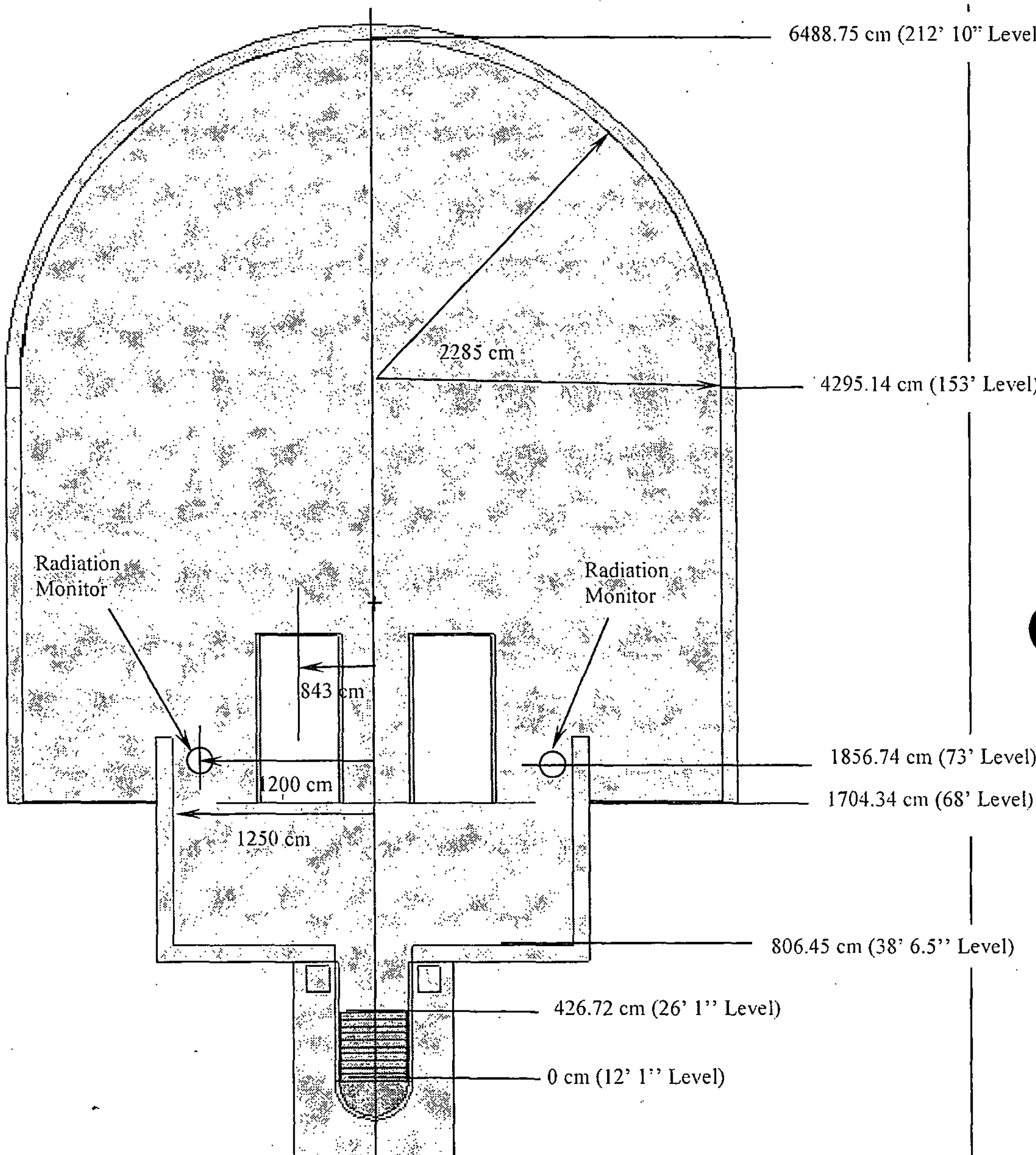


Figure 7-3 X-Z VISED Plot of Containment[3]



[3] Steam Generators are not full height. Also, they are not on the same X-Z plane as the core shown above. They are included for visualization purposes.

Figure 7-4 X-Y VISED Plot of the Containment Geometry at Radiation Monitor Level

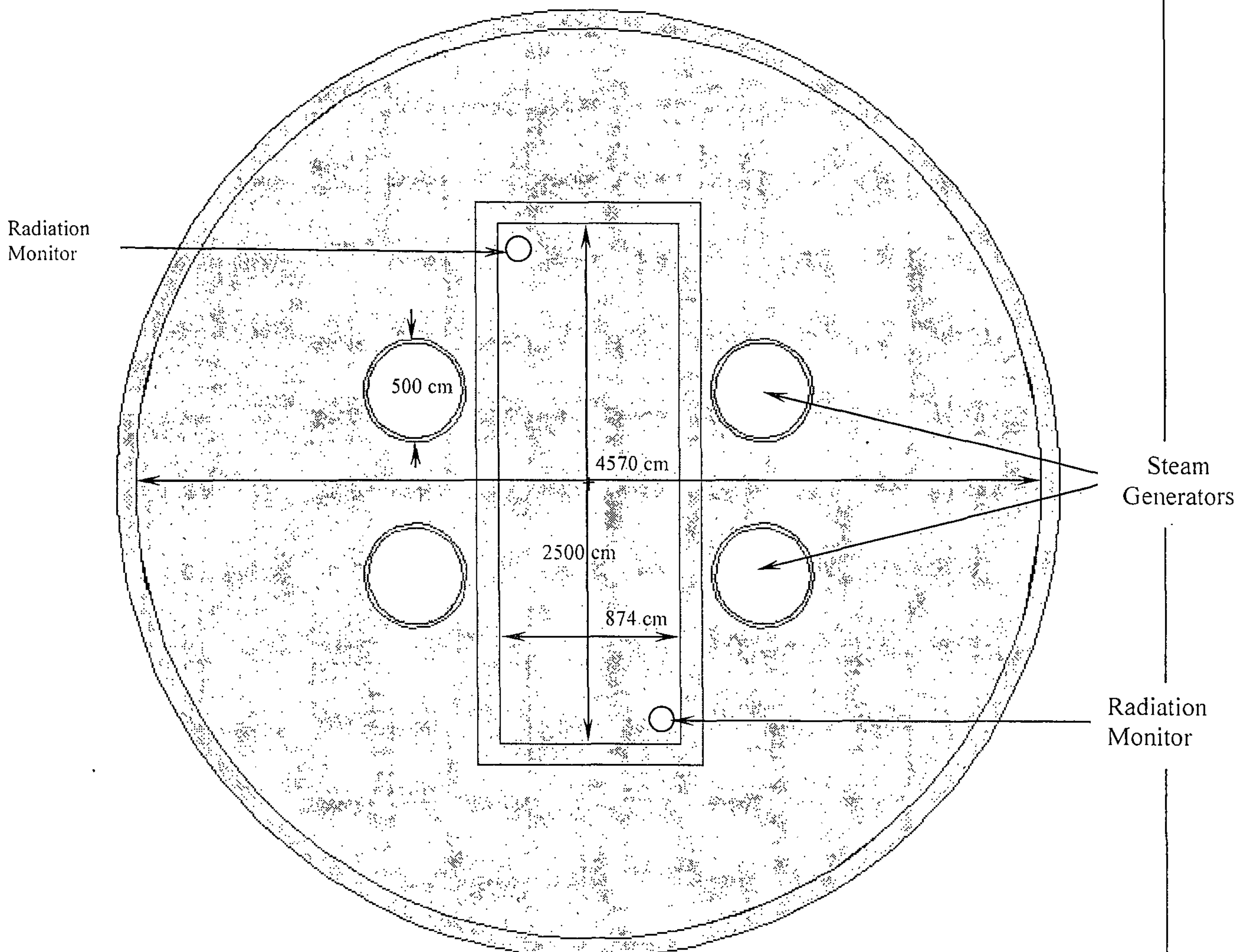


Table 7-2 Summary of Surfaces Used for MCNP Models

Surface Type	Surface Number	Dimensions							Description
RCC		X_0	Y_0	Z_0	V_X	V_Y	V_Z	R	
	1	0	0	0	0	0	426.72	209.71	Active Fuel Region
	2	0	0	0	0	0	700.45	219.71	Reactor Pressure Vessel Inner Surface
	3	0	0	0	0	0	700.45	244.71	Reactor Pressure Vessel Outer Surface
	31	0	0	700.45	0	0	18.26	244.71	Reactor Pressure Vessel Head
	41	0	0	512.81	0	0	167.64	274.71	Concrete Void for Primary Loop
	42	0	0	512.81	0	0	167.64	411.71	Concrete Void for Primary Loop
	10	0	0	700.45	0	0	106	244.71	Concrete Wall Cutout
	11	444.71	843	700.45	0	0	2050	250	Steam Generator 1
	12	444.71	843	720.45	0	0	2010	230	Steam Generator Inner 1
	13	-444.71	843	700.45	0	0	2050	250	Steam Generator 2
	14	-444.71	843	720.45	0	0	2010	230	Steam Generator Inner 2
	15	-444.71	-843	700.45	0	0	2050	250	Steam Generator 3
	16	-444.71	-843	720.45	0	0	2010	230	Steam Generator Inner 3
	17	444.71	-843	700.45	0	0	2050	250	Steam Generator 4
	18	444.71	-843	720.45	0	0	2010	230	Steam Generator Inner 4
	21	0	0	1694.34	0	0	2600.8	2285	Containment Inner Liner Surface
	22	0	0	1694.34	0	0	2600.8	2285.95	Containment Inner Concrete Surface
	23	0	0	1694.34	0	0	2600.8	2377.39	Containment Outer Concrete Surface
RPP		-X	X	-Y	Y	-Z	Z		
	4	-498	498	-498	498	-498	700.45		Concrete Surrounding RPV
	8	-1250	1250	-437	437	806.45	2116.45		Concrete Wall Fuel Pit Inner
	9	-1356	1356	-543	543	700.45	2116.45		Concrete Wall Fuel Pit Outer
SPH		X_0	Y_0	Z_0	R				

Surface Type	Surface Number	Dimensions							Description
	5	0	0	0	219.71				Bottom of Reactor Pressure Vessel Inner
	6	0	0	0	244.71				Bottom of Reactor Pressure Vessel Outer
	24	0	0	4295.14	2285				Containment Dome Inner Liner Surface
	25	0	0	4295.14	2285.95				Containment Dome Inner Concrete Surface
	26	0	0	4295.14	2377.39				Containment Dome Outer Concrete Surface
PZ		Z							
	7	0							Fuel Bottom
	71	700.45							Top of RPV
	20	Variable							Water Level
	27	4295.14							Spring Line
	28	1704.34							68' Level
	101-110	42.672	-	426.72					Geometric Importance Divisions in Active Zone

Figure 7-5 MCNP Model Surface Cards[4]

```
c surfaces
1 rcc 0 0 0 0 0 426.72 209.71                 $ Active Fuel Region
2 rcc 0 0 0 0 0 700.45 219.71                 $ Reactor Pressure Vessel Inner Surface
3 rcc 0 0 0 0 0 700.45 244.71                 $ Reactor Pressure Vessel Outer Surface
31 rcc 0 0 700.45 0 0 18.26 244.71            $ Reactor Vessel Head
4 rpp -498 498 -498 498 -498 700.45           $ Concrete Surrounding RPV
41 rcc 0 0 512.81 0 0 167.64 274.71           $ Concrete Void for Primary Loop
42 rcc 0 0 512.81 0 0 167.64 411.71           $ Concrete Void for Primary Loop
5 sph 0 0 0 219.71                            $ Bottom of Reactor Pressure Vessel
6 sph 0 0 0 244.71                            $ Bottom of Reactor Pressure Vessel
7 pz 0                                        $ Bottom of Active Zone
71 pz 700.45                                  $ Top of RPV
8 rpp -1250 1250 -437 437 806.45 2116.45      $ Concrete Walls Fuel Pit Inner
9 rpp -1356 1356 -543 543 700.45 2116.45      $ Concrete Wall Fuel Pit Outer
10 rcc 0 0 700.45 0 0 106 244.71              $ Concrete Wall Cutout
11 rcc 444.71 843 700.45 0 0 2050 250         $ Steam Generator 1
12 rcc 444.71 843 720.45 0 0 2010 230         $ Inner Steam Generator 1
13 rcc -444.71 843 700.45 0 0 2050 250        $ Steam Generator 2
14 rcc -444.71 843 720.45 0 0 2010 230        $ Inner Steam Generator 2
15 rcc -444.71 -843 700.45 0 0 2050 250       $ Steam Generator 3
16 rcc -444.71 -843 720.45 0 0 2010 230       $ Inner Steam Generator 3
17 rcc 444.71 -843 700.45 0 0 2050 250        $ Steam Generator 4
18 rcc 444.71 -843 720.45 0 0 2010 230        $ Inner Steam Generator 4
20 pz 365.76                                  $ Water Elevation Surface
21 rcc 0 0 1694.34 0 0 2600.8 2285            $ Containment Inner Liner Surface
22 rcc 0 0 1694.34 0 0 2600.8 2285.95         $ Containment Inner Concrete Surface
23 rcc 0 0 1694.34 0 0 2600.8 2377.39         $ Containment Outer Concrete Surface
24 sph 0 0 4295.14 2285                       $ Containment Dome Inner Liner Surface
25 sph 0 0 4295.14 2285.95                    $ Containment Dome Inner Concrete Surface
26 sph 0 0 4295.14 2377.39                    $ Containment Dome Outer Concrete Surface
27 pz 4295.14                                 $ Spring Line
28 pz 1704.34                                 $ 68' Level
101 pz 42.672                                 $ Geometric Importance Division Fuel Zone
102 pz 85.344                                 $ Geometric Importance Division Fuel Zone
103 pz 128.016                                $ Geometric Importance Division Fuel Zone
104 pz 170.688                                $ Geometric Importance Division Fuel Zone
105 pz 213.36                                 $ Geometric Importance Division Fuel Zone
106 pz 256.032                                $ Geometric Importance Division Fuel Zone
107 pz 298.704                                $ Geometric Importance Division Fuel Zone
108 pz 341.376                                $ Geometric Importance Division Fuel Zone
109 pz 384.048                                $ Geometric Importance Division Fuel Zone
110 pz 426.72                                 $ Geometric Importance Division Fuel Zone
```

[4] The surface cards for the MCNP model without the reactor vessel head does not have surface 31. The other surfaces are identical.

Figure 7-6 MCNP Model Cell Cards (No Head)

```
c cells
101 1 -4.57 -1 -101                  imp:p=1        $ Active Fuel Region
102 1 -4.57 -1 101 -102              imp:p=2        $ Active Fuel Region
103 1 -4.57 -1 102 -103              imp:p=3        $ Active Fuel Region
104 1 -4.57 -1 103 -104              imp:p=4        $ Active Fuel Region
105 1 -4.57 -1 104 -105              imp:p=8        $ Active Fuel Region
106 1 -4.57 -1 105 -106              imp:p=16       $ Active Fuel Region
107 1 -4.57 -1 106 -107              imp:p=32       $ Active Fuel Region
108 1 -4.57 -1 107 -108              imp:p=64       $ Active Fuel Region
109 1 -4.57 -1 108 -109              imp:p=128      $ Active Fuel Region
110 1 -4.57 -1 109 -110              imp:p=256      $ Active Fuel Region
2 2 -0.9982 1 -3 #4 -20              imp:p=256      $ Water Region
4 4 -7.94 2 -3 7 -71                 imp:p=256      $ RPV Shell
5 4 -7.94 5 -6 -7 #7                 imp:p=256      $ Bottom RPV Shell
6 2 -0.9982 -5 -7                    imp:p=256      $ Water Above Fuel
61 2 -0.9982 -20 71 (-10:-8)         imp:p=256      $ Water Above Vessel Head
71 3 -1.21E-03 -42 41                imp:p=256      $ Void for Primary Loop
7 5 -2.3 6 3 -4 #71                  imp:p=256      $ Concrete Surrounding RPV
8 5 -2.3 8 -9 10                     imp:p=256      $ Concrete above RPV
9 4 -7.94 -11 12 28                   imp:p=256      $ Steam Generator 1
10 0 -12 28                          imp:p=0        $ Inner Steam Generator 1
11 4 -7.94 -13 14 28                 imp:p=256      $ Steam Generator 2
12 0 -14 28                          imp:p=0        $ Inner Steam Generator 2
13 4 -7.94 -15 16 28                 imp:p=256      $ Steam Generator 3
14 0 -16 28                          imp:p=0        $ Inner Steam Generator 3
15 4 -7.94 -17 18 28                 imp:p=256      $ Steam Generator 4
16 0 -18 28                          imp:p=0        $ Inner Steam Generator 4
20 4 -7.94 21 -22                    imp:p=256      $ Containment Liner
21 5 -2.3 22 -23                     imp:p=256      $ Containment Wall
22 4 -7.94 24 -25 27                 imp:p=256      $ Containment Dome Liner
23 5 -2.3 25 -26 27                  imp:p=256      $ Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13
        15 17                        imp:p=256      $ 68 foot level
30 3 -1.21E-03 (-24:-21:-8:-10:-2)
      11 13 15 17 20 #8 #24 #2 1     imp:p=256      $ Air in Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10
    #11 #12 #13 #14 #15 #16 #20 #21
    #22 #23 #24 #30 #61              imp:p=0        $ Problem Boundary
```

Figure 7-7 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (With Head)

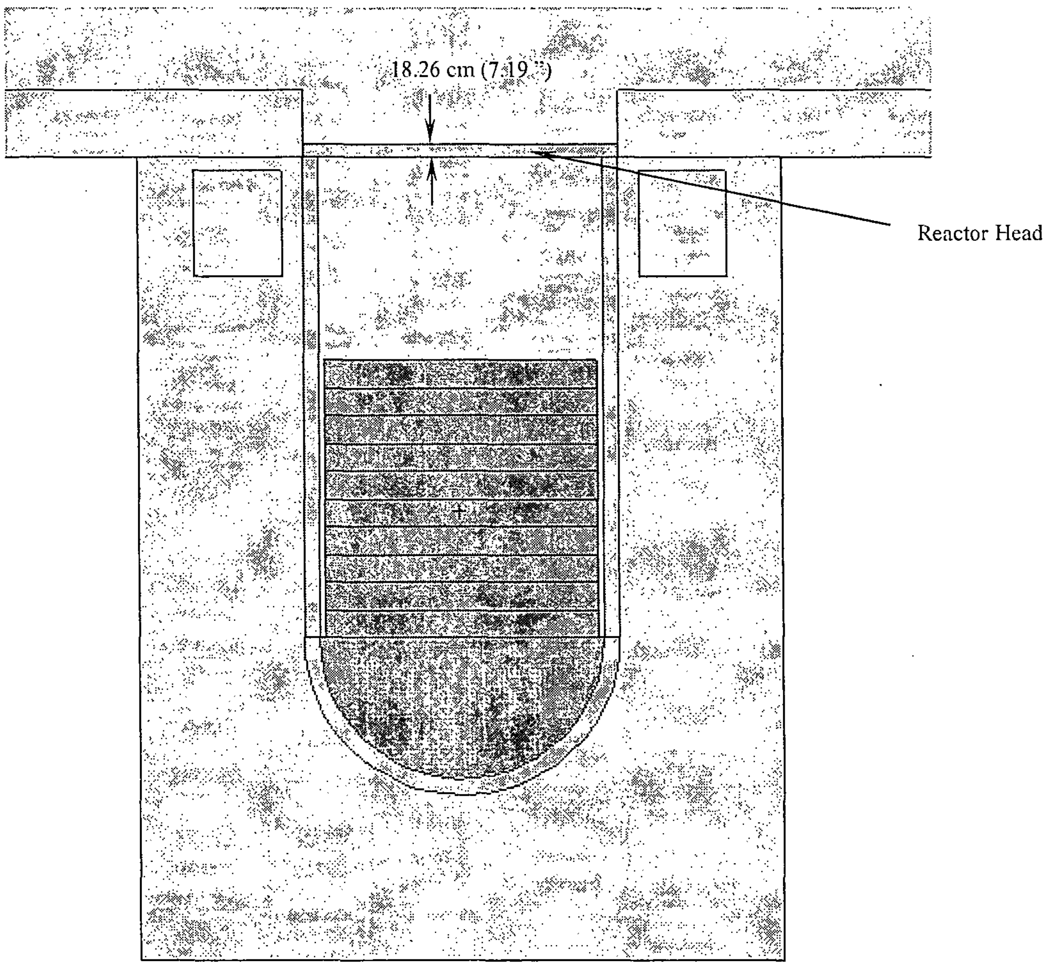


Figure 7-8 MCNP Cell Cards (With Head)

```
c cells
101 1 -4.57 -1 -101                    imp:p=1                        $ Active Fuel Region
102 1 -4.57 -1 101 -102                imp:p=2                        $ Active Fuel Region
103 1 -4.57 -1 102 -103                imp:p=3                        $ Active Fuel Region
104 1 -4.57 -1 103 -104                imp:p=4                        $ Active Fuel Region
105 1 -4.57 -1 104 -105                imp:p=8                        $ Active Fuel Region
106 1 -4.57 -1 105 -106                imp:p=16                       $ Active Fuel Region
107 1 -4.57 -1 106 -107                imp:p=32                       $ Active Fuel Region
108 1 -4.57 -1 107 -108                imp:p=64                       $ Active Fuel Region
109 1 -4.57 -1 108 -109                imp:p=128                      $ Active Fuel Region
110 1 -4.57 -1 109 -110                imp:p=256                      $ Active Fuel Region
2 2 -0.9982 1 -3 #4 -20 31             imp:p=256                      $ Water Region
4 4 -7.94 2 -3 7 -71                   imp:p=256                      $ RPV Shell
5 4 -7.94 5 -6 -7 #7                   imp:p=256                      $ Bottom RPV Shell
6 2 -0.9982 -5 -7                      imp:p=256                      $ Water Above Fuel
62 6 -7.8212 -31                       imp:p=256                      $ Reactor Vessel Head
61 2 -0.9982 -20 71 (-10:-8) 31        imp:p=256                      $ Water Above Vessel Head
71 3 -1.21E-03 -42 41                  imp:p=256                      $ Void for Primary Loop
7 5 -2.3 6 3 -4 #71                    imp:p=256                      $ Concrete Surrounding RPV
8 5 -2.3 8 -9 10                       imp:p=256                      $ Concrete above RPV
9 4 -7.94 -11 12 28                    imp:p=256                      $ Steam Generator 1
10 0 -12 28                            imp:p=0                        $ Inner Steam Generator 1
11 4 -7.94 -13 14 28                   imp:p=256                      $ Steam Generator 2
12 0 -14 28                            imp:p=0                        $ Inner Steam Generator 2
13 4 -7.94 -15 16 28                   imp:p=256                      $ Steam Generator 3
14 0 -16 28                            imp:p=0                        $ Inner Steam Generator 3
15 4 -7.94 -17 18 28                   imp:p=256                      $ Steam Generator 4
16 0 -18 28                            imp:p=0                        $ Inner Steam Generator 4
20 4 -7.94 21 -22                      imp:p=256                      $ Containment Liner
21 5 -2.3 22 -23                       imp:p=256                      $ Containment Wall
22 4 -7.94 24 -25 27                   imp:p=256                      $ Containment Dome Liner
23 5 -2.3 25 -26 27                    imp:p=256                      $ Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13
        15 17                          imp:p=256                      $ 68 foot level
30 3 -1.21E-03 (-24:-21:-8:-10:-2)
11 13 15 17 20 31 #8 #24 #2 1          imp:p=256                      $ Air inside Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10
     #11 #12 #13 #14 #15 #16 #20 #21
     #22 #23 #24 #30 #61 31            imp:p=0                        $ External to Problem
```

7.4 MCNP Source Definition

The core source term is assumed to be uniformly distributed throughout the volume, and has an energy spectra based on the core inventory [2]. Only the gamma source term is taken into account for this evaluation. Because the source term is generated immediately after shutdown, the fuel gamma source term will predominate. Therefore the N-gamma and hardware activation source terms can be neglected (Assumption 3). The source is defined on the MCNP *sdef* card using distributions to define the particle location and energy. The radius of the core is defined with the *rad* parameter, which automatically creates a uniform distribution based on a cylindrical geometry. The *ext* and *axs* parameters define the direction and distance of the cylinder axis. These parameters combined define the core where the particles can be born. The *erg* parameter defines the energy spectrum of source particles and is based on the results of the ORIGEN-S calculation discussed previously. This distribution is a histogram of energies represented by activities. These are automatically normalized by MCNP to create a probability distribution. The total activity is preserved in the tally multiplier. The MCNP source definition cards are shown below in Figure 7-9. The *sb* card is a source biasing card, which in this case biases the particle generation to the upper end of the core. This is a variance reduction technique to improve the statistical certainty in the results.

Figure 7-9 MCNP Source Definition Cards

```
sdef rad=d1 ext=d2 axs=0 0 1 erg=d8                                      ←Source Definition Card
                                                                                -Radius = d1
                                                                                -Extent = d2
                                                                                -Axis = +Z
                                                                                -Energy = d8
si1 209.71                                                                ←Core Radius Distribution
si2 h 0 42.672 85.344 128.016 170.688 213.36 256.032 298.704             ←Core Axial Distribution
       341.376 384.048 426.72
sp2 0 1 1 1 1 1 1 1 1 1 1                                                 ←Actual Uniform Distribution
sb2 0 0.001 0.001 0.01 0.01 0.01 0.1 0.1 0.1 1 1                          ←Biased to Top Distribution
c  Fuel Gamma Spectra
si8  h 1.000e-002 5.000e-002 1.000e-001 2.000e-001 3.000e-001 4.000e-001    ←Source Energy Groups
      6.000e-001 8.000e-001 1.000e+000 1.330e+000 1.660e+000 2.000e+000
      2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.000e+000
      1.000e+001 1.100e+001
sp8    0.00E+00 9.288E+19 2.926E+19 6.537E+19 4.277E+19 1.521E+19 3.578E+19 ←Source Emission on
                                                                                  Energy Basis
       4.352E+19 2.66E+19 1.289E+19 1.649E+19 5.572E+18 5.527E+18 1.984E+18
       7.812E+17 3.48E+16 3.947E+11 1.75E+08 37100000 2009000
```

7.5 MCNP Tally Specification

The tallies used in this evaluation are point detectors placed at approximate locations of radiation monitors RE-8055 and RE-8099. Point detectors are chosen because they use quasi-deterministic dose calculations that will provide better results than surface or cell based tallies that require the particles to enter those regions. The inputs to this card are the coordinates of the dose points followed by an exclusion zone (reduce variance), as well as a multiplier card, which represents the total core activity in photons/sec. The tally cards are shown in Figure 7-10.

Figure 7-10 MCNP Tally Cards

```
f5c RE-8055 and RE-8099                    ←Tally Comment Card
f5:p -1200 -400 1909.24 20                 ←Tally 5 (point detector)
1200  400 1909.24 20                         x     y     z     exclusion
                                            -1200 -400 1909.24    20
                                             1200  400 1909.24    20
fm5 3.947E+20                               ← Tally Multiplier
                                              (Total Activity)
```

In addition, the flux is multiplied by ANSI/ANS flux-dose conversion factors [11]. This is specified in MCNP using the *de/df* cards. These are shown in Figure 7-11.

Figure 7-11 ANSI/ANS-6.1.1-1977 Gamma Flux to Dose Conversion Factors

```
c -----------------------------------------------------------------
c ANSI/ANS-6.1.1-1977
c Gamma Flux to Dose Conversion Factors
c (mrem/hr)/(photons/cm2-s)
c -----------------------------------------------------------------
de0  .01 .03 .05 .07 .10 .15 .20 .25 .30 .35 .40             ←Energy Bins for Flux
     .45 .50 .55 .60 .65 .70 .80 1. 1.4 1.8 2.2                to Dose Conversion
     2.6 2.8 3.25 3.75 4.25 4.75 5. 5.25 5.75 6.25
     6.75 7.5 9. 11.
df0  3.96E-03   5.82E-04   2.90E-04   2.58E-04   2.83E-04   3.79E-04    ←Energy Dependent
     5.01E-04   6.31E-04   7.59E-04   8.78E-04   9.85E-04   1.08E-03      Flux Multipliers
     1.17E-03   1.27E-03   1.36E-03   1.44E-03   1.52E-03   1.68E-03
     1.98E-03   2.51E-03   2.99E-03   3.42E-03   3.82E-03   4.01E-03
     4.41E-03   4.83E-03   5.23E-03   5.60E-03   5.80E-03   6.01E-03
     6.37E-03   6.74E-03   7.11E-03   7.66E-03   8.77E-03   1.03E-02
```

7.6 MCNP Material Cards

The MCNP material cards are provided in Figure 7-12. These are based on the compositions described in Table 5-4.

Figure 7-12 MCNP Material Cards[5]

```
m1      92235     -0.0245
        92238     -0.5891
        8016      -0.2521
        40000     -0.1118
        50000     -0.0017
        24000     -0.0001
        26000     -0.0002
        1001      -0.0211
        6012      -0.0001
m2 1001 2 8016 1               $ Water
m3     6012 -0.000126          $ Air
       7014 -0.76508
       8016 -0.234793
m4     6000 -0.0008            $ SS 304
       14000 -0.01
       15031 -0.00045
       24000 -0.19
       25055 -0.02
       26000 -0.68375
       28000 -0.095
m5     26000 -0.014            $ Reg-Concrete
       1001 -0.01
       13027 -0.034
       20000 -0.044
       8016 -0.532
       14000 -0.337
       11023 -0.029
m6     6012 -0.01               $ Carbon Steel
       26056 -0.99
```

[5] Material 1 composition will change based on the water level relative to the core. This only applies to water heights below 14 feet.

7.7 Results

<u>File Naming Scheme:</u>

The MCNP input files are named with the following convention:

P-height-condition where:

P = Project (STP)

Height = water height from bottom of core (ft)

Condition = h – with head
n – no head

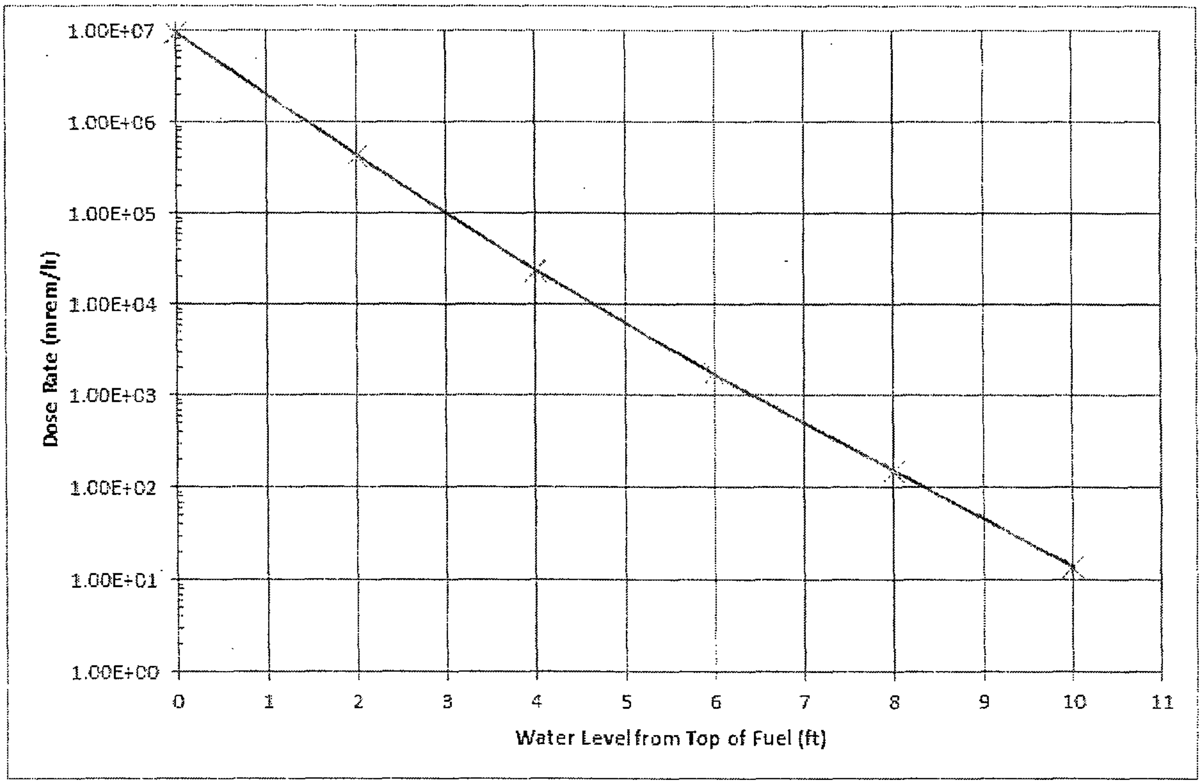
7.7.1 *Results without Head*

The dose rate as a function of water level is provided in Table 7-3 and plotted in Figure 7-13, below. Because the MCNP model geometry is symmetric in the x and y planes, the two point detector locations should provide the same dose rate. To increase the statistical certainty in the final result, the two individual dose rate responses and uncertainties are combined using inverse variance averaging. All of the water levels described in the following sections refer to the level at the top of the fuel (i.e. 0 foot water level is at the top of the fuel assemblies and ~13 feet is flange level).

Table 7-3 Dose Rate Response as a Function of Water Level for no Head Configuration (mrem/h)

Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	9.27E+06	0.0081	9.34E+06	0.0109	9.30E+06	0.0065
2	4.26E+05	0.0078	4.31E+05	0.0093	4.28E+05	0.0060
4	2.31E+04	0.0236	2.32E+04	0.0247	2.32E+04	0.0171
6	1.73E+03	3.10E-02	1.69E+03	2.44E-02	1.70E+03	0.0192
8	1.51E+02	0.0302	1.51E+02	0.0287	1.51E+02	0.0208
10	1.40E+01	0.036	1.36E+01	0.0323	1.38E+01	0.0240

Figure 7-13 Dose Rate versus Water Height Plot for no Head Configuration



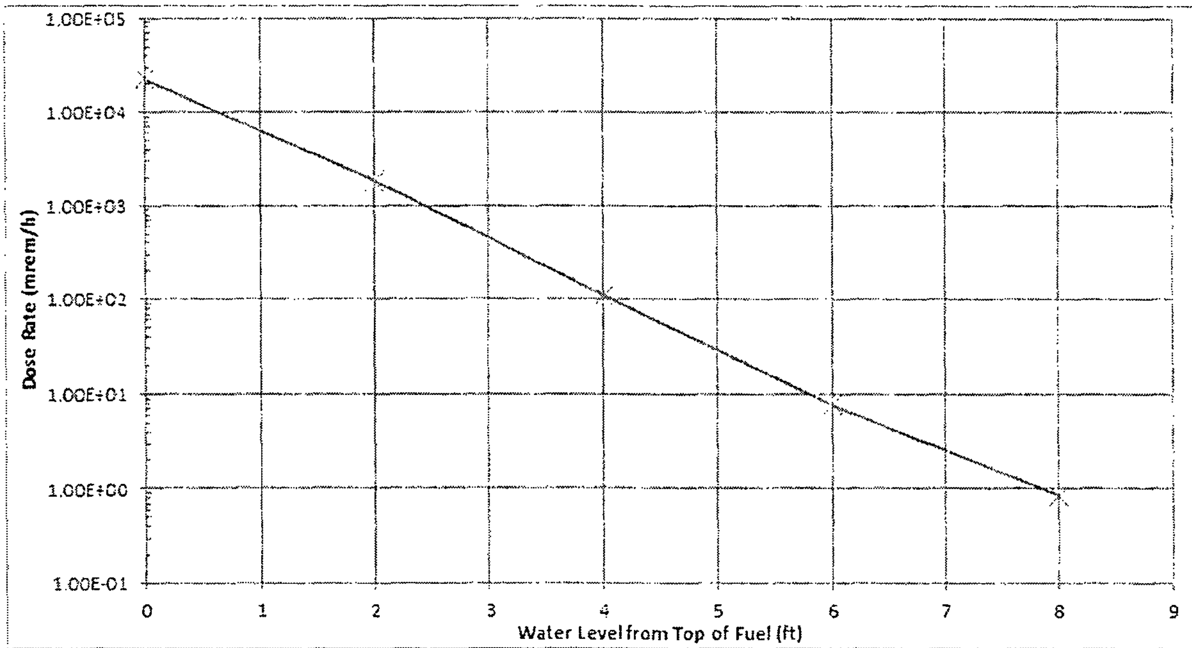
7.7.2 *Results with Head*

The dose rate results for the cases with the head in place are the same, except the minimum detectable dose rate is lower due to the lower ambient dose rate in the containment. The dose rates are listed in Table 7-4 and plotted in Figure 7-14.

Table 7-4 Dose Rate Response as a Function of Water Level for Head on Configuration (mrem/h)

Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	2.16E+04	0.094	2.56E+04	0.185	2.24E+04	0.0838
2	1.87E+03	0.083	1.83E+03	0.074	1.85E+03	0.0554
4	1.11E+02	0.061	1.08E+02	0.069	1.10E+02	0.0455
6	8.89E+00	0.085	7.48E+00	0.048	7.82E+00	0.0418
8	8.95E-01	0.125	8.12E-01	0.093	8.42E-01	0.0742

Figure 7-14 Dose Rate versus Water Height Plot for with Head Configuration



Appendix A – ENERCON Reference EMAILS

Drew Blackwell

From: Paul Sudnak
Sent: Monday, December 09, 2013 9:55 AM
To: Chad Cramer; Joanne Morris
Cc: Marvin Morris; Jeff Gromatzky; Michael Falkner; Jay Maisler; Caleb Trainor
Subject: RE: STP Refueling Cavity Level Calc

Sure, let me find the elevation drawing for the cavity. The water level during refueling is the same water level as the spent fuel pool during fuel transfer. The height of the active fuel is 28'-2 inches. The vessel flange level is 39'-3", and mid-loop is 32'-3".

RCS radiation monitors (RE-8055 and RE-8099) read from < 1 mR/hr to 2.5 mR/hr during refueling. If the upper-internal package or head are being removed, levels can increase to over ~ 100mR/hr for the upper internals. Levels on the refueling deck (68'0") at mid-loop will only increase to ~ 10 mR/hr with the water level that low. When the head is being de-tensioned by worker on the head level platform (~ 39' El.), dose rates at that location can read ~ 50 to over 100 mR/hr. The general area dose rates from core radiation is usually less than 100 mR/hr, unless there are lots of fuel leaks or high RCS corrosion and activation products. Dose rates at the monitors at flange level are usually less than 5 mR/hr.

Paul J. Sudnak
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30 ENERCON

Please consider the environment before printing this e-mail.

From: Chad Cramer
Sent: Friday, December 06, 2013 2:08 PM
To: Joanne Morris
Cc: Marvin Morris; Jeff Gromatzky; Paul Sudnak; Michael Falkner
Subject: STP Refueling Cavity Level Calc

Joanne,

Michael Falkner has completed the STP SFP calc and sent it to me for review. I spoke with he and Jeff Gromatzky and they indicated that he should have ability over the next week or so to do the refueling cavity level calc.

Paul,

Drew Blackwell

From:	Sudnak, Paul <pjsudnak@STPEGS.COM>
Sent:	Monday, February 03, 2014 3:44 PM
To:	Caleb Trainor; Drew Blackwell
Cc:	Domal, Michael; Jay Maisler
Subject:	RE: Fuel Assembly Dimension

Thanks Caleb,
Assume the only attenuation is from the materials between the detectors and the core. Disregard reflection. I don't think the SGs are between the core and detector, nor is the containment wall. Concrete should be high density. Atmosphere should be saturated steam at greater than 9.5 psi (containment spray initiation) and less than 56 psi (containment design pressure) mixed with air at the original containment volume at STP. The detectors are Ion chambers. Do not include neutrons. The reactor vessel head is around ~ 8" thick and carbon steel. I will get you the actual drawing of the RPV head thickness, but I think it is from the UFSAR,

Again, a peer check from Mike or Jay?
Paul

From: Caleb Trainor [mailto:ctrainor@enercon.com]
Sent: Monday, February 03, 2014 2:21 PM
To: Sudnak, Paul; Drew Blackwell
Cc: Domal, Michael; Jay Maisler
Subject: RE: Fuel Assembly Dimension

Drew is working on CS1/CG1 where the concern is direct shine from the core due to lowered water levels and no fuel damage assumed. I think you may be thinking of the fission product barrier calcs that I'm working on.

-Caleb

From: Sudnak, Paul <pjsudnak@STPEGS.COM>
Sent: Monday, February 3, 2014 3:05 PM
To: Caleb Trainor; Drew Blackwell
Cc: Domal, Michael; Jay Maisler
Subject: RE: Fuel Assembly Dimension

I think Caleb is right here. Once the concentration is known, the detectors are going to respond to the gases primarily above the 68' Elevation, all the rest will be significantly attenuated by the concrete floors, inner and outer Bio-shield wall, steam generators, and the pressurizer. To model all of those structures would require an extensive geometry and a considerable amount of data. Our intent here is to identify the concentration of gases above the 68' El and determine the monitor response. Disregard the Steam Generators, the inner and outer bio-shield walls, and the Pressurizer. With an assumed homogenous mix based on 20% fuel damage, the dose rates should be significant. Factoring in additional structures and elevations will not significantly change the outcome: A General Emergency will be declared.

Earlier today, I sent the location of the containment high range monitors (73' Elevation). Assume they see the volume of the reactor containment building above the 68' El.

Mike/Jay, can you give me a peer check here?
Paul

Appendix B – Electronic File Listing

```
Volume in drive F is My Passport
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02/07/2014  10:26 AM             8,795 Inverse Variance Weighting.xlsx
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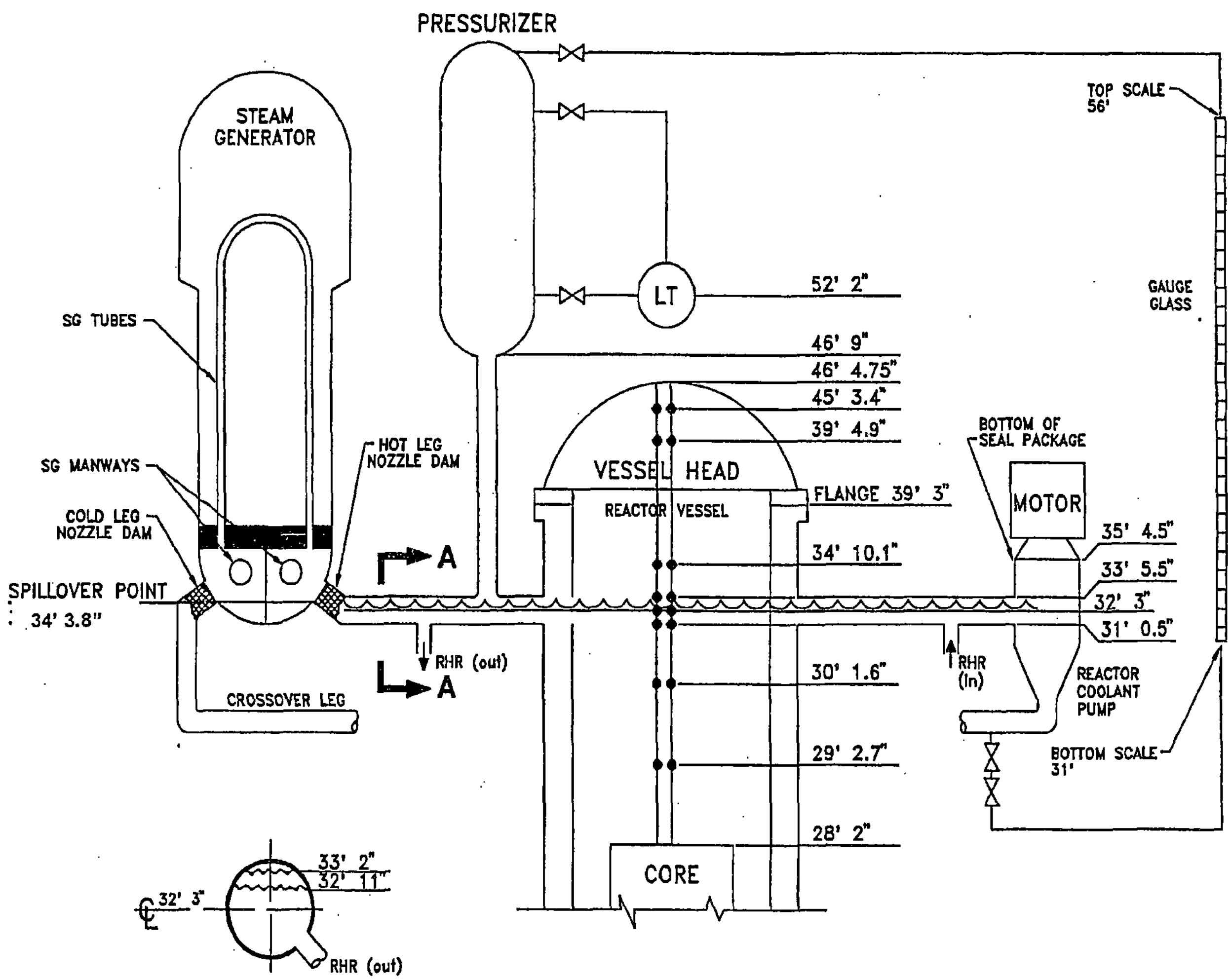
REACTOR COOLANT SYSTEM



SECTION A-A
HOT LEG

STP D-0794
Rev 2

NOTE

- Top of Core is elevation 28 ft 2 inches.
- SG spillover is elevation 34 ft 3.8 inches.

SENSOR UNCOVERED	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
All Covered	100	100	46' 4.75"	Upper Head Full
1	64	100	45' 3.4"	Upper Head Partially Drained
2	0	100	39' 4.9"	Plenum Full
3	0	85	34' 10.1"	Plenum **NOT** Full (Enter Reduced Inventory)
4	0	66	33' 5.5"	Top of Hot Leg Nozzle
5	0	48	32' 3"	Hot Leg Centerline
6	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
7	0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
8	0	0	29' 2.7"	Upper Core Plate

TABLE 12.3.4-1

AREA RADIATION MONITORS

Reactor Containment Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8052 Incore Instrumentation Room (-1ft-6 in.)	10^{-1}-10^{4}	1,000
N1RA-RE-8053 Support across from elevator (-11 ft-3 in.)	10^{-1}-10^{4}	100
N1RA-RE-8054 West Stair Landing (19 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8055 North SG wall across from the head laydown area (68 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8056 Support across from elevator (52 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8099 South SG wall across from the in-containment fuel pool (68 ft-0 in.)	10^{-1}-10^{4}	100

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8081 ~11 ft S of cols. 30.2 and S_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8084 ~24 ft S of cols. 28 and T_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8085 ~24 ft S of col. 28 and ~6 ft E of col. S_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8086 ~24 ft S of col. 28 and ~11 ft E of col. R_1 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8087 col. 30.2 and 12 ft W of col. R_1 (4 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8088 3 ft S of col. 30.9 and col. R_1 (30 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8089 col. 28 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8090 18 ft N of col. 30.2 and col. T_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8091 col. 34 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building (Continued)

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8097 33 ft S of cols. 28 and 10 ft W of col. N (68 ft-0 in.)	10^{-2}-10^{7}	1,000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8057 col. 22 and ~10 ft E of col. J (10 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8058 col. 26 and col. J (10 ft-0 in.)	10^{-1}-10^{4}	2.5
NIRA-RE-8059 col. 27 and col G (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8060 ~10 ft S of col. 30 and col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8061 ~10 ft S of col. 24 and ~11 ft W of col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8062 ~6 ft S of col. 31 and col. C (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8063 ~9 ft S of col. 28 and col. B (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8064 ~12 ft S of col. 24 and col. F (29 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8065 ~5 ft N of col. 32 and col. C (29 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.
2. The alarm setpoints listed are typical and may be varied as necessary.
3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [(1)]	Range (mR/hr) [(3)]	High Alarm Setpoint (mr/hr) [(2)]
N1RA-RE-8066 ~4 ft N of col. 22 and 14 ft E of col. C (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8067 col. 22 and 10 ft E of col. J (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8068 ~10 ft N of col. 25 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N2RA-RE-8068 ~10 ft S of col. 24 and col. G (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8069 ~12 ft S of col. 24 and ~14 ft E of col. C (41 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8070 col. 29 and col. C (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8071 ~18 ft S of col. 28 and 3 ft W of col. B (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8072 ~11 ft N of col. 29 and 5 ft W of col. D (41 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8073 col. 29 and col. E (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8074 ~5 ft S of col. 31 and ~7 ft W of col. C (41 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.
2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [(1)]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [(2)]
N1RA-RE-8075 col. 28 and ~3 ft W of col. G (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8076 col. 22 and ~10 ft E of col. J (60 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8077 col. 27 and col. J (60 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8078 col. 27 and col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8079 col. 25 and ~2 ft W of col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8080 col. 26 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8082 col. 28 and ~8 ft E of col. H (69 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8083 ~10 ft S of col. 29 and 15 ft W of col. E (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8098 ~6 ft N of col. 25 and col. H (60 ft-0 in.)	10^{2}-10^{7}	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Miscellaneous Buildings

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8092 col. 9 and col. P TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8093 col. 7 and col. M TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8094 ~3 ft N of col. 23 and ~14 ft W of col. B TSC-MEAB (72 ft-0 in.)	10^{-2}-10^{7}	1000

CN-2963

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Post-Accident Monitors

Tag Number and Location [1]	Range (R/hr)	High Alarm Setpoint (R/hr) [2]
A1RA-RE-8050 RCB (68 ft-0 in.)	10^0-10^8	2000
C1RA-RE-8051 RCB (68 ft-0 in.)	10^0-10^8	2000

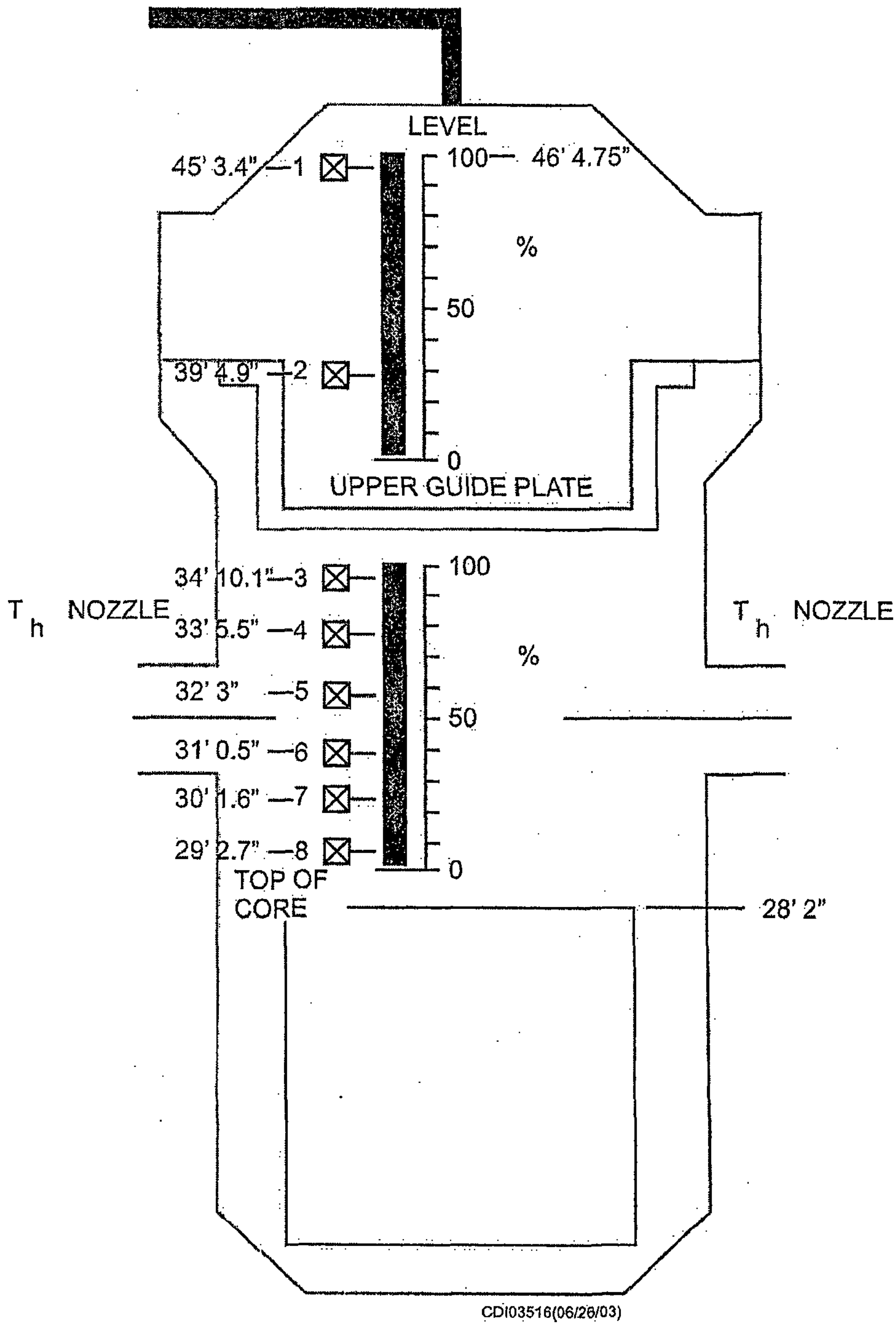
1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

NOTE

- Top of Core is elevation 28 ft 2 inches.
- SG spillover is elevation 34 ft. 3.8 inches.
- IF the Delta T is between 25°F and 200°F, THEN RECORD the sensor is "wet" (covered with water) in the Wet/Dry column of the table on Data Sheet 1 or 2.
- IF the Delta T is GREATER THAN 200°F, THEN RECORD the sensor is "dry" (**NOT** covered with water) in the Wet/Dry column of the table on Data Sheet 1 or 2.
- Example: IF SENSOR No. 3 DRY is circled AND SENSOR No. 4 WET is circled, THEN the RVWL PLENUM INDICATED LEVEL (%) would be 85%, SENSOR Location 34' 10.1" and the LEVEL DESCRIPTION would be Plenum **NOT** Full.

SENSOR No. WET/DRY (circle one)	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
All Wet	100	100	46' 4.75"	Upper Head Full
SENSOR No. 1 WET/DRY	64	100	45' 3.4"	Upper Head Partially Drained
SENSOR No. 2 WET/DRY	0	100	39' 4.9"	Plenum Full
SENSOR No. 3 WET/DRY	0	85	34' 10.1"	Plenum **NOT** Full
SENSOR No. 4 WET/DRY	0	66	33' 5.5"	Top of Hot Leg Nozzle
SENSOR No. 5 WET/DRY	0	48	32' 3"	Hot Leg Centerline
SENSOR No. 6 WET/DRY	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
SENSOR No. 7 WET/DRY	0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
SENSOR No. 8 WET/DRY	0	0	29' 2.7"	Upper Core Plate



RVWL Sensor Graphic

STEP DESCRIPTION FOR 0POP04-RC-0003 STEP 3.0

STEP: CHECK Trends For Any Of The Following Indications Of RCS Leakage:

- Rad Monitor RT8011 Particulate – Rising
- Reactor Coolant Drain Tank Level – Rising
- Pressurizer Relief Tank Level – Rising
- RCB Normal Sump Level – Rising

PURPOSE: To determine if leakage is from RCS and not CVCS.

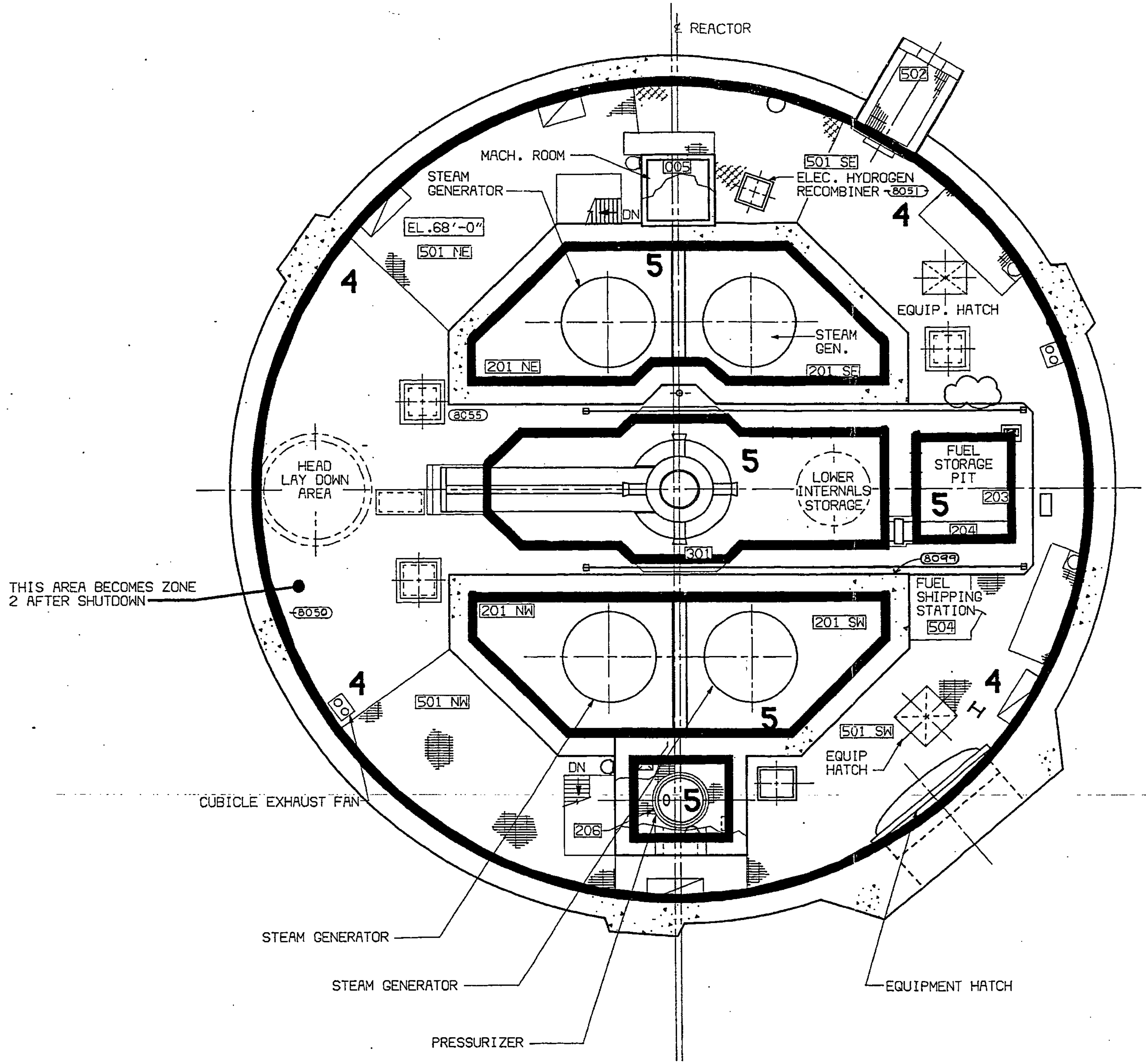
BASIS: Indication of RT8011, RCDT, PRT or RCB Normal Sump levels rising will confirm that the leakage is from RCS and not CVCS which is normally tied to the RCS.

ACTIONS: Monitor trends from RT8011, RCDT, PRT or RCB Normal Sump.

INSTRUMENTATION: Level indications located on CP004 and various plant computer monitors located in control room. Radiation Monitor Computer RM-11.

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A



RAD. MONITOR

RAD. MONITOR-SAFETY RELATED

PLANT RADIATION SHIELDING ZONES			
ZONE NUMBER	MAX. DOSE RATE (MREM/HR.)	POSTING REQUIRED	ANTICIPATED ACCESS
1	≤ 0.5	NO	UNRESTRICTED ACCESS
2	≤ 2.5	NO	CONTROLLED, 40HR/WK PERMISSIBLE
3	≤ 15	YES	CONTROLLED ACCESSIBLE ON A PERIODIC BASIS
4	≤ 100	YES	CONTROLLED LIMITED ACCESS
5	> 100	YES	NORMALLY INACCESSIBLE

NO.	ISSUE DATE	REVISION	BY	CKR	RE	DV	NA	SE	PE
3	7-1-97	EDITORIAL CHANGE TO REMOVE EXTRA RAD. MONITOR #8099. (F-4) REF. CR 96-15540-4.	VJM			—	—		—
0-2	6-17-83	ISSUED FOR CONSTRUCTION.	SIGNATURES ON FILE						

EDWG. DGN

NO.	ISSUE DATE	REVISION	BY	CKR	RE	DV	NA	SE	PE

HOUSTON LIGHTING & POWER COMPANY

SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION

RADIATION ZONES
REACTOR CONTAINMENT BUILDING
PLAN AT EL. 68' -0"

SCALE	DWG. NO.	REV.
1/8" = 1'	9C129A81105	3

RMSECM NO. 9C129A81105 | PRIORITY 2 | DC. NO. 0070928 | A1043.CIT A1043.031

8 7 6 5 4 3 2

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CG1

ENERCON Excellence—Every project. Every day	CALCULATION COVER SHEET	CALC. NO. STPNOC013-CALC-006
		REV. 1
		PAGE NO. 1 of 42
Title:	Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds	**Client:** STP
		Project: STPNOC013

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions that require confirmation? (If **YES**, identify the assumptions) ______	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ______	☐	☒
3	Does this calculation Supersede an existing Calculation? (If **YES**, identify the superseded calculation.) **Superseded Calculation No.** ______	☐	☒

Scope of Revision:

Added reference for reactor vessel head thickness, and updated calculations with new value (7.19 in). Removed any detector specific calculations so results can be applied to any detector at these locations. Made several editorial changes.

Revision Impact on Results:

The dose rates for the cases with reactor vessel head attached are higher due to the reduction in head thickness.

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☒ **Non-Safety Related** ☐

(Print Name and Sign)

Originator: Andrew Blackwell	[signature]	**Date:** 3/21/14
Design Verifier: Curt Lindner	[signature]	**Date:** 3/21/14
Approver: Marvin Morris	Marvin Morris (digitally signed)	**Date:**

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC. NO. STPNOC13-CALC-006
		REV. 1
		PAGE NO. 2 of 42

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	02/07/2014	
1	03/21/2014	Updated containment dimensions including reactor vessel head thickness. Added more detail to calculations section. Made editorial changes.

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	0		
5-16,18,19,21-26,28,30,31,34-42	1		

APPENDIX REVISION STATUS

APPENDIX NO.	PAGE NO.	REVISION NO.	APPENDIX NO.	PAGE NO.	REVISION NO.
A	All	0			
B	All	1			

Calculation Design Verification Plan:

The calculation will be reviewed for correctness of inputs, design criteria, analyzed methods, and acceptance criteria.

The stated objectives and conclusions will be confirmed to be reasonable and valid.

Assumptions will be reviewed and confirmed to be appropriate and verified to be valid based on sound engineering principles and practices.

(Print Name and Sign for Approval – mark "N/A" if not required)

Approver: Marvin Morris Marvin Morris	**Date:**

Calculation Design Verification Summary:

The calculation has been designated as **Safety Related** as noted in the cover sheet.

The calculation has been verified to be correct and performed using appropriate design inputs, assumptions, analytical methods, design criteria, and acceptance criteria.

The conclusions developed in the calculation are reasonable, valid, and consistent with the purpose and scope.

The assumptions are appropriate and valid.

Based On The Above Summary, The Calculation Is Determined To Be Acceptable.

(Print Name and Sign)

Design Verifier: Curt Lindner	**Date:** 3/21/14
Others:	**Date:**

ENERCON Excellence—Every project. Every day.	CALCULATION DESIGN VERIFICATION CHECKLIST	CALC. NO. STPNOC13-CALC-006
		REV. 1
		PAGE NO. 4 of 42

Item	CHECKLIST ITEMS	Yes	No	N/A
1	**Design Inputs** - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis, and incorporated in the calculation?	X		
2	**Assumptions** - Were the assumptions reasonable and adequately described, justified and/or verified, and documented?	X		
3	**Quality Assurance** - Were the appropriate QA classification and requirements assigned to the calculation?	X		
4	**Codes, Standards, and Regulatory Requirements** - Were the applicable codes, standards, and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied?	X		
5	**Construction and Operating Experience** - Have applicable construction and operating experience been considered?		X	
6	**Interfaces** - Have the design-interface requirements been satisfied, including interactions with other calculations?	X		
7	**Methods** - Was the calculation methodology appropriate and properly applied to satisfy the calculation objective?	X		
8	**Design Outputs** - Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives, and are the results reasonable compared to the inputs?	X		
9	**Radiation Exposure** - Has the calculation properly considered radiation exposure to the public and plant personnel?		X	
10	**Acceptance Criteria** - Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	X		
11	**Computer Software** - Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?	X		

COMMENTS:

In accordance with CSP 3.02, MCNP5 and SCALE6.0 have been verified for use on ENERCON computers.

	(Print Name and Sign)	
Design Verifier:	Curt Lindner [signature]	**Date:** 3/21/14
Others:		**Date:**

Table of Contents

List of Figures

List of Tables

1. Purpose and Scope

The purpose of this calculation is to evaluate dose rates as a function of water height in the reactor vessel during refueling operations in order to set Emergency Action Level (EAL) thresholds for core uncovery. The dose rates are calculated at the locations of the containment monitors RE-8055 and RE-8099 so that dose rate measurements by these devices can be used to estimate water level in the core, upon failure of other water level detection systems. This evaluation will calculate the dose rate at full core uncovery, as well as maximum water levels with a detectable dose rate response. Since the scope of this calculation concerns uncovering the reactor core, the effects of future fuel element storage in the nearby Fuel Storage Pit are not analyzed, since it's effects are negligible in comparison. The containment building, components within the building, and the reactor vessel and contents are modeled simplistically because only order of magnitude results are needed. As such, the dose rate results should be considered as reasonably representative of the magnitude of the actual dose rate only.

2. Summary of Results and Conclusion

The dose rate results for the configuration without the reactor vessel head and with the reactor vessel head are provided in Section 7.7.1 and Section 7.7.2, respectively. The dose rate with the core uncovered (i.e. water at the top of the active length) is 2.23E+04 mrem/h with the head in place and 9.30E+06 mrem/h with the head removed. Detailed results of the dose rate as a function of water height are provided in Figure 7-13 with the head removed and Figure 7-14 with the head attached.

3. References

1. "Standard Composition Library," ORNL/NUREG/CSD-2/V1/R6, Volume 3, Section M8, March 2000.
2. Calculation NC-6510. "Core Radionuclide Inventory for Chapter 15 Accident Analysis."
3. RSICC Code Package CCC-750, "SCALE 6.0: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Jan. 2009.
4. "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms", I.C. Gauld, O.W. Hermann, & R. M. Westfall. Jan. 2009.
5. STP001-CPC-001. Computer Program Certification MCNP5 Version 1.4 and SCALE 6.0.
6. ENERCON email from Paul Sudnak, dated December 9, 2013. (Appendix A).
7. Drawing 6C-18-N-5006, Rev. 9. "General Arrangement Reactor Containment Building Plan at El. 68' 0" Area G."
8. Drawing 6C-18-9-N-5007, Rev. 6. "General Arrangement Reactor Containment Building Section A-A Area G."
9. Drawing 6C-18-9-N-5008, Rev. 8. "General Arrangement Reactor Containment Building Section B-B Area G."
10. RSICC Code Package CCC-730, "MCNP/MCNPX Monte Carlo N-Particle Transport Code System 12 Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," January 2006.
11. ANSI/ANS 6.1.1-1977, Neutron and Gamma Flux-To-Dose Conversion Factors.
12. ENERCON email from Paul Sudnak, dated February 3, 2014 (Appendix A).
13. ~~Drawing L5-01EM101, Rev. 1. "Closure Head General Assembly."~~
14. Drawing 1142E24. "Model 4XLR Reactor 173 in. I.D. Vessel."
15. Drawing 2C26-9-S-1004, Rev. 4. "Steel Reactor Containment Building Cylindrical Shell Liner Sects. And Dets. Unit N° 1 & 2."
16. Drawing 1211E6. "4 Loop Rapid XL Reactor General Assembly."

4. Assumptions

The following assumptions are used in the core uncovery dose rate calculation:

1. The core is homogenized based on the typical Vantage 5 fuel assembly dimensions, taking into account the fuel rods and space between. Any small variations in fuel parameters will have a negligible effect on containment dose rates.
2. Any non-fuel hardware is ignored since the primary self-shielding occurs in the fuel itself, and there may be some unknown streaming effects through the non-fuel hardware. This homogenization takes into account the water level when calculating the isotopic weight fraction and homogenized density.
3. The source term for this evaluation is based on the fission product inventory at the time of shutdown. Because there is no cooling time, the fuel gamma source term will predominate and the N-gamma and hardware activation can be neglected.
4. The compositions of the containment structure and components are based on the values in the SCALE standard composition library [1].
5. The RE-8055 and RE-8099 monitors are assumed to be 5 feet above the 68 foot level in order to take into account the mounting device.
6. The containment outer concrete thickness is modeled as 3 feet thick. Because the backscattering off the containment walls is due to the steel liner, this dimension has a negligible impact on dose rates near the reactor vessel.

5. Design Inputs

5.1 Fuel Assembly Parameters

The following fuel assembly parameters are used in the core homogenization in the MCNP model. They are based on typical fuel assembly values for Westinghouse Vantage 5 fuel.

Table 5-1 Design Input Fuel Assembly Parameters for Westinghouse Vantage 5 Fuel

Parameters	Value	Unit	Reference
Fuel Type	Westinghouse Vantage +		Assumption 1
# Fuel Rods per Assy	264		Assumption 1
Assembly Array	17x17		Assumption 1
Enrichment	4	wt %	Assumption 1
Density (% of theoretical)	0.95		Assumption 1
Fuel Pellet OD	0.3225	[in]	Assumption 1
Fuel Rod Pitch	0.496	[in]	Assumption 1

Parameters	Value	Unit	Reference
Fuel Rod OD	0.374	[in]	Assumption 1
Clad Thickness	0.0225	[in]	Assumption 1
Guide Tube OD	0.482	[in]	Assumption 1
Guide Tube Thickness	0.020	[in]	Assumption 1
# Guide Tubes	24		Assumption 1
Instrument Tube OD	0.482	[in]	Assumption 1
Instrument Tube Thickness	0.020	[in]	Assumption 1
# Instrument Tubes	1		Assumption 1
Active Length	14	[ft]	Assumption 1

5.2 Containment Dimensions

The following dimensions are based on drawings of the STP containment building and equipment. Some parameters are estimated using scaling when the drawings do not detail the exact dimension. These estimations are only applied to dimensions that have a negligible effect on the core uncovery dose rate analysis.

Table 5-2 Design Input Containment Dimensions

Dimension:	ft	in	cm	reference
Reactor Pressure Vessel				
Elevation at top of active fuel	28	2	858.52	[6]
Elevation at head level platform	38	6.5	1174.75	[8]
Elevation at full water level in refueling cavity	66	6	2026.92	[8]
Closure head thickness	0	7.19	18.2626	[13]
Reactor pressure vessel inside diameter at shell	0	173	439.42	[14]
Height of reactor vessel from bottom of fuel to head level			742.95	Calculated
Steam Generator				
Elevation at bottom of SG	38	4	1168.4	[9]
Elevation at top of SG	105	9.875	3225.4825	[9]
Total SG height			2057.0825	Calculated
SG outer diameter			500	[7] Scaled

Dimension:	ft.	in	cm	reference
Active Fuel				
Active fuel bottom elevation	12	1	368.3	[9]
Active fuel height	14	0	426.72	[14]
Concrete Wall				
Lower Height	38	6.5	1174.75	[9]
Upper Height	85	0	2590.8	[9]
Overall Height			1416.05	Calculated
Thickness	2	0	106	[7] Scaled
Width			874.776	[7] Scaled
Length			2499.36	[7] Scaled
Steam Generators				
Lower Modeled Height	85	0	2590.8	[9]
Upper Modeled Height	105	9.875	3225.4825	[9]
Overall Modeled Height			634.6825	Calculated
Diameter			500	[7] Scaled
Containment				
Upper modeled height	153	0	4663.44	[8]
Lower modeled height	68	0	2072.64	[8]
Net Height			2590.8	Calculated
Inner Diameter	149	$11^1/_4$	4570	[15]
Liner Thickness	0	0.375	0.9525	[15]
Dome Inner Radius	74	$11^5/_8$	2285	[15]
Concrete Thickness	3	0	91.44	Assumption 6

5.3 Core Isotopic Inventory

Core isotopic activities are provided in Table 11 of [2]. The isotope specific activities are given in terms of Ci/MWt, which is converted to curies based on the total core thermal power of 4,100 MWt [2]. These calculations are performed in EXCEL spreadsheet *STP.xlsx*. A table of the input values is shown in Table 5-3, below.

Table 5-3 Design Basis Core Shutdown Source Term[1]

Isotope	Ci/MWt	Ci	Isotope	Ci/MWt	Ci
Kr83m	3.41E+03	1.40E+07	Ru106	1.34E+04	5.49E+07
Kr85m	7.07E+03	2.90E+07	Rh105	3.05E+04	1.25E+08
Kr85	2.93E+02	1.20E+06	Zr95	4.39E+04	1.80E+08
Kr87	1.34E+04	5.49E+07	Zr97	4.39E+04	1.80E+08
Kr88	1.90E+04	7.79E+07	Nb95	4.32E+04	1.77E+08
Kr89	2.32E+04	9.51E+07	La140	4.63E+04	1.90E+08
Xe131m	2.68E+02	1.10E+06	La141	4.62E+04	1.89E+08
Xe133m	1.66E+03	6.81E+06	La142	4.15E+04	1.70E+08
Xe133	5.37E+04	2.20E+08	Pr143	3.90E+04	1.60E+08
Xe135m	1.02E+04	4.18E+07	Nd147	1.73E+04	7.09E+07
Xe135	1.34E+04	5.49E+07	Am241	2.75E+00	1.13E+04
Xe137	4.63E+04	1.90E+08	Cm242	1.05E+03	4.31E+06
Xe138	4.39E+04	1.80E+08	Cm244	6.17E+01	2.53E+05
I131	2.59E+04	1.06E+08	Ce141	4.39E+04	1.80E+08
I132	3.71E+04	1.52E+08	Ce143	4.15E+04	1.70E+08
I133	5.37E+04	2.20E+08	Ce144	3.41E+04	1.40E+08
I134	5.85E+04	2.40E+08	Np239	5.12E+05	2.10E+09
I135	4.88E+04	2.00E+08	Pu238	8.71E+01	3.57E+05
Sb127	3.05E+03	1.25E+07	Pu239	1.96E+01	8.04E+04
Sb129	8.29E+03	3.40E+07	Pu240	2.48E+01	1.02E+05
Te127m	4.32E+02	1.77E+06	Pu241	4.17E+03	1.71E+07
Te127	3.05E+03	1.25E+07	Rb86	9.92E+01	4.07E+05
Te129m	1.22E+03	5.00E+06	Cs134	5.37E+03	2.20E+07
Te129	8.05E+03	3.30E+07	Cs136	1.54E+03	6.31E+06
Te131m	3.66E+03	1.50E+07	Cs137	3.17E+03	1.30E+07
Te132	3.82E+04	1.57E+08	Y90	3.56E+03	1.46E+07
Ba137m	2.93E+03	1.20E+07	Y91	3.41E+04	1.40E+08
Ba139	4.98E+04	2.04E+08	Y92	3.41E+04	1.40E+08
Ba140	4.63E+04	1.90E+08	Y93	3.90E+04	1.60E+08
Mo99	4.83E+04	1.98E+08	Sr89	2.68E+04	1.10E+08
Tc99m	4.07E+04	1.67E+08	Sr90	2.37E+03	9.72E+06
Ru103	3.90E+04	1.60E+08	Sr91	3.17E+04	1.30E+08
Ru105	2.68E+04	1.10E+08	Sr92	3.41E+04	1.40E+08

[1] Ci = Ci/MWt × 4,100 MWt

5.4 Material Compositions

The following compositions used in the MCNP model are taken from the SCALE standard composition library [1] and are shown in Table 5-4.

Table 5-4 SCALE Standard Compositions used in MCNP Model

Material	Isotope	Weight Fraction	Reference
Zry- 4	Zr	0.9823	[1]
(6.56 g/cm^3)	Sn	0.0145	
	Cr	0.0010	
	Fe	0.0021	
	Hf	0.0001	
UO_2	U-235	0.0353	[1]
(10.412 g/cm^3)[2]	U-238	0.8461	
	O	0.1186	
Air	C	0.0001	[1]
(1.21E-03 g/cm^3)	N	0.7651	
	O	0.2348	
Water	H	0.1111	[1]
(0.9982 g/cm^3)	O	0.8889	
SS-304	Fe	0.6838	[1]
(7.94 g/cm^3)	Cr	0.1900	
	Ni	0.0950	
	Mn	0.0200	
	Si	0.0100	
	C	0.0008	
	P	0.0004	
Concrete	O	0.5320	[1]
(2.30 g/cm^3)	Si	0.3370	
	Ca	0.0440	
	Al	0.0340	
	Na	0.0290	
	Fe	0.0140	
	H	0.0100	
Carbon Steel	C	0.0100	[1]
(7.82 g/cm^3)	Fe	0.9900	

[2] Based on 95% of theoretical density, Assumption 1.

6. Methodology

The reactor source terms are computed with ORIGEN-S of the SCALE 6.0 code package [3, 4]. The ORIGEN-S decay sequence is used to bin design input isotope specific activities into energy dependent photon bins. These energy specific photon emission bins are used as input for the energy distribution described by the MCNP source definitions.

The ORIGEN-S sequence in the SCALE6.0 program package is verified for use in safety related calculations [5]. The program certification form is maintained in the project file.

MCNP5, release 1.40 [10], Monte Carlo transport is used to determine the dose rates. The ENDF/B-VI neutron cross section library, ENDF60, and the ENDF/B-VI Release 8 Photo-atomic Data gamma cross section library, MCPLIB04 are utilized in the transport computations. This software has been verified for use in safety related calculations [5].

The detailed engineering drawings are converted into MCNP surface and cell cards in the proper dimensions. The radiation monitors of interest are modeled as point detectors to determine the expected dose rate for those detectors. The dose rates are calculated as a function of water height for two reactor refueling conditions:

1. With Head – the reactor is modeled with an 7.19 inch carbon steel plate as indicated in Table 5-2, which is additional attenuation between source and detector.
2. Without head – the reactor is modeled with nothing between the active fuel zone and containment.

For low water levels, variance reduction is accomplished with a geometric importance map that is imposed on the homogenized core. Without significant amounts of water present, this is enough to calculate statistically sound dose rate results. Once the water depth reaches a height where the variance of the results reaches an unacceptable level, a superimposed weight windows mesh is utilized to improve the variance reduction of the simple geometric scheme. The weight windows are iteratively generated using the MCNP weight windows generator card with a mesh over the existing geometry. All final dose rates presented in this calculation include weight windows variance reduction.

7. Calculations

7.1 Source Terms

In order to convert the isotope specific activity into an energy spectrum, ORIGEN-S of the SCALE6.0 code package is used to initiate a decay and bin into 19 photon energy groups. The energy groups along with their associated activities are used in the MCNP source definition to model the anticipated radiation emission following shutdown.

The ORIGEN-S input deck, *STPEAL.inp*, is provided below in Figure 7-1. This input has a simple decay case where the inputted isotopic composition in curies is decayed. The isotope is specified in the 73$$ card using the special identifier described in Section F7.6.2 of the ORIGEN-S manual, and the activity in curies is specified in the 74** card. The time steps for the decay are given on the 60** card in years. Although multiple time steps are calculated, the source term with zero decay time is used in this calculation to model the core immediately after shutdown. The output of the decay is given in terms of photons/s/Energy-Group, which is automatically normalized in the MCNP input.

Figure 7-1 ORIGEN-S Input Deck for MCNP Source Term Calculation

```
=origens                                                     ←Call Origen-S Sequence
0$$ a11 71 e t                                               ←Logical Unit Assignments
                                                                  -Binary Photon Library (71)
PWR Source Term STP ELA Analysis                             ←Case Title
3$$  21 1 1 a4 27 a16 4 a33 19 e t                           ←Library Integer Constants
                                                                  -Units 83** Card Ci (4)
                                                                  -Gamma Energy Groups (19)
35$$ 0 t                                                     ←Not Used
54$$ a8 0 a11 2 e                                            ←Special Calculation Options
                                                                  -Cutoff Value (Default)
                                                                  -(α,n) Composition Dependent
56$$ 0 6 a6 1 a10 0 a13 66 5 3 0 2 0 e                       ←Subcase Control Constants
                                                                  -Decay Only Subcase (0)
                                                                  -Number of Time Intervals (6)
                                                                  -Number of Nuclides (66)
                                                                  -Unit of Time in Years (5)
57** 0  a3 1-16 e                                            ←Not Used
95$$ 0 t                                                     ←Not Used
STPEAL                                                       ←Subcase Title
Ci Source Terms                                              ←Subcase Basis
60** 0 0.1 0.2 0.3 0.4 0.5                                   ←Time (years)
61** 5r1-8 1+6 1+4                                           ←Cutoff Values
65$$                                                         ←Decay Period Print Triggers
'GRAM-ATOMS  GRAMS    CURIES    WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0    1 0 0     1 0 0        3Z          6Z
     3Z      1  1  1    1 0 1     1 1 1        3Z          6Z
     3Z      1  1  1    1 1 1     1 1 1        3Z          6Z
81$$ 2 0 26 1 e                                              ←Gamma Source Constants
82$$ f2                                                      ←Produces Gamma Source Spectrum
83** 1.10E+07 1.00E+07 8.00E+06 6.50E+06 5.00E+06 4.00E+06 3.00E+06 ←Gamma Energy Groups
     2.50E+06 2.00E+06 1.66E+06 1.33E+06 1.00E+06 8.00E+05 6.00E+05
     4.00E+05 3.00E+05 2.00E+05 1.00E+05 5.00E+04 1.00E+04 e
84** 2.00E+07 6.43E+06 3.00E+06 1.85E+06 1.40E+06 9.00E+05 4.00E+05 ←Neutron Energy Groups
     1.00E+05 1.70E+04 3.00E+03 5.50E+02 1.00E+02 3.00E+01 1.00E+01 (Not Used)
     3.05E+00 1.77E+00 1.30E+00 1.13E+00 1.00E+00 8.00E-01 4.00E-01
     3.25E-01 2.25E-01 1.00E-01 5.00E-02 3.00E-02 1.00E-02 1.00E-05 e
73$$ 360831 360851 360850 360870 360880 360890 541311 541331 ←Nuclide Identifiers
    541330 541351 541350 541370 541380 531310 531320 531330
    531340 531350 511270 511290 521271 521270 521291 521290
    521311 521320 561371 561390 561400 420990 430991 441030
    441050 441060 451050 400950 450970 410950 571400 571410
    571420 591430 601470 952410 962420 962440 581410 581430
    581440 932390 942380 942390 942400 942410 370860 551340
    551360 551370 390900 390910 390920 390930 380890 380900
    380910 380920
74** 1.40E+07 2.90E+07 1.20E+06 5.49E+07 7.79E+07 9.51E+07 1.10E+06 ←Nuclide Concentrations (Ci)
     6.81E+06 2.20E+08 4.18E+07 5.49E+07 1.90E+08 1.80E+08 1.06E+08
     1.52E+08 2.20E+08 2.40E+08 2.00E+08 1.25E+07 3.40E+07 1.77E+06
     1.25E+07 5.00E+06 3.30E+07 1.50E+07 1.57E+08 1.20E+07 2.04E+08
     1.90E+08 1.98E+08 1.67E+08 1.60E+08 1.10E+08 5.49E+07 1.25E+08
     1.80E+08 1.80E+08 1.77E+08 1.90E+08 1.89E+08 1.70E+08 1.60E+08
     7.09E+07 1.13E+04 4.31E+06 2.53E+05 1.80E+08 1.70E+08 1.40E+08
     2.10E+09 3.57E+05 8.04E+04 1.02E+05 1.71E+07 4.07E+05 2.20E+07
     6.31E+06 1.30E+07 1.46E+07 1.40E+08 1.40E+08 1.60E+08 1.10E+08
     9.72E+06 1.30E+08 1.40E+08
75$$ 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 ←Library Kind
                                                              2-Actinide 3-Fission Product
    3 3 3 3 3 3 3 3 3 3 3 2 2 2 3 3 3 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3
    3 3
t
56$$ f0 t
End
```

The results of this calculation are summarized below in Table 7-1. These values will be used in the MCNP input source definition.

Table 7-1 Binned Total Core Source Term

Energy Group	Energy Boundaries (MeV)	Photons/sec
1	0.01-0.05	9.29E+19
2	0.05-0.1	2.93E+19
3	0.1-0.2	6.54E+19
4	0.2-0.3	4.28E+19
5	0.3-0.4	1.52E+19
6	0.4-0.6	3.58E+19
7	0.6-0.8	4.35E+19
8	0.8-1	2.66E+19
9	1-1.33	1.29E+19
10	1.33-1.66	1.65E+19
11	1.66-2	5.57E+18
12	2-2.5	5.53E+18
13	2.5-3	1.98E+18
14	3-4	7.81E+17
15	4-5	3.48E+16
16	5-6.5	3.95E+11
17	6.5-8	1.75E+08
18	8-10	3.71E+07
19	10-11	2.01E+06
totals		3.95E+20

7.2 MCNP Model Core Homogenization

Because the source term is given for the entire core, the self-shielding from the assemblies is an important part of the dose rate response in regions above the core. Particles born in the lower section of the core are very unlikely to penetrate through the core itself, and make it to the radiation monitors. For simplicity, the core is modeled as a 3 dimensional cylinder with a uniformly distributed spatial particle distribution. The calculations for the homogenization are done in the worksheet *Compositions* of the EXCEL workbook *STP.xlsx*. A density and isotopic composition is calculated with the water level above the top of the fuel. A summary of the calculations for the core composition and density is shown below. The inputs are based on the dimensions in Table 5-1 and the compositions in Table 5-4.

$$Rod\ Volume = \pi(Pellet\ Radius)^2 \times Active\ Length = (3.14)(0.16125\ in)^2(168\ in) = 13.7\ in^3$$

$$Rod\ Mass_{UO_2} = \rho \times V = \left(10.96\frac{g}{cc}\right)(0.95)(13.72\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 2341.5\ g$$

$$Assembly\ Mass_{UO_2} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (2341.5\ g)(264) = 618.2\ kg$$

$$Clad\ Volume = \pi\left(\frac{OD^2}{4} - \frac{ID^2}{4}\right) \times Active\ Length = (3.14)\left[\frac{(0.374\ in)^2}{4} - \frac{(0.329\ in)^2}{4}\right](168\ in) = 4.17\ in^3$$

$$Rod\ Mass_{Zry-4} = \rho \times V = \left(6.56\ \frac{g}{cc}\right)(4.17\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 448.7\ g$$

$$Assembly\ Mass_{Zry-4} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (448.7\ g)(264) = 118.5\ kg$$

$$Assembly\ \ H_2O\ Volume = [(Assembly\ Width)^2 - \pi(Rod\ Radius)^2 \times 264] \times Active\ Length = [(8.404\ in)^2 - (3.14)(0.187\ in)^2(264)](168\ in) = 6993\ in^3$$

$$Assembly\ Mass_{H_2O} = \rho \times V = \left(0.9982\frac{g}{cc}\right)(6993\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 114.4\ kg$$

$$Assembly\ Volume = Active\ Length \times (Assembly\ Width)^2 = (168\ in)(8.404\ in)^2 = 11865.4\ in^3$$

$$Density = \frac{Total\ Mass}{Volume} = \frac{1000(618.2 + 118.5\ kg + 114.4)\ kg}{11865.4\ in^3\left(2.54\frac{cm}{in}\right)^3} = 4.38\ g/cc$$

7.3 MCNP Model Geometry

The following MCNP model geometry is based on the containment dimensions summarized in Table 5-2. The model only focuses on the primary systems and components that provide shielding or reflection from the core to the radiation monitors. These components include the reactor vessel, concrete in reactor pit, containment walls (reflection), and steam generators (reflection). VISED plots of the model geometry are provided in Figure 7-2, Figure 7-3, and Figure 7-4. The MCNP surface cards with the model dimensions (cm) are shown in Figure 7-5, and the cell cards are shown in Figure 7-6 for the cases with no reactor head. A VISED plot of the model with the reactor head is shown in Figure 7-7. The surface and cell cards for the cases with the reactor head are shown in and Figure 7-8, respectively. Areas that are not of interest are given an importance of zero (white areas) so MCNP will not track particles in locations that will not contribute to the detector response. A summary of surfaces used in constructing this geometry is shown in Table 7-2, including a description of macrobody dimensions.

Figure 7-2 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (No Head)

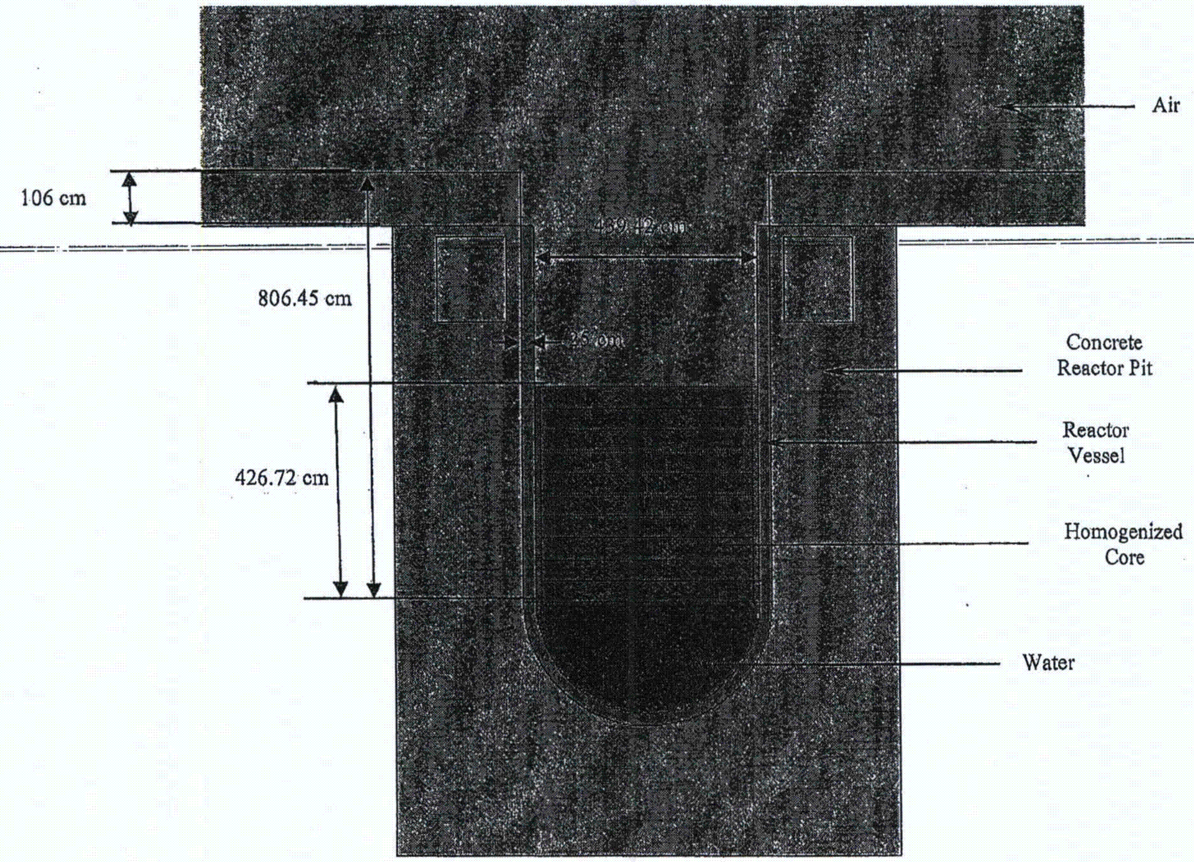
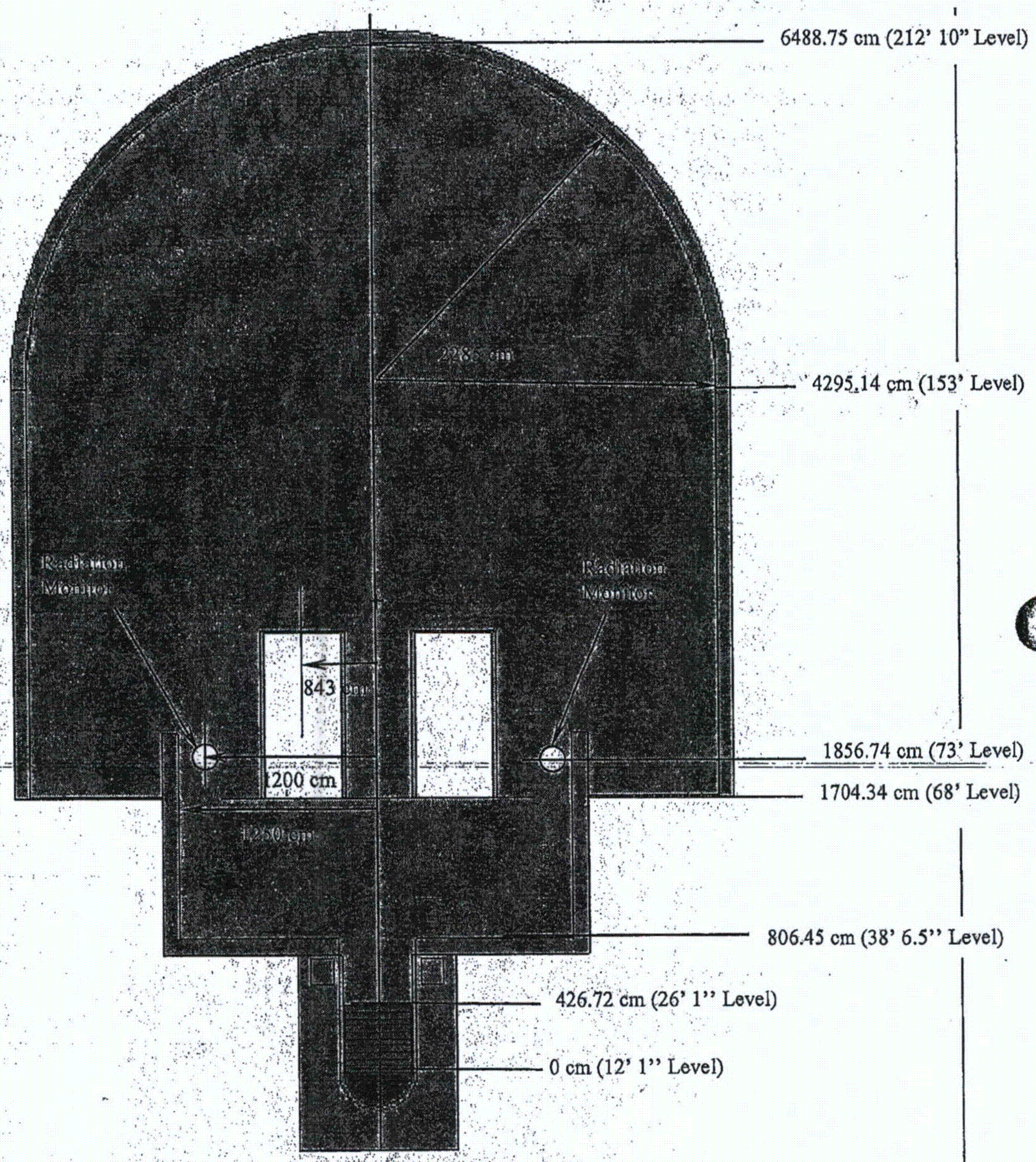


Figure 7-3 X-Z VISED Plot of Containment[3]



[3] Steam Generators are not full height. Also, they are not on the same X-Z plane as the core shown above. They are included for visualization purposes.

Figure 7-4 X-Y VISED Plot of the Containment Geometry at Radiation Monitor Level

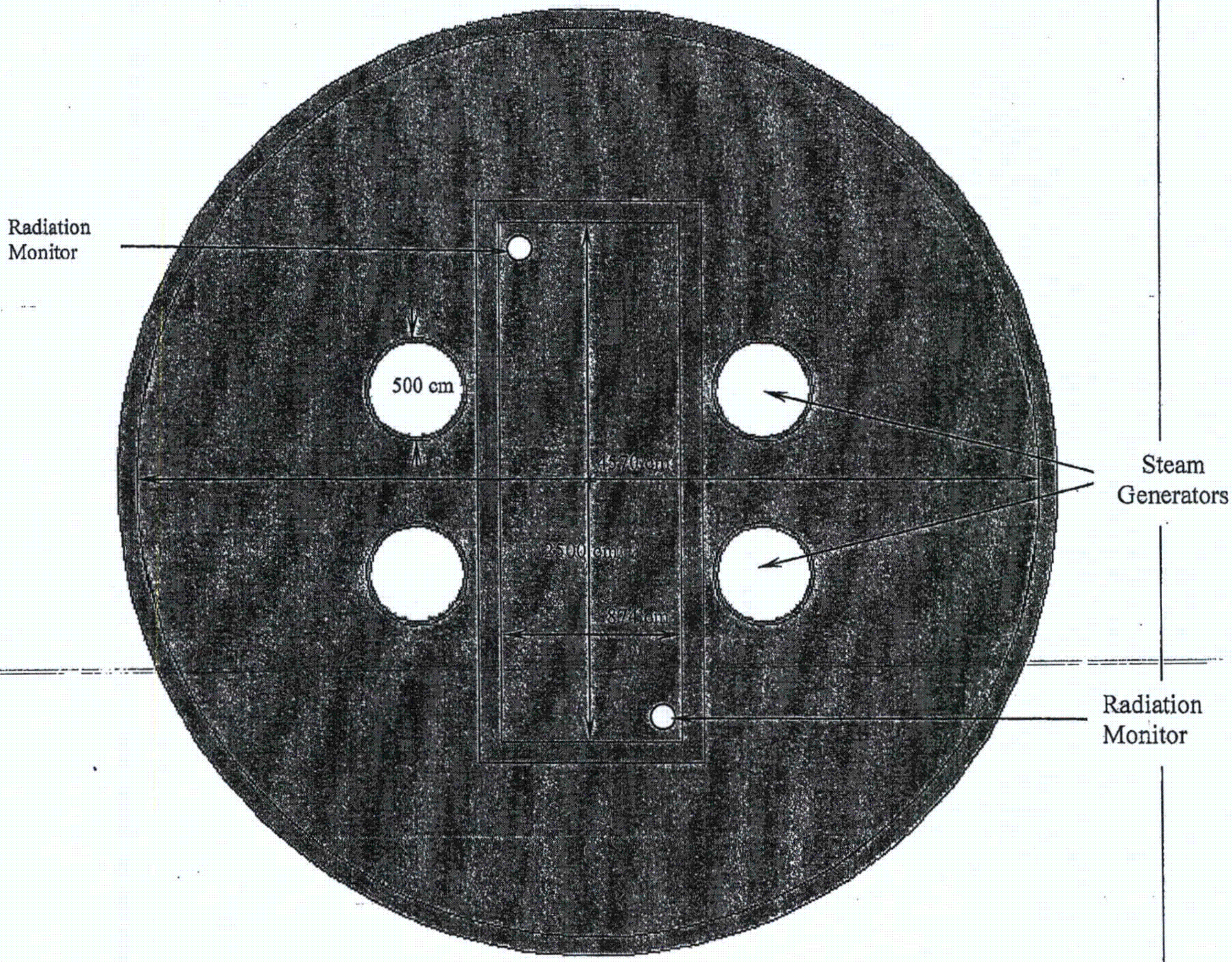


Table 7-2 Summary of Surfaces Used for MCNP Models

Surface Type	Surface Number	Dimensions							Description
RCC		X_0	Y_0	Z_0	V_X	V_Y	V_Z	R	
	1	0	0	0	0	0	426.72	209.71	Active Fuel Region
	2	0	0	0	0	0	700.45	219.71	Reactor Pressure Vessel Inner Surface
	3	0	0	0	0	0	700.45	244.71	Reactor Pressure Vessel Outer Surface
	31	0	0	700.45	0	0	18.26	244.71	Reactor Pressure Vessel Head
	41	0	0	512.81	0	0	167.64	274.71	Concrete Void for Primary Loop
	42	0	0	512.81	0	0	167.64	411.71	Concrete Void for Primary Loop
	10	0	0	700.45	0	0	106	244.71	Concrete Wall Cutout
	11	444.71	843	700.45	0	0	2050	250	Steam Generator 1
	12	444.71	843	720.45	0	0	2010	230	Steam Generator Inner 1
	13	-444.71	843	700.45	0	0	2050	250	Steam Generator 2
	14	-444.71	843	720.45	0	0	2010	230	Steam Generator Inner 2
	15	-444.71	-843	700.45	0	0	2050	250	Steam Generator 3
	16	-444.71	-843	720.45	0	0	2010	230	Steam Generator Inner 3
	17	444.71	-843	700.45	0	0	2050	250	Steam Generator 4
	18	444.71	-843	720.45	0	0	2010	230	Steam Generator Inner 4
	21	0	0	1694.34	0	0	2600.8	2285	Containment Inner Liner Surface
	22	0	0	1694.34	0	0	2600.8	2285.95	Containment Inner Concrete Surface
	23	0	0	1694.34	0	0	2600.8	2377.39	Containment Outer Concrete Surface
RPP		-X	X	-Y	Y	-Z	Z		
	4	-498	498	-498	498	-498	700.45		Concrete Surrounding RPV
	8	-1250	1250	-437	437	806.45	2116.45		Concrete Wall Fuel Pit Inner
	9	-1356	1356	-543	543	700.45	2116.45		Concrete Wall Fuel Pit Outer
SPH		X_0	Y_0	Z_0	R				

Surface Type	Surface Number	Dimensions							Description
	5	0	0	0	219.71				Bottom of Reactor Pressure Vessel Inner
	6	0	0	0	244.71				Bottom of Reactor Pressure Vessel Outer
	24	0	0	4295.14	2285				Containment Dome Inner Liner Surface
	25	0	0	4295.14	2285.95				Containment Dome Inner Concrete Surface
	26	0	0	4295.14	2377.39				Containment Dome Outer Concrete Surface
PZ		Z							
	7	0							Fuel Bottom
	71	700.45							Top of RPV
	20	Variable							Water Level
	27	4295.14							Spring Line
	28	1704.34							68' Level
	101-110	42.672	-	426.72					Geometric Importance Divisions in Active Zone

Figure 7-5 MCNP Model Surface Cards[4]

```
c surfaces
1 rcc 0 0 0 0 0 426.72 209.71                    $ Active Fuel Region
2 rcc 0 0 0 0 0 700.45 219.71                    $ Reactor Pressure Vessel Inner Surface
3 rcc 0 0 0 0 0 700.45 244.71                    $ Reactor Pressure Vessel Outer Surface
31 rcc 0 0 700.45 0 0 18.26 244.71               $ Reactor Vessel Head
4 rpp -498 498 -498 498 -498 700.45              $ Concrete Surrounding RPV
41 rcc 0 0 512.81 0 0 167.64 274.71              $ Concrete Void for Primary Loop
42 rcc 0 0 512.81 0 0 167.64 411.71              $ Concrete Void for Primary Loop
5 sph 0 0 0 219.71                               $ Bottom of Reactor Pressure Vessel
6 sph 0 0 0 244.71                               $ Bottom of Reactor Pressure Vessel
7 pz 0                                           $ Bottom of Active Zone
71 pz 700.45                                     $ Top of RPV
8 rpp -1250 1250 -437 437 806.45 2116.45         $ Concrete Walls Fuel Pit Inner
9 rpp -1356 1356 -543 543 700.45 2116.45         $ Concrete Wall Fuel Pit Outer
10 rcc 0 0 700.45 0 0 106 244.71                 $ Concrete Wall Cutout
11 rcc 444.71 843 700.45 0 0 2050 250            $ Steam Generator 1
12 rcc 444.71 843 720.45 0 0 2010 230            $ Inner Steam Generator 1
13 rcc -444.71 843 700.45 0 0 2050 250           $ Steam Generator 2
14 rcc -444.71 843 720.45 0 0 2010 230           $ Inner Steam Generator 2
15 rcc -444.71 -843 700.45 0 0 2050 250          $ Steam Generator 3
16 rcc -444.71 -843 720.45 0 0 2010 230          $ Inner Steam Generator 3
17 rcc 444.71 -843 700.45 0 0 2050 250           $ Steam Generator 4
18 rcc 444.71 -843 720.45 0 0 2010 230           $ Inner Steam Generator 4
20 pz 365.76                                     $ Water Elevation Surface
21 rcc 0 0 1694.34 0 0 2600.8 2285               $ Containment Inner Liner Surface
22 rcc 0 0 1694.34 0 0 2600.8 2285.95            $ Containment Inner Concrete Surface
23 rcc 0 0 1694.34 0 0 2600.8 2377.39            $ Containment Outer Concrete Surface
24 sph 0 0 4295.14 2285                          $ Containment Dome Inner Liner Surface
25 sph 0 0 4295.14 2285.95                       $ Containment Dome Inner Concrete Surface
26 sph 0 0 4295.14 2377.39                       $ Containment Dome Outer Concrete Surface
27 pz 4295.14                                    $ Spring Line
28 pz 1704.34                                    $ 68' Level
101 pz 42.672                                    $ Geometric Importance Division Fuel Zone
102 pz 85.344                                    $ Geometric Importance Division Fuel Zone
103 pz 128.016                                   $ Geometric Importance Division Fuel Zone
104 pz 170.688                                   $ Geometric Importance Division Fuel Zone
105 pz 213.36                                    $ Geometric Importance Division Fuel Zone
106 pz 256.032                                   $ Geometric Importance Division Fuel Zone
107 pz 298.704                                   $ Geometric Importance Division Fuel Zone
108 pz 341.376                                   $ Geometric Importance Division Fuel Zone
109 pz 384.048                                   $ Geometric Importance Division Fuel Zone
110 pz 426.72                                    $ Geometric Importance Division Fuel Zone
```

[4] The surface cards for the MCNP model without the reactor vessel head does not have surface 31. The other surfaces are identical.

Figure 7-6 MCNP Model Cell Cards (No Head)

```
c cells
101 1 -4.57 -1 -101                       imp:p=1            $ Active Fuel Region
102 1 -4.57 -1 101 -102                   imp:p=2            $ Active Fuel Region
103 1 -4.57 -1 102 -103                   imp:p=3            $ Active Fuel Region
104 1 -4.57 -1 103 -104                   imp:p=4            $ Active Fuel Region
105 1 -4.57 -1 104 -105                   imp:p=8            $ Active Fuel Region
106 1 -4.57 -1 105 -106                   imp:p=16           $ Active Fuel Region
107 1 -4.57 -1 106 -107                   imp:p=32           $ Active Fuel Region
108 1 -4.57 -1 107 -108                   imp:p=64           $ Active Fuel Region
109 1 -4.57 -1 108 -109                   imp:p=128          $ Active Fuel Region
110 1 -4.57 -1 109 -110                   imp:p=256          $ Active Fuel Region
2 2 -0.9982 1 -3 #4 -20                   imp:p=256          $ Water Region
4 4 -7.94 2 -3 7 -71                      imp:p=256          $ RPV Shell
5 4 -7.94 5 -6 -7 #7                      imp:p=256          $ Bottom RPV Shell
6 2 -0.9982 -5 -7                         imp:p=256          $ Water Above Fuel
61 2 -0.9982 -20 71 (-10:-8)              imp:p=256          $ Water Above Vessel Head
71 3 -1.21E-03 -42 41                     imp:p=256          $ Void for Primary Loop
7 5 -2.3 6 3 -4 #71                       imp:p=256          $ Concrete Surrounding RPV
8 5 -2.3 8 -9 10                          imp:p=256          $ Concrete above RPV
9 4 -7.94 -11 12 28                       imp:p=256          $ Steam Generator 1
10 0 -12 28                               imp:p=0            $ Inner Steam Generator 1
11 4 -7.94 -13 14 28                      imp:p=256          $ Steam Generator 2
12 0 -14 28                               imp:p=0            $ Inner Steam Generator 2
13 4 -7.94 -15 16 28                      imp:p=256          $ Steam Generator 3
14 0 -16 28                               imp:p=0            $ Inner Steam Generator 3
15 4 -7.94 -17 18 28                      imp:p=256          $ Steam Generator 4
16 0 -18 28                               imp:p=0            $ Inner Steam Generator 4
20 4 -7.94 21 -22                         imp:p=256          $ Containment Liner
21 5 -2.3 22 -23                          imp:p=256          $ Containment Wall
22 4 -7.94 24 -25 27                      imp:p=256          $ Containment Dome Liner
23 5 -2.3 25 -26 27                       imp:p=256          $ Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13
        15 17                             imp:p=256          $ 68 foot level
30 3 -1.21E-03 (-24:-21:-8:-10:-2)
        11 13 15 17 20 #8 #24 #2 1        imp:p=256          $ Air in Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10
     #11 #12 #13 #14 #15 #16 #20 #21
     #22 #23 #24 #30 #61                  imp:p=0            $ Problem Boundary
```

Figure 7-7 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (With Head)

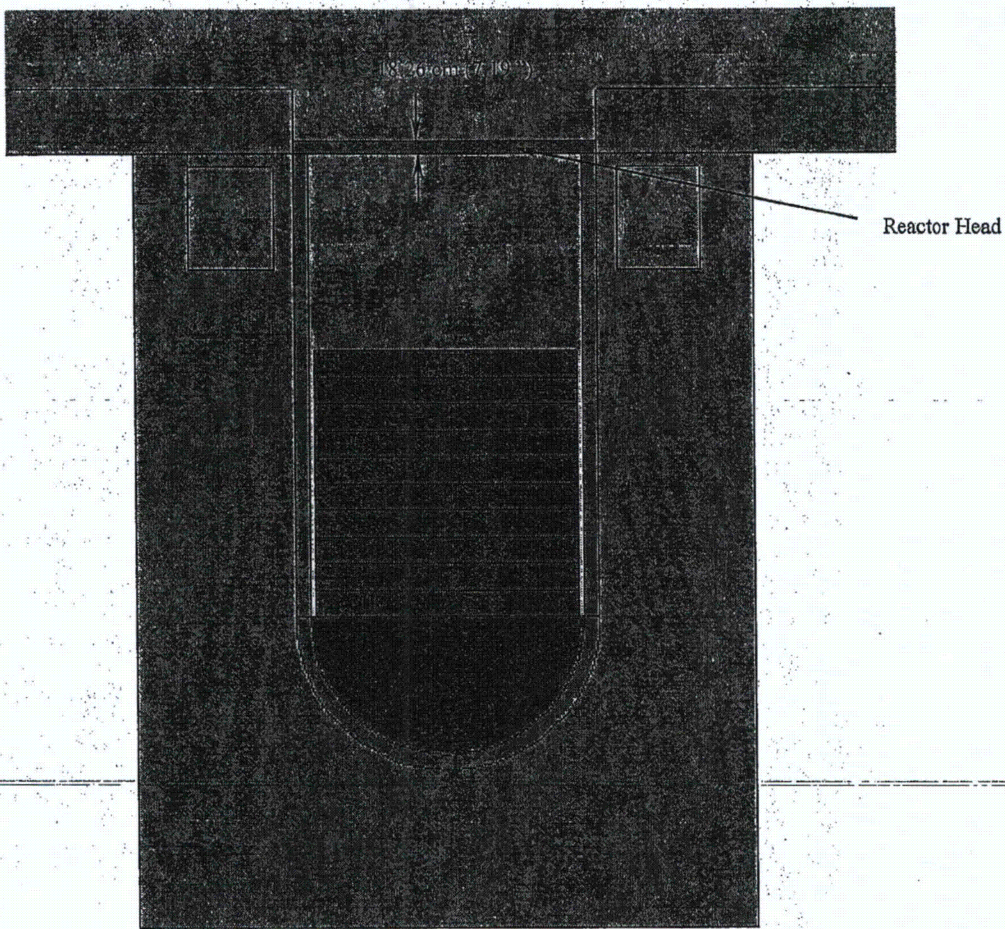


Figure 7-8 MCNP Cell Cards (With Head)

```
c cells
101 1 -4.57 -1 -101                      imp:p=1          $ Active Fuel Region
102 1 -4.57 -1 101 -102                  imp:p=2          $ Active Fuel Region
103 1 -4.57 -1 102 -103                  imp:p=3          $ Active Fuel Region
104 1 -4.57 -1 103 -104                  imp:p=4          $ Active Fuel Region
105 1 -4.57 -1 104 -105                  imp:p=8          $ Active Fuel Region
106 1 -4.57 -1 105 -106                  imp:p=16         $ Active Fuel Region
107 1 -4.57 -1 106 -107                  imp:p=32         $ Active Fuel Region
108 1 -4.57 -1 107 -108                  imp:p=64         $ Active Fuel Region
109 1 -4.57 -1 108 -109                  imp:p=128        $ Active Fuel Region
110 1 -4.57 -1 109 -110                  imp:p=256        $ Active Fuel Region
2 2 -0.9982 1 -3 #4 -20 31               imp:p=256        $ Water Region
4 4 -7.94 2 -3 7 -71                     imp:p=256        $ RPV Shell
5 4 -7.94 5 -6 -7 #7                     imp:p=256        $ Bottom RPV Shell
6 2 -0.9982 -5 -7                        imp:p=256        $ Water Above Fuel
62 6 -7.8212 -31                         imp:p=256        $ Reactor Vessel Head
61 2 -0.9982 -20 71 (-10:-8) 31          imp:p=256        $ Water Above Vessel Head
71 3 -1.21E-03 -42 41                    imp:p=256        $ Void for Primary Loop
7 5 -2.3 6 3 -4 #71                      imp:p=256        $ Concrete Surrounding RPV
8 5 -2.3 8 -9 10                         imp:p=256        $ Concrete above RPV
9 4 -7.94 -11 12 28                      imp:p=256        $ Steam Generator 1
10 0 -12 28                              imp:p=0          $ Inner Steam Generator 1
11 4 -7.94 -13 14 28                     imp:p=256        $ Steam Generator 2
12 0 -14 28                              imp:p=0          $ Inner Steam Generator 2
13 4 -7.94 -15 16 28                     imp:p=256        $ Steam Generator 3
14 0 -16 28                              imp:p=0          $ Inner Steam Generator 3
15 4 -7.94 -17 18 28                     imp:p=256        $ Steam Generator 4
16 0 -18 28                              imp:p=0          $ Inner Steam Generator 4
20 4 -7.94 21 -22                        imp:p=256        $ Containment Liner
21 5 -2.3 22 -23                         imp:p=256        $ Containment Wall
22 4 -7.94 24 -25 27                     imp:p=256        $ Containment Dome Liner
23 5 -2.3 25 -26 27                      imp:p=256        $ Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13
         15 17                           imp:p=256        $ 68 foot level
30 3 -1.21E-03 (-24 : -21 : -8 : -10 : -2)
11 13 15 17 20 31 #8 #24 #2 1            imp:p=256        $ Air inside Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10
     #11 #12 #13 #14 #15 #16 #20 #21
     #22 #23 #24 #30 #61 31              imp:p=0          $ External to Problem
```

7.4 MCNP Source Definition

The core source term is assumed to be uniformly distributed throughout the volume, and has an energy spectra based on the core inventory [2]. Only the gamma source term is taken into account for this evaluation. Because the source term is generated immediately after shutdown, the fuel gamma source term will predominate. Therefore the N-gamma and hardware activation source terms can be neglected (Assumption 3). The source is defined on the MCNP *sdef* card using distributions to define the particle location and energy. The radius of the core is defined with the *rad* parameter, which automatically creates a uniform distribution based on a cylindrical geometry. The *ext* and *axs* parameters define the direction and distance of the cylinder axis. These parameters combined define the core where the particles can be born. The *erg* parameter defines the energy spectrum of source particles and is based on the results of the ORIGEN-S calculation discussed previously. This distribution is a histogram of energies represented by activities. These are automatically normalized by MCNP to create a probability distribution. The total activity is preserved in the tally multiplier. The MCNP source definition cards are shown below in Figure 7-9. The *sb* card is a source biasing card, which in this case biases the particle generation to the upper end of the core. This is a variance reduction technique to improve the statistical certainty in the results.

Figure 7-9 MCNP Source Definition Cards

```
sdef rad=d1 ext=d2 axs=0 0 1 erg=d8                          ←Source Definition Card
                                                                      -Radius = d1
                                                                      -Extent = d2
                                                                      -Axis = +Z
                                                                      -Energy = d8
si1 209.71                                                   ←Core Radius Distribution
si2 h 0 42.672 85.344 128.016 170.688 213.36 256.032 298.704 ←Core Axial Distribution
      341.376 384.048 426.72
sp2 0 1 1 1 1 1 1 1 1 1 1                                    ←Actual Uniform Distribution
sb2 0 0.001 0.001 0.01 0.01 0.01 0.1 0.1 0.1 1 1             ←Biased to Top Distribution
c  Fuel Gamma Spectra
si8  h 1.000e-002 5.000e-002 1.000e-001 2.000e-001 3.000e-001 4.000e-001  ←Source Energy Groups
     6.000e-001 8.000e-001 1.000e+000 1.330e+000 1.660e+000 2.000e+000
     2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.000e+000
     1.000e+001 1.100e+001
sp8    0.00E+00 9.288E+19 2.926E+19 6.537E+19 4.277E+19 1.521E+19 3.578E+19 ←Source Emission on
                                                                              Energy Basis
       4.352E+19 2.66E+19 1.289E+19 1.649E+19 5.572E+18 5.527E+18 1.984E+18
       7.812E+17 3.48E+16 3.947E+11 1.75E+08 37100000 2009000
```

7.5 MCNP Tally Specification

The tallies used in this evaluation are point detectors placed at approximate locations of radiation monitors RE-8055 and RE-8099. Point detectors are chosen because they use quasi-deterministic dose calculations that will provide better results than surface or cell based tallies that require the particles to enter those regions. The inputs to this card are the coordinates of the dose points followed by an exclusion zone (reduce variance), as well as a multiplier card, which represents the total core activity in photons/sec. The tally cards are shown in Figure 7-10.

Figure 7-10 MCNP Tally Cards

```
f5c RE-8055 and RE-8099                        ←Tally Comment Card
f5:p -1200 -400 1909.24 20                     ←Tally 5 (point detector)
1200  400 1909.24 20                              x      y      z    exclusion
                                                 -1200 -400 1909.24    20
                                                  1200  400 1909.24    20
fm5 3.947E+20                                  ← Tally Multiplier
                                                 (Total Activity)
```

In addition, the flux is multiplied by ANSI/ANS flux-dose conversion factors [11]. This is specified in MCNP using the *de/df* cards. These are shown in Figure 7-11.

Figure 7-11 ANSI/ANS-6.1.1-1977 Gamma Flux to Dose Conversion Factors

```
c -----------------------------------------------------------------
c ANSI/ANS-6.1.1-1977
c Gamma Flux to Dose Conversion Factors
c (mrem/hr)/(photons/cm2-s)
c -----------------------------------------------------------------
de0 .01 .03 .05 .07 .10 .15 .20 .25 .30 .35 .40        ←Energy Bins for Flux
     .45 .50 .55 .60 .65 .70 .80 1. 1.4 1.8 2.2         to Dose Conversion
     2.6 2.8 3.25 3.75 4.25 4.75 5. 5.25 5.75 6.25
     6.75 7.5 9. 11.
df0  3.96E-03  5.82E-04  2.90E-04  2.58E-04  2.83E-04  3.79E-04   ←Energy Dependent
     5.01E-04  6.31E-04  7.59E-04  8.78E-04  9.85E-04  1.08E-03    Flux Multipliers
     1.17E-03  1.27E-03  1.36E-03  1.44E-03  1.52E-03  1.68E-03
     1.98E-03  2.51E-03  2.99E-03  3.42E-03  3.82E-03  4.01E-03
     4.41E-03  4.83E-03  5.23E-03  5.60E-03  5.80E-03  6.01E-03
     6.37E-03  6.74E-03  7.11E-03  7.66E-03  8.77E-03  1.03E-02
```

7.6 MCNP Material Cards

The MCNP material cards are provided in Figure 7-12. These are based on the compositions described in Table 5-4.

Figure 7-12 MCNP Material Cards[5]

```
m1      92235      -0.0245
        92238      -0.5891
        8016       -0.2521
        40000      -0.1118
        50000      -0.0017
        24000      -0.0001
        26000      -0.0002
        1001       -0.0211
        6012       -0.0001
m2 1001 2 8016 1               $ Water
m3     6012 -0.000126          $ Air
       7014 -0.76508
       8016 -0.234793
m4     6000 -0.0008            $ SS 304
       14000 -0.01
       15031 -0.00045
       24000 -0.19
       25055 -0.02
       26000 -0.68375
       28000 -0.095
m5     26000 -0.014            $ Reg-Concrete
       1001 -0.01
       13027 -0.034
       20000 -0.044
       8016 -0.532
       14000 -0.337
       11023 -0.029
m6     6012 -0.01               $ Carbon Steel
       26056 -0.99
```

[5] Material 1 composition will change based on the water level relative to the core. This only applies to water heights below 14 feet.

7.7 Results

File Naming Scheme:

The MCNP input files are named with the following convention:

P-height-condition where:

P = Project (STP)

Height = water height from bottom of core (ft)

Condition = h – with head
n – no head

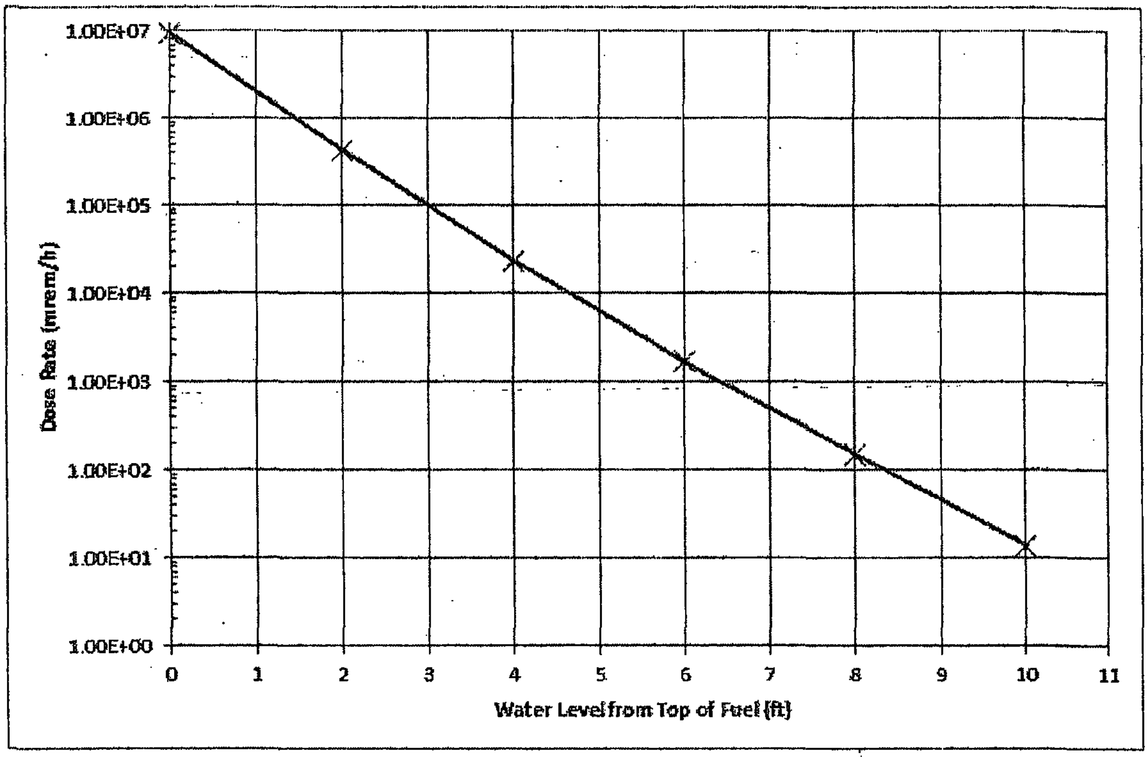
7.7.1 Results without Head

The dose rate as a function of water level is provided in Table 7-3 and plotted in Figure 7-13, below. Because the MCNP model geometry is symmetric in the x and y planes, the two point detector locations should provide the same dose rate. To increase the statistical certainty in the final result, the two individual dose rate responses and uncertainties are combined using inverse variance averaging. All of the water levels described in the following sections refer to the level at the top of the fuel (i.e. 0 foot water level is at the top of the fuel assemblies and ~13 feet is flange level).

Table 7-3 Dose Rate Response as a Function of Water Level for no Head Configuration (mrem/h)

Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	9.27E+06	0.0081	9.34E+06	0.0109	9.30E+06	0.0065
2	4.26E+05	0.0078	4.31E+05	0.0093	4.28E+05	0.0060
4	2.31E+04	0.0236	2.32E+04	0.0247	2.32E+04	0.0171
6	1.73E+03	3.10E-02	1.69E+03	2.44E-02	1.70E+03	0.0192
8	1.51E+02	0.0302	1.51E+02	0.0287	1.51E+02	0.0208
10	1.40E+01	0.036	1.36E+01	0.0323	1.38E+01	0.0240

Figure 7-13 Dose Rate versus Water Height Plot for no Head Configuration



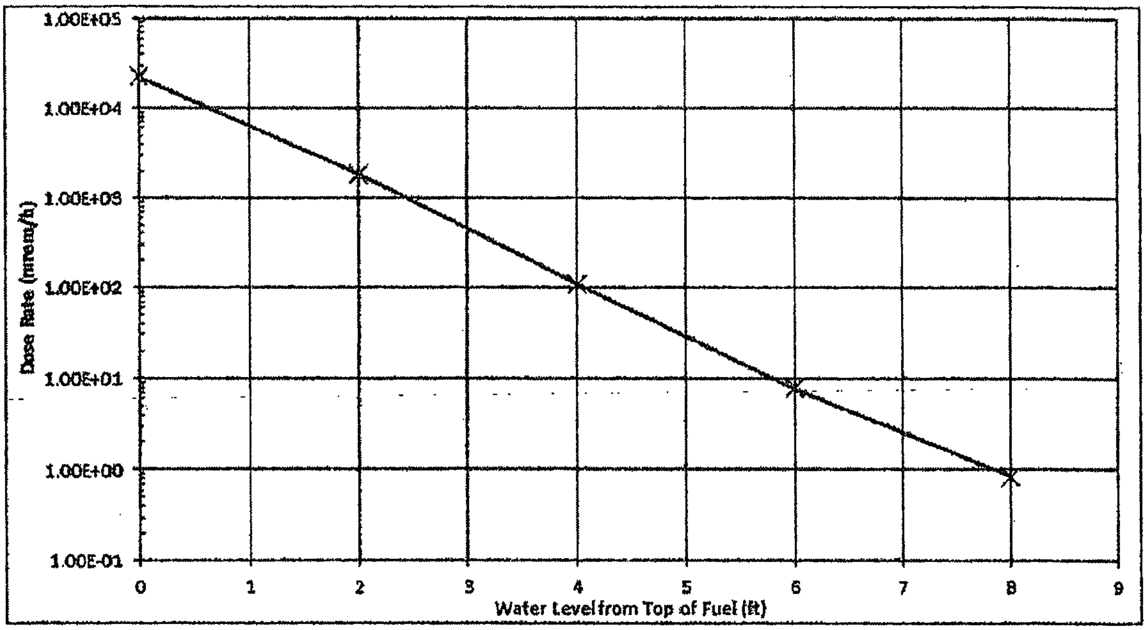
7.7.2 Results with Head

The dose rate results for the cases with the head in place are the same, except the minimum detectable dose rate is lower due to the lower ambient dose rate in the containment. The dose rates are listed in Table 7-4 and plotted in Figure 7-14.

Table 7-4 Dose Rate Response as a Function of Water Level for Head on Configuration (mrem/h)

Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	2.16E+04	0.094	2.56E+04	0.185	2.24E+04	0.0838
2	1.87E+03	0.083	1.83E+03	0.074	1.85E+03	0.0554
4	1.11E+02	0.061	1.08E+02	0.069	1.10E+02	0.0455
6	8.89E+00	0.085	7.48E+00	0.048	7.82E+00	0.0418
8	8.95E-01	0.125	8.12E-01	0.093	8.42E-01	0.0742

Figure 7-14 Dose Rate versus Water Height Plot for with Head Configuration



Appendix A – ENERCON Reference EMAILS

Drew Blackwell

From:	Paul Sudnak
Sent:	Monday, December 09, 2013 9:55 AM
To:	Chad Cramer; Joanne Morris
Cc:	Marvin Morris; Jeff Gromatzky; Michael Falkner; Jay Maisler; Caleb Trainor
Subject:	RE: STP Refueling Cavity Level Calc

Sure, let me find the elevation drawing for the cavity. The water level during refueling is the same water level as the spent fuel pool during fuel transfer. The height of the active fuel is 28'-2 inches. The vessel flange level is 39'-3", and mid-loop is 32'-3".

RCB radiation monitors (RE-8055 and RE-8099) read from <1 mR/hr to 2.5 mR/hr during refueling. If the upper internal package or head are being removed, levels can increase to over ~ 100mR/hr for the upper internals. Levels on the refueling deck (68'0") at mid-loop will only increase to ~ 10 mR/hr with the water level that low. When the head is being de-tensioned by worker on the head level platform (~39' EL), dose rates at that location can read ~ 50 to over 100 mR/hr. The general area dose rates from core radiation is usually less than 100 mR/hr, unless there are lots of fuel leaks or high RCS corrosion and activation products. Dose rates at the monitors at flange level are usually less than 5 mR/hr.

Paul J. Sudnak
Enercon Services, Inc.
12906 Tampa Oaks Boulevard
Suite 131
Temple Terrace, Florida 33637

Office: (813) 962-1800 X603 Fax: (813) 962-1881 Cell: (813) 389-0960

From: Chad Cramer
Sent: Friday, December 06, 2013 2:08 PM
To: Joanne Morris
Cc: Marvin Morris; Jeff Gromatzky; Paul Sudnak; Michael Falkner
Subject: STP Refueling Cavity Level Calc

Joanne,

Michael Falkner has completed the STP SFP calc and sent it to me for review. I spoke with he and Jeff Gromatzky and they indicated that he should have ability over the next week or so to do the refueling cavity level calc.

Paul,

Drew Blackwell

From: Sudnak, Paul <pjsudnak@STPEGS.COM>
Sent: Monday, February 03, 2014 3:44 PM
To: Caleb Trainor; Drew Blackwell
Cc: Domal, Michael; Jay Maisler
Subject: RE: Fuel Assembly Dimension

Thanks Caleb,
Assume the only attenuation is from the materials between the detectors and the core. Disregard reflection. I don't think the SGs are between the core and detector, nor is the containment wall. Concrete should be high density. Atmosphere should be saturated steam at greater than 9.5 psi (containment spray initiation) and less than 56 psi (containment design pressure) mixed with air at the original containment volume at STP. The detectors are ion chambers. Do not include neutrons. The reactor vessel head is around ~8" thick and carbon steel. I will get you the actual drawing of the RPV head thickness, but I think it is from the UFSAR.

Again, a peer check from Mike or Jay?
Paul

From: Caleb Trainor [mailto:ctrainor@enercon.com]
Sent: Monday, February 03, 2014 2:21 PM
To: Sudnak, Paul; Drew Blackwell
Cc: Domal, Michael; Jay Maisler
Subject: RE: Fuel Assembly Dimension

Drew is working on CS1/CG1 where the concern is direct shine from the core due to lowered water levels and no fuel damage assumed. I think you may be thinking of the fission product barrier calcs that I'm working on.

-Caleb

From: Sudnak, Paul <pjsudnak@STPEGS.COM>
Sent: Monday, February 3, 2014 3:05 PM
To: Caleb Trainor; Drew Blackwell
Cc: Domal, Michael; Jay Maisler
Subject: RE: Fuel Assembly Dimension

I think Caleb is right here. Once the concentration is known, the detectors are going to respond to the gases primarily above the 68' Elevation, all the rest will be significantly attenuated by the concrete floors, inner and outer Bio-shield wall, steam generators, and the pressurizer. To model all of those structures would require an extensive geometry and a considerable amount of data. Our intent here is to identify the concentration of gases above the 68' El and determine the monitor response. Disregard the Steam Generators, the inner and outer bio-shield walls, and the Pressurizer. With an assumed homogenous mix based on 20% fuel damage, the dose rates should be significant. Factoring in additional structures and elevations will not significantly change the outcome. A General Emergency will be declared.

Earlier today, I sent the location of the containment high range monitors (73' Elevation). Assume they see the volume of the reactor containment building above the 68' El.

Mike/Jay, can you give me a peer check here?
Paul

Appendix B – Electronic File Listing

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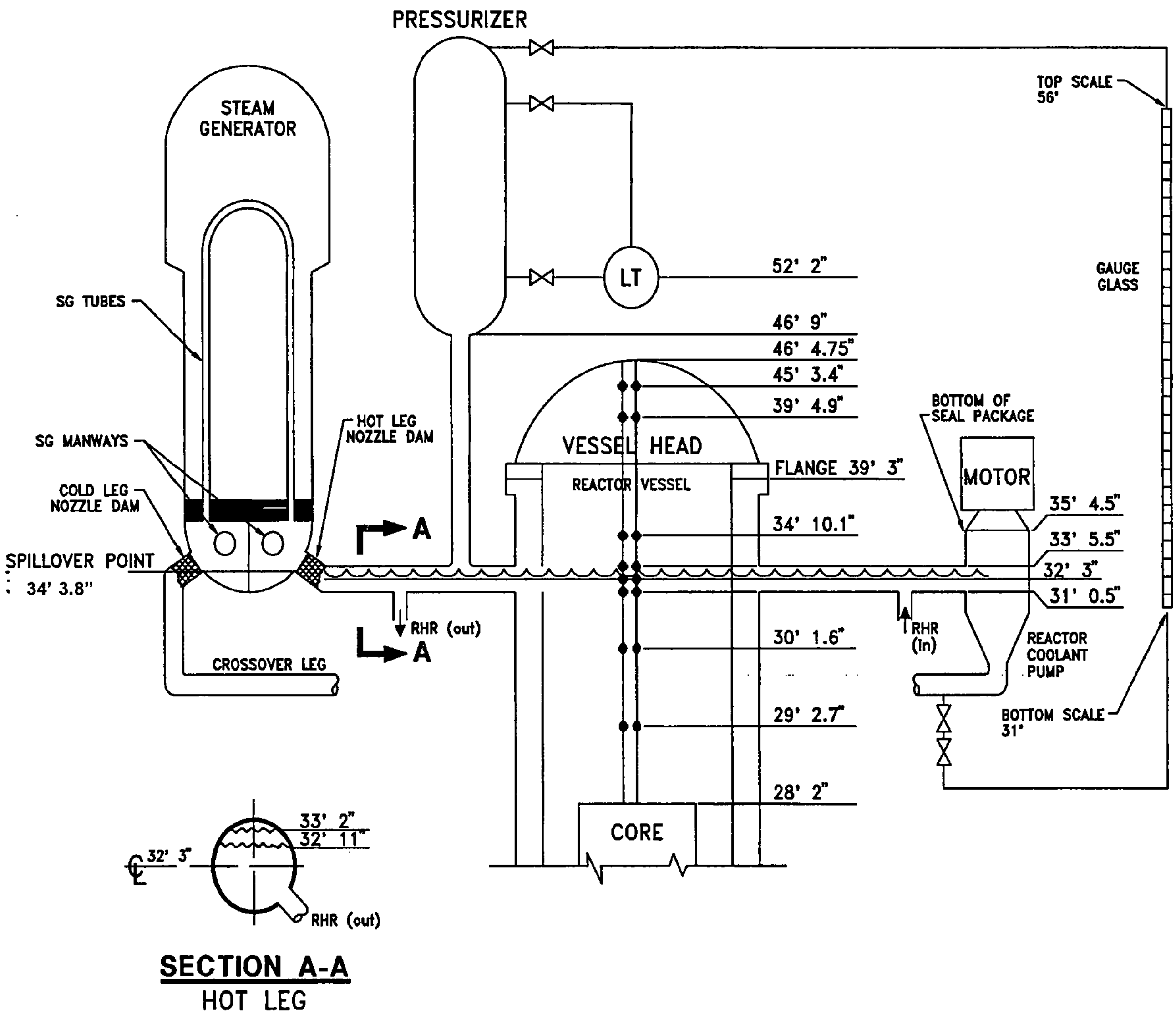
REACTOR COOLANT SYSTEM



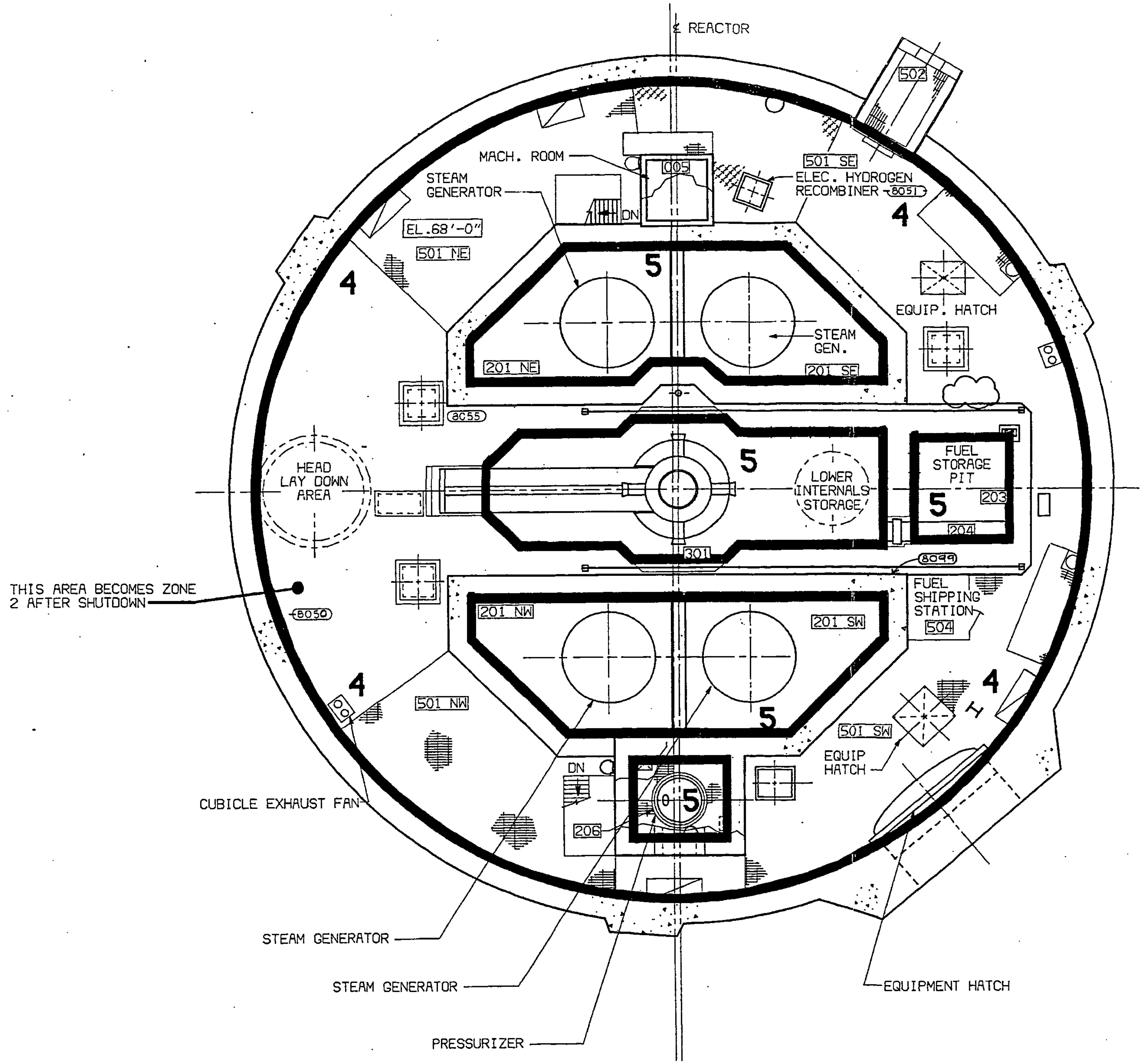
SECTION A-A
HOT LEG

STP D-0794
Rev 2

NOTE

- Top of Core is elevation 28 ft 2 inches.
- SG spillover is elevation 34 ft 3.8 inches.

SENSOR UNCOVERED	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
All Covered	100	100	46' 4.75"	Upper Head Full
1	64	100	45' 3.4"	Upper Head Partially Drained
2	0	100	39' 4.9"	Plenum Full
3	0	85	34' 10.1"	Plenum **NOT** Full (Enter Reduced Inventory)
4	0	66	33' 5.5"	Top of Hot Leg Nozzle
5	0	48	32' 3"	Hot Leg Centerline
6	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
7	0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
8	0	0	29' 2.7"	Upper Core Plate



RAD. MONITOR

RAD. MONITOR-SAFETY RELATED

PLANT RADIATION SHIELDING ZONES			
ZONE NUMBER	MAX. DOSE RATE (MREM/HR.)	POSTING REQUIRED	ANTICIPATED ACCESS
1	≤ 0.5	NO	UNRESTRICTED ACCESS
2	≤ 2.5	NO	CONTROLLED, 40HR/WK PERMISSIBLE
3	≤ 15	YES	CONTROLLED ACCESSIBLE ON A PERIODIC BASIS
4	≤ 100	YES	CONTROLLED LIMITED ACCESS
5	> 100	YES	NORMALLY INACCESSIBLE

NO.	ISSUE DATE	REVISION	BY	CKR	RE	DV	NA	SE	PE
3	7-1-97	EDITORIAL CHANGE TO REMOVE EXTRA RAD. MONITOR #8099. (F-4) REF. CR 96-15540-4.	VJM			—	—		—
0-2	6-17-83	ISSUED FOR CONSTRUCTION.	SIGNATURES ON FILE						

EDWG. DGN

HOUSTON LIGHTING & POWER COMPANY
SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION

RADIATION ZONES
REACTOR CONTAINMENT BUILDING
PLAN AT EL. 68' -0"

SCALE: 1/8" = 1' | DWG. NO. 9C129A81105 | REV. 3

TABLE 12.3.4-1

AREA RADIATION MONITORS

Reactor Containment Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8052 Incore Instrumentation Room (-1ft-6 in.)	10^{-1}-10^{4}	1,000
N1RA-RE-8053 Support across from elevator (-11 ft-3 in.)	10^{-1}-10^{4}	100
N1RA-RE-8054 West Stair Landing (19 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8055 North SG wall across from the head laydown area (68 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8056 Support across from elevator (52 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8099 South SG wall across from the in-containment fuel pool (68 ft-0 in.)	10^{-1}-10^{4}	100

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8081 ~11 ft S of cols. 30.2 and S_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8084 ~24 ft S of cols. 28 and T_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8085 ~24 ft S of col. 28 and ~6 ft E of col. S_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8086 ~24 ft S of col. 28 and ~11 ft E of col. R_1 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8087 col. 30.2 and 12 ft W of col. R_1 (4 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8088 3 ft S of col. 30.9 and col. R_1 (30 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8089 col. 28 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8090 18 ft N of col. 30.2 and col. T_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8091 col. 34 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building (Continued)

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8097 33 ft S of cols. 28 and 10 ft W of col. N (68 ft-0 in.)	10^{-2}-10^{7}	1,000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building

Tag Number and Location (1)	Range (mR/hr) (3)	High Alarm Setpoint (mr/hr) (2)
N1RA-RE-8057 col. 22 and ~10 ft E of col. J (10 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8058 col. 26 and col. J (10 ft-0 in.)	10^{-1}-10^{4}	2.5
NIRA-RE-8059 col. 27 and col G (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8060 ~10 ft S of col. 30 and col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8061 ~10 ft S of col. 24 and ~11 ft W of col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8062 ~6 ft S of col. 31 and col. C (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8063 ~9 ft S of col. 28 and col. B (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8064 ~12 ft S of col. 24 and col. F (29 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8065 ~5 ft N of col. 32 and col. C (29 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8066 ~4 ft N of col. 22 and 14 ft E of col. C (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8067 col. 22 and 10 ft E of col. J (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8068 ~10 ft N of col. 25 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N2RA-RE-8068 ~10 ft S of col. 24 and col. G (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8069 ~12 ft S of col. 24 and ~14 ft E of col. C (41 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8070 col. 29 and col. C (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8071 ~18 ft S of col. 28 and 3 ft W of col. B (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8072 ~11 ft N of col. 29 and 5 ft W of col. D (41 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8073 col. 29 and col. E (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8074 ~5 ft S of col. 31 and ~7 ft W of col. C (41 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8075 col. 28 and ~3 ft W of col. G (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8076 col. 22 and ~10 ft E of col. J (60 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8077 col. 27 and col. J (60 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8078 col. 27 and col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8079 col. 25 and ~2 ft W of col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8080 col. 26 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8082 col. 28 and ~8 ft E of col. H (69 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8083 ~10 ft S of col. 29 and 15 ft W of col. E (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8098 ~6 ft N of col. 25 and col. H (60 ft-0 in.)	10^{2}-10^{7}	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Miscellaneous Buildings

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8092 col. 9 and col. P TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8093 col. 7 and col. M TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8094 ~3 ft N of col. 23 and ~14 ft W of col. B TSC-MEAB (72 ft-0 in.)	10^{-2}-10^{7}	1000

CN-2963

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.
2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Post-Accident Monitors

Tag Number and Location [1]	Range (R/hr)	High Alarm Setpoint (R/hr) [2]
A1RA-RE-8050 RCB (68 ft-0 in.)	10^0-10^8	2000
C1RA-RE-8051 RCB (68 ft-0 in.)	10^0-10^8	2000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Containment H2 concentration should be continuously monitored following a LOCA, to avoid explosive H2 concentrations.

___12 **MONITOR Containment H2 Concentration:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___a. Containment H2 - GREATER THAN OR EQUAL TO ZERO (QDPS QUAL PAMS)	a. PLACE containment H2 monitoring system in service per ADDENDUM 1, ESTABLISHING CONTAINMENT H2 MONITORING.
___b. H2 concentration - GREATER THAN OR EQUAL TO 0.5%	b. PERFORM the following: 1) <u>WHEN</u> H2 concentration is GREATER THAN 0.5%, <u>THEN</u> PERFORM Step 12.c and 12.d. 2) GO TO Step 13.
___c. H2 concentration - LESS THAN 4% BY VOLUME	c. PERFORM the following: 1) CONSULT TSC staff for additional recovery actions. 2) GO TO Step 13.
___d. PLACE hydrogen recombiners in service per 0POP02-CG-0001, ELECTRIC HYDROGEN RECOMBINERS	

STEP DESCRIPTION FOR 0POP04-RC-0003 STEP 3.0

STEP: CHECK Trends For Any Of The Following Indications Of RCS Leakage:

- Rad Monitor RT8011 Particulate – Rising
- Reactor Coolant Drain Tank Level – Rising
- Pressurizer Relief Tank Level – Rising
- RCB Normal Sump Level – Rising

PURPOSE: To determine if leakage is from RCS and not CVCS.

BASIS: Indication of RT8011, RCDT, PRT or RCB Normal Sump levels rising will confirm that the leakage is from RCS and not CVCS which is normally tied to the RCS.

ACTIONS: Monitor trends from RT8011, RCDT, PRT or RCB Normal Sump.

INSTRUMENTATION: Level indications located on CP004 and various plant computer monitors located in control room. Radiation Monitor Computer RM-11.

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A

EHU1

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

.e U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1032	June 13, 2011	June 12, 2051	72-1032	0		USA/72-1032

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International
Final Safety Analysis Report for the
HI-STORM FW MPC Storage System

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached ·endix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), and the conditions .ecified below:

APPROVED SPENT FUEL STORAGE CASK

Model No.: HI-STORM FW MPC Storage System

DESCRIPTION:

The HI-STORM FW MPC Storage System consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) a storage overpack (HI-STORM FW), which contains the MPC during storage; and (3) a transfer cask (HI-TRAC VW), which contains the MPC during loading, unloading and transfer operations. The MPC stores up to 37 pressurized water reactor fuel assemblies or up to 89 boiling water reactor fuel assemblies.

The HI-STORM FW MPC Storage System is certified as described in the Final Safety Analysis Report (FSAR) and in the U. S. Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) accompanying the Certificate of Compliance (CoC).

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. All MPC components that may come into contact with spent fuel pool water or the ambient environment are made entirely of stainless steel or passivated aluminum/aluminum alloys. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. All confinement boundary components are made entirely of stainless steel. The honeycombed basket provides criticality control.

DESCRIPTION (continued)

There are two types of MPCs: the MPC-37 and MPC-89. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. Both MPC models have the same external diameter.

The HI-TRAC VW transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the cask loading area to the storage overpack. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a neutron shield jacket attached to the exterior and a retractable bottom lid used during transfer operations.

The HI-STORM FW storage overpack provides shielding and structural protection of the MPC during storage. The overpack is a heavy-walled steel and concrete, cylindrical vessel. Its side wall consists of plain (un-reinforced) concrete that is enclosed between inner and outer carbon steel shells. The overpack has air inlets at the bottom and air outlets at the top to allow air to circulate naturally through the cavity to cool the stored MPC. The inner shell has supports attached to its interior surface to guide the MPC during insertion and removal and provide a means to protect the MPC confinement boundary against impactive or impulsive loadings. A loaded MPC is stored within the HI-STORM FW storage overpack in a vertical orientation.

CONDITIONS

1. OPERATING PROCEDURES

Written operating procedures shall be prepared for handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 9 of the FSAR.

2. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written acceptance tests and a maintenance program shall be prepared consistent with the technical basis described in Chapter 10 of the FSAR. At completion of welding the MPC shell to baseplate, an MPC confinement weld helium leak test shall be performed using a helium mass spectrometer. The confinement boundary welds leakage rate test shall be performed in accordance with ANSI N14.5 to "leaktight" criterion. If a leakage rate exceeding the acceptance criteria is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

3. QUALITY ASSURANCE

Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important-to-safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the storage system

4. HEAVY LOADS REQUIREMENTS

Each lift of an MPC, a HI-TRAC VW transfer cask, or any HI-STORM FW overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific review of the heavy load handling procedures (under 10 CFR 50.59 or 10 CFR 72.48, as applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.2 of Appendix A.

5. APPROVED CONTENTS

Contents of the HI-STORM FW MPC Storage System must meet the fuel specifications given in Appendix B to this certificate.

6. DESIGN FEATURES

Features or characteristics for the site or system must be in accordance with Appendix B to this certificate.

7. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

8. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

The air mass flow rate through the cask system will be determined by direct measurements of air velocity in the overpack cooling passages for the first HI-STORM FW MPC Cask System placed into service by any user with a heat load equal to or greater than 30 kW. The velocity will be measured in the annulus formed between the MPC shell and the overpack inner shell. An analysis shall be performed that demonstrates the measurements validate the analytic methods and thermal performance predicted by the licensing-basis thermal models in Chapter 4 of the FSAR.

A letter report summarizing the results of the thermal validation test and analysis shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy this requirement by referencing a validation test report submitted to the NRC by another cask user.

9. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM FW MPC Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

a. Moving the MPC and the transfer cask into the spent fuel pool or cask loading pool.

b. Preparation of the HI-STORM FW MPC Storage System for fuel loading.

c. Selection and verification of specific fuel assemblies to ensure type conformance.

d. Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.

e. Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool or cask loading pool.

f. MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. (A mockup may be used for this dry-run exercise.)

g. Transfer of the MPC from the transfer cask to the overpack.

h. Placement of the HI-STORM FW MPC Storage System at the ISFSI.

i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

Any of the above steps can be omitted if they have already been successfully carried out at a site to load a HI-STORM 100 System (USNRC Docket 72-1014).

10. AUTHORIZATION

The HI-STORM FW MPC Storage System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, this certificate, and the attached Appendices A and B. The HI-STORM FW MPC Storage System may be fabricated and used in accordance with any approved amendment to CoC No. 1032 listed in 10 CFR 72.214. Each of the licensed HI-STORM FW MPC Storage System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility. The HI-STORM FW MPC Storage System may be installed on an ISFSI pad with the HI-STORM 100 Cask System (USNRC Docket 72-1014) provided an assessment is performed by the CoC holder that demonstrates design compatibility.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Kimberly J. Hardin, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards
Washington, DC 20555

Dated July 14, 2011

Attachments:
1. Appendix A
2. Appendix B

CERTIFICATE OF COMPLIANCE NO. 1032

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STORM FW MPC STORAGE SYSTEM

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program

5.3.1 Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.

5.3.2 As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.

5.3.3 Based on the analysis performed pursuant to Section 5.3.2, the licensee shall establish individual cask surface dose rate limits for the TRANSFER CASK and the OVERPACK to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:

a. The top of the OVERPACK.
b. The side OVERPACK
c. The side of the TRANSFER CASK
d. The inlet and outlet ducts on the OVERPACK

5.3.4 Notwithstanding the limits established in Section 5.3.3, the measured dose rates on a loaded OVERPACK or TRANSFER CASK shall not exceed the following values:

a. 30 mrem/hr (gamma + neutron) on the top of the OVERPACK
b. 300 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts
c. 3500 mrem/hr (gamma + neutron) on the side of the TRANSFER CASK

5.3.5 The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates as described in Section 5.3.8 for comparison against the limits established in Section 5.3.3 or Section 5.3.4, whichever are lower.

HU1

1.0 Purpose and Scope

1.1 This procedure specifies the actions to be completed by the Acting Security Manager during a declared emergency.

1.2 This procedure implements the necessary Security emergency response actions for an Unusual Event and for initial immediate response for higher emergency classifications until relieved by the Security Manager.

1.3 This procedure implements the requirements of the South Texas Project Electric Generating Station (STPEGS) Emergency Plan specific to the Acting Security Manager.

2.0 Responsibilities

2.1 The Security Force Supervisor assumes the responsibilities of the Acting Security Manager until relieved. Those responsibilities include:

2.1.1 Directing the implementation of on-site security emergency response activities.

2.1.2 Implementing assembly and accountability efforts.

2.1.3 Assisting with Protected and Owner Controlled Area evacuation.

2.1.4 Establishing special access controls.

2.1.5 Providing for the expedient entry/exit of emergency vehicles.

2.1.6 Directing changes to security operations based on radiological conditions.

2.1.7 Determining level of compliance with current security procedures.

3.0 Precautions and Limitations

3.1 0ERP01-ZV-IN04, Assembly and Accountability are required at a Site Area Emergency Classification or greater unless to do so would put site personnel at risk. The Emergency Director at anytime as dictated by conditions may order assembly and Accountability.

3.2 0ERP01-ZV-IN05, Site Evacuation is required at a Site Area Emergency Classification or greater unless to do so would put site personnel at risk. The Emergency Director at anytime as dictated by conditions may order site Evacuation.

4.0 References

4.1 STPEGS Emergency Plan

4.2 0ERP01-ZV-IN03, Emergency Response Organization Notification

4.3 0ERP01-ZV-IN04, Assembly and Accountability

4.4 0ERP01-ZV-IN05, Site Evacuation

4.5 0ERP01-ZV-RE02, Documentation

4.6 0POP04-ZO-0007, Aircraft Crash Onsite

4.7 0PGP05-ZV-0004, Emergency Plan Implementing Procedure Users Guide

4.8 Security Instruction SI 2202, Owner Controlled Area Vehicle Patrol

5.0 Procedure

5.1 IF an Unusual Event or higher emergency classification is declared, implement Data Sheet 1, Acting Security Manager Checklist. Use Checklist to help direct emergency activities.

5.2 IF contacted by the Security Manager, provide a briefing of the current situation and the security activities underway using Data Sheet 2, Security Briefing Checklist.

5.3 WHEN responsibilities have been transferred to the Security Manager, THEN return to the implementation of Security procedures and discontinue the use of this procedure.

5.4 During an Alert or higher classification, ensure an ERO Qualified EMT is onsite.

6.0 Support Documents

6.1 Form 1, Medical Emergency Information Data

6.2 Data Sheet 1, Acting Security Manager Checklist

6.3 Data Sheet 2, Security Briefing Checklist

3.2 0ERP01-ZV-IN05, Site Evacuation is required at a Site Area Emergency Classification or greater unless to do so would put site personnel at risk. Site Evacuation may be ordered by the Emergency Director at anytime as dictated by conditions.

3.3 The Technical Support Center is activated at an Alert Emergency or higher classification in accordance with Procedure 0ERP01-ZV-IN01, Emergency Classification.

3.3.1 The Emergency Director has ordered the activation of the Technical Support Center to support response activities.

4.0 References

4.1 STP Emergency Plan

4.2 0ERP01-ZV-IN01, Emergency Classification

4.3 0ERP01-ZV-IN03, Emergency Response Organization Notification

4.4 0ERP01-ZV-IN04, Assembly and Accountability

4.5 0ERP01-ZV-IN05, Site Evacuation

4.6 0ERP01-ZV-SH03, Acting Security Manager

4.7 0ERP01-ZV-RE01, Recovery Operations

4.8 0ERP01-ZV-RE02, Documentation

4.9 0PGP05-ZV-0004, Emergency Plan Implementing Procedure Users Guide

4.10 0POP04-ZO-0007, Aircraft Crash Onsite

4.11 Security Instruction 2203, Owner Controlled Area Checkpoints

4.12 NRC Regulatory Issue Summary 2009-10, Communications Between the NRC and Reactor Licensees During Emergencies and Significant Events.

5.0 Procedure

5.1 At an Alert or higher Emergency Classification or as directed by the Emergency Director report to the affected Unit's Technical Support Center and implement Data Sheet 1, Step 1.0 Initial Activities.

5.2 Complete Checklist activities as follows:

5.2.1 Use the right column to log the time an activity is performed.

HU2

NOTE

- Operational Basis Earthquake (OBE) is defined as "That earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; vibratory ground motion for which those features of the nuclear power plant necessary to continued operation without undue risk to the health and safety of the public are designed to remain functional (10CFR100Appendix A)."
- Safe Shutdown Earthquake (SSE) is defined as "That earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain safety-related structures, systems, and components are designed to remain functional (10CFR100 Appendix A)." (From USFAR Section 2.5.1)
- The station design basis values for an OBE is vibratory ground motion equal to or exceeding a horizontal acceleration of 0.05g, but less than an SSE. The station design basis values for an SSE is vibratory ground motion equal to or exceeding a horizontal acceleration of 0.10g.
- The accelerometer recorded information can be analyzed and displayed using a personal computer and software supplied with the machine. This software will display the measured response spectrum to be compared with the OBE and SSE response spectrum which will determine if the OBE or SSE has been exceeded.

1.0 Purpose

1.1 This procedure provides instructions for determining if a seismic event has occurred, and the appropriate actions to ensure plant safety following an actuation of the Seismic Monitoring System. Instructions are also included for determining if the Operational Basis Earthquake (OBE) or Safe Shutdown Earthquake (SSE) limits have been exceeded.

1.2 This procedure is applicable in all modes.

2.0 Symptoms and Entry Conditions

2.1 **(Unit 1 Only)** "SEISMIC EVENT" alarm. (Lampbox 9M01, Window E-8)

2.2 **(Unit 1 Only)** "SEISMIC TRIGGER" triggers the "SEISMIC EVENT" alarm.

2.2.1 "SEISMIC TRIGGER" – This alarm indicates that an acceleration signal greater than 0.02g in the vertical or horizontal direction has been detected from the RCB Foundation seismic trigger accelerometer (0-SY-XR-0011, - 37 ft RCB Tendon Gallery AZ 295°).

2.3 Physical symptoms of a seismic event have been observed. (e.g., ground motion felt by plant personnel).

Initials

NOTE

Determination of OBE or SSE should be complete within 4 hours.

4.4 DIRECT I&C personnel to retrieve all seismic instrumentation data per 0PSP02-SY-0012, Seismic Monitoring Data Retrieval and System Functional Test. ____

4.5 DIRECT I&C to provide ERO TECH Support I&C Engineers with data. ____

4.6 DIRECT ERO TECH Support I&C Engineers to determine if OBE or SSE was exceeded per 0PSP09-SY-0001, Seismic Monitoring Data Analysis. ____

NOTE

The information from the National Earthquake Information Center is just for confirmation that a Seismic Event or strong explosion happened; not to be used to determine OBE or SSE unless on-site instrumentation not available.

4.7 CONTACT the National Earthquake Information Center in Denver Colorado at phone number (303) 273-8500 (voice) or (303) 273-8516 (tape) for confirmation that a seismic event has taken place. ____

NOTE

- Modified Mercalli Intensity VI as defined by the USGS: Felt by nearly everyone; many awakened. Some dishes, windows broken. Unstable objects overturned. Pendulum clocks may stop.
- National Earthquake Information Center will list magnitude of earthquakes and distance from nearby cities in km.

4.8 IF Seismic Monitoring Data can NOT be obtained in Step 4.6, THEN any one of the following can be used as confirmation of exceeding an OBE: ____

- The earthquake resulted in Modified Mercalli Intensity VI or greater within 5 km (3.1miles) of the plant,
- The earthquake was felt within the plant and was of magnitude 6.0 or greater,
- The earthquake was of magnitude 5.0 or greater and occurred within 200 km (124.3 miles) of the plant.

HU3

3.4 WATER LEVEL (FLOOD) DESIGN

The methods of analysis used to determine the design basis flood are discussed in Section 2.4. These methods are consistent with the requirements of Regulatory Guide (RG) 1.59.

The protection measures used to accommodate static and dynamic flood loads on Category I structures generally fall under the category of "incorporated barriers" as specified in regulatory position C.1 of RG 1.102.

3.4.1 Flood Protection

3.4.1.1 External Flood Protection Measures for Seismic Category I Structures. The flooding due to a postulated Main Cooling Reservoir (MCR) embankment breach produces the maximum water level around the power block structures as well as the controlling water elevations for buoyancy calculations. This is also the controlling phenomena in determining the maximum water level at the Essential Cooling Water Intake Structure (ECWIS). Studies and analyses on the MCR embankment have demonstrated that an adequate margin of safety can be maintained for all credible failure mechanisms (Section 2.5.6). Accordingly, mechanistic effects (such as scour and erosion) associated with a postulated failure of the MCR embankment need not be evaluated.

The maximum water level on a vertical face at the south end of the plant structures is El. 50.8 ft mean sea level (MSL), which is El. 22.8 ft above plant grade. This maximum elevation occurs during a quasi-steady-state condition after a breach of the MCR embankment and is based on an instantaneous removal of approximately 2,000 ft of the embankment opposite the power block structures. This maximum elevation occurs on the south face of the Fuel-Handling Building (FHB) of Unit 1. The selection of postulated embankment breach widths and the assumptions made in determining the maximum flood elevations are described in Section 2.4.4.

Total inundation of the Essential Cooling Pond (ECP) occurs only under the condition of MCR embankment breach and does not affect the safe shutdown capability of the plant. The maximum water level calculated to occur at the ECWIS is El. 40.8 ft.

Safety-related structures, systems and components listed in Table 3.2.A-1 are protected against the effects of external flooding by:

1. Being designed to withstand the maximum flood level and associated effects and remain functional (such as seismic Category I structures and the Category I auxiliary feedwater storage tank) or

2. Being housed within seismic Category I structures which are designed as in item 1, above.

Flood protection of safety-related structures, systems, and components is provided for postulated flood levels and conditions described in Section 2.4.

Seismic Category I structures are designed to withstand the maximum flood levels by:

1. Having external walls and slabs of structures designed to resist the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady-state water level.

2. Ensuring the overall stability of the total structure against overturning and sliding due to the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady state water level, and

3. Ensuring that the total structure will not float due to buoyancy forces.

Figure 3.4-1 shows a general section through the plant. Figure 3.4-2 shows the seismic Category I Building maximum steady-state water surface profile, and the corresponding relationship of sill elevations for entrances to seismic Category I buildings.

Table 3.4-1 shows the results of hydraulic loading and buoyancy calculations which were done for the various safety-related facilities. The water depths shown on this table were developed from the maximum water surface elevations presented in Table 2.4.4-3.

An investigation of seismic Category I structures has been made for the flood levels and associated effects as previously described. The design for gross effects upon the structure incorporates safety factors greater than 1.1. All exterior seismic Category I building openings are located above the maximum steady-state flood level or are equipped with watertight doors when located below this profile, except as stated below.

Exceptions to the above-stated design basis for exterior building openings in seismic Category I structures are: (1) the opening for the truck bay in the radwaste loading area of the Mechanical-Electrical Auxiliaries Building (MEAB) and (2) the opening for the rail car access in the spent fuel cask loading area of the FHB. These areas are not protected from flooding because they do not have any safety-related systems and components located near or below the maximum flood level which is required to perform any essential function. In addition, the two areas are separated from the remainder of the building by walls which do not contain openings below the maximum water surface elevation corresponding to their location. The Tendon Gallery Access Shaftoover (TGAS) is provided with a watertight cover to prevent flood waters from entering the MEAB.

The safety-related equipment in the ECWIS is protected from the effects of the design basis flood. The personnel access doors on the west wall are provided with watertight doors; all other doors and openings are above the flood level. The dividing walls and doors between the ECWIS compartments minimize the potential for the propagation of flooding from one compartment to another.

The three maintenance knockout panels in the exterior walls of the Diesel-Generator Building (DGB), which are located below the maximum water surface elevation of 45.0 ft MSL, are watertight and designed for the hydrostatic forces. Each knockout panel allows access to only one of the three separate compartments within the structure, and only one panel may be removed at one time. The dividing walls between the compartments preclude propagation of flooding from one compartment to another.

The maintenance knockout panels in the exterior wall of the room, housing the component cooling water heat exchangers in the MEAB are located below the maximum steady-state water level shown on Figure 3.4-2. These panels are watertight. Since mechanistic effects from the MCR breach need not be evaluated, there is adequate time to replace the knockout panels for the remaining flood events of concern.

All exterior seismic Category I building wall and slab surfaces below grade are waterproofed. This conservatively protects the substructure of seismic Category I buildings from groundwater, which is expected to stabilize between El. 17 ft and 26 ft (1 to 10 ft below grade) after decommissioning of the dewatering system. No waterproofing is provided on exterior wall or slab surfaces above grade to protect against the effects of surge-wave run-up because of its short duration. All construction joints in exterior walls and slabs (except for localized areas of blockouts) are provided with waterstops to the maximum flood level for that location and can withstand hydrostatic and hydrodynamic effects.

All seismic joints between Category I structures contain dual 9-in. water stops capable of withstanding potential seismic and hydrostatic effects. Cracks in concrete are minimized by imposing strict QA and QC procedures on the quality of concrete and construction techniques.

Drains are provided with check valves such that the external flooding would not result in internal flooding through the inadvertent introduction of water through these drains into seismic Category I structures.

The duct banks are sealed so as to prevent backflow into safety-related areas. The cable in the duct banks is designed/specified for submerged installations.

Leakage from groundwater into the FHB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater inleakage occur, it is handled by the pumps in the FHB sump, the three-train compartment sumps, and the transfer cart area sump. For Unit 1 only, accumulated groundwater inleakage to the 64 degree tendon buttress area drains through a penetration in the RCB tendon gallery outer wall and is collected in the tendon gallery sump.

Leakage of groundwater into the MEAB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater leakage occur, it will be collected in sumps. Discharge from non-radioactive sumps are routed to the reservoir via a circulating water discharge line. Potentially radioactive discharge is pumped to the Liquid Waste Processing System (LWPS).

3.4.2 Analysis Procedures

3.4.2.1 Phenomena Considered in Design Load Calculations. For external flooding, the design basis events considered in design load calculations are as described in Section 3.4.1.

3.4.2.2 Flood-Force Application. The design flood conditions and elevations have been determined from an analysis of the phenomena discussed in Section 3.4.1.1.

In order to establish the controlling load conditions resulting from the embankment breach, both instantaneous surge wave runup as well as the longer term, quasi-steady-state conditions were analyzed. The wave runup condition conservatively assumes that the maximum total force perpendicular to the south face of the plant structures includes a dynamic component in addition to the associated hydrostatic forces. The quasi-steady state condition assumes that only the hydrostatic component contributes to the development of the total force for this case. The latter condition resulted in higher water surface elevations and greater hydraulic loads on power block structures.

The vertical buoyant loading condition is the force equal to the weight of water displaced by a structure. The discussion of lateral and vertical loadings is presented in the following subsections. Table 3.4-1 shows a summary of different lateral loadings at various locations around plant and ECP structures, caused by their respective controlling flood conditions. Procedures used to determine flood loadings are identified in Sections 3.4.2.2.1 and 3.4.2.2.2.

3.4.2.2.1 Lateral Loading:

3.4.2.2.1.1 Lateral Loading on the Power Block Structures – The analysis of the lateral force on the power block structures considered both the instantaneous wave runup and the quasi-steady state conditions. This analysis determined that the maximum total lateral force on the power block structures occurs when the maximum water level is reached during the quasi-steady state condition. Table 3.4-1 shows the controlling lateral forces (hydrostatic) exerted on different power block structures. These lateral forces are treated as triangular loadings on a vertical surface, varying at a rate of 62.4 lb/ft2/ft of structure depth. The procedures used to determine the dynamic and hydrostatic loadings for the above analysis conditions are discussed below:

1. Dynamic Force

The dynamic force on the south side of the power block structures is determined by application of linear momentum principles. The flow from the MCR is assumed to be normal to the south side of the power block structures. Therefore, the dynamic force exerted on the structures can be expressed by the following momentum equation (Ref. 3.4-2):

$$F = p\,Q\,V_o$$

where:

F = dynamic force normal to plant structure
p = density of flow
Q = flow rate
V_o = velocity of flow

The maximum value of pQv_o during surge formation is calculated. This is the contribution of momentum flux to the dynamic force. The contribution of the unsteadiness of momentum field is insignificant.

2. Hydrostatic Force