Group B

FOIA/PA NO: 2014-0024

RECORDS BEING RELEASED IN THEIR ENTIRETY

Allen, Don Release

From: Miller, Geoffrey
Sent: Thursday, May 09, 2013 3:45 PM
To: Uselding, Lara
Cc: Allen, Don; Dricks, Victor; Kennedy, Kriss; Scott, Michael; Blount, Tom; Azua, Ray
Subject: ANO AIT Public Meeting Notes
Attachments: ANO AIT Script for Miller.docx; AIT_pres_ANO.ppt

Lara,

Victor asked me to forward you the attached speaking notes from the meeting today. I may not have used these exact words, but I stuck pretty close to the script. I also attached the final version of the slides, though I don't think they changed from the last one I sent out.

There were about 150 people present, including the family of the deceased (and their lawyer). Most of the rest were plant employees, with a couple newspapers and a TV camera (Victor gave an interview with them). The presentation lasted about 25 minutes, and there were only a few questions afterwards. Overall, I was surprised at how short it was – I'm hoping that was because we were effective at getting our message out (guess we'll know when the newspapers come out).

Please let me know if you have questions. I'll be back in the office on Monday.

Thank you,

Geoff

Robase

ANO AIT Script for Miller
SUMMARY

A. Opening Remarks

1. Why we're here
2. Why an AIT warranted
3. What is an AIT?
4. Logistics of the CAT-1 Meeting (agenda, feedback forms, public participation)

B. Summary of Inspection Results

1. SYNOPSIS OF EVENT
2. AIT ACCOMPLISHED ITS PURPOSE
3. REACTOR PLANT SAFETY SYSTEMS RESPONDED AS DESIGNED TO THE LOSS OF OFFSITE POWER AND UNIT 2 REACTOR TRIP
4. LICENSEE TOOK APPROPRIATE ACTIONS TO RECOVER PLANT EQUIPMENT ON UNITS 1 AND 2 AND HAS INITIATED AN EXTENSIVE CAUSE EVALUATION EFFORT
5. NRC RESPONDED PROMPTLY AND CONTINUES TO INSPECT
6. SUMMARY OF INSPECTION AND TEN UNRESOLVED ITEMS
7. FOLLOW-UP INSPECTION TEAM WILL REVIEW THE SIGNIFICANCE OF THESE URIs AND DETERMINE ANY ENFORCEMENT ACTION WARRANTED

C. Questions/Remarks from Arkansas Nuclear One

D. Concluding Remarks by Kennedy

B. Summary of Inspection Results
DETAILS

OPENING REMARKS

GOOD AFTERNOON. MY NAME IS GEOFFREY MILLER. I'M WITH THE NUCLEAR REGULATORY COMMISSION, AND I AM THE TEAM LEAD FOR THE RECENTLY COMPLETED AUGMENTED INSPECTION AT ARKANSAS NUCLEAR ONE. I'D LIKE TO START BY OFFERING SINCERE CONDOLENCES TO THE FAMILY AND FRIENDS OF THOSE WHO INJURED OR KILLED BY THE EVENT ON MARCH 31. WE RECOGNIZE THAT THIS EVENT HAD A SIGNIFICANT EMOTIONAL IMPACT ON THE PLANT AND SURROUNDING COMMUNITY, AND THAT THERE IS UNDERSTANDABLY A GREAT DEAL OF INTEREST IN THE CAUSES THAT LED TO THE EVENT. THE CAUSES OF THE INDUSTRIAL ACCIDENT ARE THE SUBJECT OF AN ONGOING INVESTIGATION BY THE OCCUPATIONAL SAFETY AND HEALTH ADMINISTRATION (OSHA). OUR MEETING TODAY WILL NOT INCLUDE A DISCUSSION OF THE CAUSES. OUR INSPECTION FOCUSED ON THE EFFECTS THE EVENT HAD ON THE NUCLEAR PLANTS AT THE STATION AND THE STEPS TAKEN BY OPERATORS IN RESPONSE TO PROTECT THE PUBLIC HEALTH AND SAFETY.

WE'RE MEETING TODAY WITH ENTERGY OPERATIONS TO PROVIDE A STATUS REPORT OF OUR ONGOING INSPECTION ACTIONS. FOR MEMBERS OF THE PUBLIC WHO ARE IN ATTENDANCE AT THIS MEETING, NRC STAFF WILL BE AVAILABLE TO ANSWER QUESTIONS AND RECEIVE COMMENTS AFTER THE BUSINESS PORTION OF THE MEETING.

WITH ME HERE TODAY . . .. [*INTRODUCE THOSE IN ATTENDANCE INCLUDE VICTOR*]
NOW, MR. BROWNING, WOULD YOU LIKE TO INTRODUCE YOUR STAFF?

ONE OTHER ADMINISTRATIVE ITEM: THERE ARE FEEDBACK FORMS AVAILABLE AT THE BACK TABLE. IN OUR CONTINUING EFFORT TO PROVIDE MORE MEANINGFUL MEETINGS WITH OUR STAKEHOLDERS, WE WOULD APPRECIATE YOU TAKING THE TIME TO COMPLETE ONE OF THE FORMS AND RETURN IT TO US. WE WILL USE YOUR FEEDBACK IN OUR CONTINUING PROCESS TO IMPROVE THE QUALITY OF OUR INTERACTIONS WITH OUR STAKEHOLDERS.

[REVIEW AGENDA]

SUMMARY OF THE INSPECTION RESULTS

1. AIT ACCOMPLISHED ITS PURPOSE

AUGMENTED INSPECTION TEAMS ARE USED BY THE NRC TO REVIEW MORE SIGNIFICANT EVENTS OR ISSUES AT NRC-LICENSED FACILITIES. AN AUGMENTED INSPECTION TEAM IS USED WHEN THE NRC WANTS TO PROMPTLY DIG DEEPLY INTO THE CIRCUMSTANCES SURROUNDING AN OPERATIONAL EVENT TO MAKE SURE THAT ALL OF THE CIRCUMSTANCES THAT CONTRIBUTED TO THIS EVENT ARE WELL UNDERSTOOD IN ORDER TO PREVENT A RECURRENCE.

SINCE THIS EVENT INVOLVED MULTIPLE SYSTEM FAILURES, AND BASED ON OUR ESTIMATE OF THE RISK INCREASE TO THE PLANT CAUSED BY THE EVENT, REGION IV CONCLUDED THAT THE NRC RESPONSE

B. Summary of Inspection Results

DETAILS

SHOULD BE AN AUGMENTED INSPECTION TEAM. THE PURPOSE OF TODAY'S MEETING WILL BE TO PUBLICLY PRESENT THE ITEMS IDENTIFIED BY THE INSPECTION TEAM AS POTENTIAL ISSUES REQUIRING ADDITIONAL FOLLOW UP INSPECTION.

The NRC assigns full-time inspectors, called "resident inspectors," to each operating reactor facility (ID Fred, Abin, William). The resident inspectors conduct daily inspections at ANO and live in the surrounding community. Should an event occur at the plant, the resident inspectors provide immediate response capability for the NRC to assess plant conditions and licensee actions. For this particular event, within one hour of the crane collapse, Fred and Abin were on site monitoring operator actions and the safety of the reactors.

As I mentioned earlier, the purpose of an augmented inspection for NRC to promptly assess more significant events and their causes; to gather the facts and identify issues that may be either performance deficiencies or generic safety issues for the industry. This event resulted in widespread equipment damage, including a loss of offsite power to a unit in a refueling outage and a trip and emergency declaration on the operating unit. Considering the equipment impacts and associated risk to the nuclear plants, an Augmented Inspection Team response was appropriate.

The five-person inspection team consisted of experts in electrical, fire protection and operations, and a risk expert, with decades of experience in their disciplines.

The team spent more than a week on site with additional in-office inspection, conducted interviews and physical inspections in the field, and reviewed system data and event records to independently identify and understand all the issues that would warrant follow-up inspection.

THIS EVENT IS ALSO THE SUBJECT OF AN ONGOING INVESTIGATION BY THE OCCUPATIONAL SAFETY AND HEALTH ADMINISTRATION. BOTH NRC AND OSHA HAVE JURISDICTION OVER OCCUPATIONAL SAFETY AND HEALTH AT NRC-LICENSED FACILITIES. NRC AND OSHA HAVE A MEMORANDUM OF UNDERSTANDING IN PLACE TO ENSURE A COORDINATED AGENCY EFFORT IN THE PROTECTION OF WORKERS AND TO AVOID DUPLICATION OF EFFORT. THE OSHA INVESTIGATION IS STILL ONGOING, WITH THE PRIMARY FOCUS BEING THE SAFETY AND HEALTH OF THE EMPLOYEES AND EMPLOYERS AT THE FACILITY. THE NRC INSPECTION.THAT IS THE SUBJECT OF TODAY'S MEETING FOCUSED ON THE IMPACT OF THE MARCH 31 EVENT ON THE EQUIPMENT AND SAFETY SYSTEMS ASSOCIATED WITH THE TWO NUCLEAR REACTORS AT THE SITE TO ENSURE THE HEALTH AND SAFETY OF THE PUBLIC AND THE ENVIRONMENT REMAINED PROTECTED FROM RADIOLOGICAL HAZARDS.

2. SYNOPSIS OF EVENT

THE EVENT THAT WAS THE SUBJECT OF THIS AUGMENTED INSPECTION OCCURRED ON MARCH 31 WHEN A TEMPORARY LIFTING RIG BEING USED TO MOVE THE GENERATOR STATOR FROM UNIT 1 COLLAPSED, KILLING ONE PERSON AND INJURING EIGHT OTHERS. UNIT 1 WAS IN A REFUELING OUTAGE AT THE TIME AND LOST ELECTRICAL POWER FROM OFFSITE DUE TO DAMAGE CAUSED BY THE

B. Summary of Inspection Results
DETAILS

DROPPED STATOR. UNIT 2 WAS OPERATING AT FULL POWER AND AUTOMATICALLY SHUTDOWN WHEN THE IMPACT OF THE STATOR ON THE TURBINE DECK CAUSED ELECTRICAL BREAKERS TO OPEN, REMOVING POWER FROM ONE OF FOUR OPERATING REACTOR COOLANT PUMPS. WATER FROM A RUPTURED FIRE MAIN LATER CAUSED A SHORT CIRCUIT AND SMALL EXPLOSION INSIDE AN ELECTRICAL BREAKER ON UNIT 2, AND OPERATORS SUBSEQUENTLY DECLARED A NOTICE OF UNUSUAL EVENT (LOWEST OF FOUR EMERGENCY CLASSIFICATIONS), TERMINATING IT AFTER TAKING CORRECTIVE ACTIONS TO STABILIZE THE PLANT'S POWER SUPPLIES

Before we get into the specific details of the issues the team identified, I'd like to make a couple general observations.

3. REACTOR PLANT SAFETY SYSTEMS RESPONDED AS DESIGNED TO THE EVENT

The team determined that after the event occurred, the plant safety systems responded as designed, that all assumptions in the accident analysis appropriately bounded the event, and no unanalyzed condition existed. As such, there was no danger to the public health and safety from radiological hazards.

4. ENTERGY HAS TAKEN APPROPRIATE ACTIONS TO RESTORE PLANT EQUIPMENT AND HAS INITIATED AN EXTENSIVE ROOT CAUSE EFFORT

To date, the Entergy response following the March 31 event appears appropriate. ANO installed temporary modifications to restore offsite power to both units, and implemented compensatory measures for security/fire protection; extensive RCE effort underway. They are treating this event seriously as they determine causes and establish corrective actions. The NRC will conduct additional inspection of the cause evaluation effort and the approach ANO will use in prioritizing and implementing corrective actions. Lots completed, more work to come.

5. SUMMARY OF INSPECTION AND TEN UNRESOLVED ITEMS

The team was chartered by the Region IV Administrator to focus on several specific inspection areas. I'll summarize the results of each inspection area:

1. Chronology of Significant Events.

   We established a detailed Sequence of Events for the dropped stator event through the restoration of offsite power via temporary modifications. We did not identify any issues requiring follow-up in this area

B. Summary of Inspection Results
DETAILS

2. Operator Response.

Multiple challenges: personnel emergency, reactor trip, LOOP, fire water header break, loss of spent fuel pool cooling, breaker fault which led to the declaration of an UE. Operator response appropriately protected the public health and safety.

URI #1: ANO's Control of a Modification Associated with Temporary Fire Pump

- Temporary fire pump installed to augment the fire system during the outage.
- Stator drop ruptured fire system piping in train bay and vicinity, causing significant leakage into the train bay. DD pump started as designed to raise system pressure. Operators shut down the DD pump to stop the leakage, but did not shut down the temporary pump until some time later. Additional inspection to

3. Unit 1 and 2 Equipment Impact.

The team confirmed widespread damage to components within the turbine building [including fire barriers, fire doors, fire penetrations, fire piping, cardox piping, instrument air piping, hydrogen piping, flood barriers, electrical cabinets and buswork, ventilation ducting, structural members.] Licensee assessment of damage is still in progress. A full assessment will not be possible until debris removal activities are completed. Additional follow up inspection as debris removal completed and areas become accessible. *(URI #2: Structural Impact to Units 1 and Unit 2)*

4. Plant Response during the Event.

As I stated earlier, the team concluded the safety-related systems in Units 1 and 2 responded as designed to the loss of offsite power and reactor trip, and that no unanalyzed conditions occurred as a result of this event. The team identified three items for further follow up inspection:

URI #3: Control of Steam Generator Nozzle Dams

The nozzle dams are essentially inflatable plugs that are used to allow access to the inside of the steam generators for inspection during outages. At ANO, air pressure to maintain the dams in place was provided by two separate electric air compressors. During the event, both air compressors lost power when offsite power was lost. Additional follow up inspection needed to review the methods used to provide air pressure to the nozzle dams.

URI #4: Main Feedwater Regulating Valve Maintenance Practices

- MFRV stuck partially open during the last U2 scram due to a maintenance error. During this event, the valve closed, but indicated open due to an indication problem from a separate maintenance error, complicating operator response to the event. Additional follow up inspection to review the valve maintenance. (ref NRC FIN 05000368/2012005, CR-2-2012-1432)

URI #5: Inadequate Flood Barriers
As discussed earlier, a considerable amount of water leaked into the train bay from a broken fire main. The water leaked past flood barriers (gaskets in floor plugs) in the turbine building to the safety related auxiliary building and flowed to the aux building sump. Additional inspection is needed to determine circumstances that allowed water to get from the turbine building into the safety-related auxiliary building.

5. Compensatory Measures.

The team reviewed the adequacy of the licensee's compensatory measures for damaged equipment, including security barriers, support systems (equipment cooling) and fire protection systems. The team concluded the licensee's compensatory measures were appropriate and preserved plant safety. One item identified for further inspection associated with the timeliness of actions to restore water to the fire suppression system: (*URI #5: Compensatory Measures for Fire Water System Rupture*)

6. Event Classification and Reporting.

The team conducted an independent review of the licensee's actions for event classification and reporting. The electrical fault on Unit 2 occurred at 9:23 in the morning, and an entry in the station logs a short time later confirmed water intrusion and the failure of a breaker on the associated electrical bus. Individuals from the field made several reports to the control room over the next hour (though none were logged), and operators declared a Notice of Unusual Event at 10:33 a.m. The Emergency Action Level declaration was based on a verbal report at approximately 10:20 a.m. of damage to the breaker consistent with a small explosion. The team concluded the identified Emergency Action Level (HU-4) was appropriate. However, the team concluded additional inspection was required associated with whether the emergency declaration was timely based on the information available to the control room. *(URI #6: Timeliness of Emergency Action Level Determination)*

7. Heavy Lift Preparations.

The team reviewed the licensee's plans and preparations for the movement of the stator, including their assessment of risk to the plant and identified an issue for further follow up inspection associated with the documentation of plant risk management administrative controls for the move. We identified a second issue for further follow up inspection associated with the evaluation of the vendor supplied crane per the licensee's material handling program. This issue will be examined as part of the licensee's root cause evaluation. NRC follow up inspection will be incorporated with the next charter item

B. Summary of Inspection Results
DETAILS

8. Status of Cause Evaluation Efforts.

The team reviewed the licensee's initial efforts in establishing a cause evaluation team and the beginning of the cause evaluation process. The root cause evaluation is still in progress at this time. We will conduct additional follow up inspection to assess the adequacy of the licensee's identified causes and corrective actions when completed. *(URI #9: Causes and Corrective Actions Associated with March 31, 2013, Dropped Heavy Load Event)*

9. Operating Experience.

The team reviewed the licensee's application of operating experience, with specific focus on control of heavy loads, contractor oversight, and seismic instrumentation. We expect plants to review events from industry and incorporate lessons learned into their processes. The team concluded the licensee had appropriately incorporated the insights from industry operating experience into their corporate programs and implementing procedures. The team did not identify any issues requiring follow-up in this area

6. FOLLOW-UP INSPECTION TEAM

That amounts to ten items requiring follow-up inspection that will be documented in this report as Unresolved Items. The follow-up team will be assembled and dispatched after the details of the causes and corrective actions for these issues are identified. Their job will be to assess the significance of these issues and determine if any enforcement actions are appropriate.

[BROWNING]
[KENNEDY]
Closed – Q&A

Release

U.S.NRC

Protecting People and the Environment

NRC Augmented Inspection Team Exit Meeting

Arkansas Nuclear One

Nuclear Regulatory Commission - Region IV

Russellville, AR

May 9, 2013

Agenda

- Introduction
- Purpose of an AIT
- Event Description
- Inspection Results
- Licensee Response
- NRC Closing Remarks

Augmented Inspection Team

- **Fact-finding inspection**
- **Identify issues for follow up inspection**
- **Identify generic safety concerns in a timely manner**

March 31 Dropped Stator Event

- **Unit 1 RFO Stator Replacement**
- **Structural Failure of Lifting Rig**
- **Loss of Offsite Power to Unit 1**
- **Unit 2 Reactor Trip**
- **Partial Power Loss/Breaker Failure on Unit 2**
- **Notice of Unusual Event**

Inspection Results

- **Reactor plant safety systems responded as designed**
- **Entergy took appropriate actions to recover plant equipment on Units 1 and 2 and has initiated an extensive root cause evaluation effort**
- **NRC responded promptly and continues to inspect**

Summary of Charter Items

- **Event Chronology**
- **Operator Response**
  - **Control of Temporary Modification**
- **Equipment Impact**
  - **Structural Impact to Units 1 and 2**

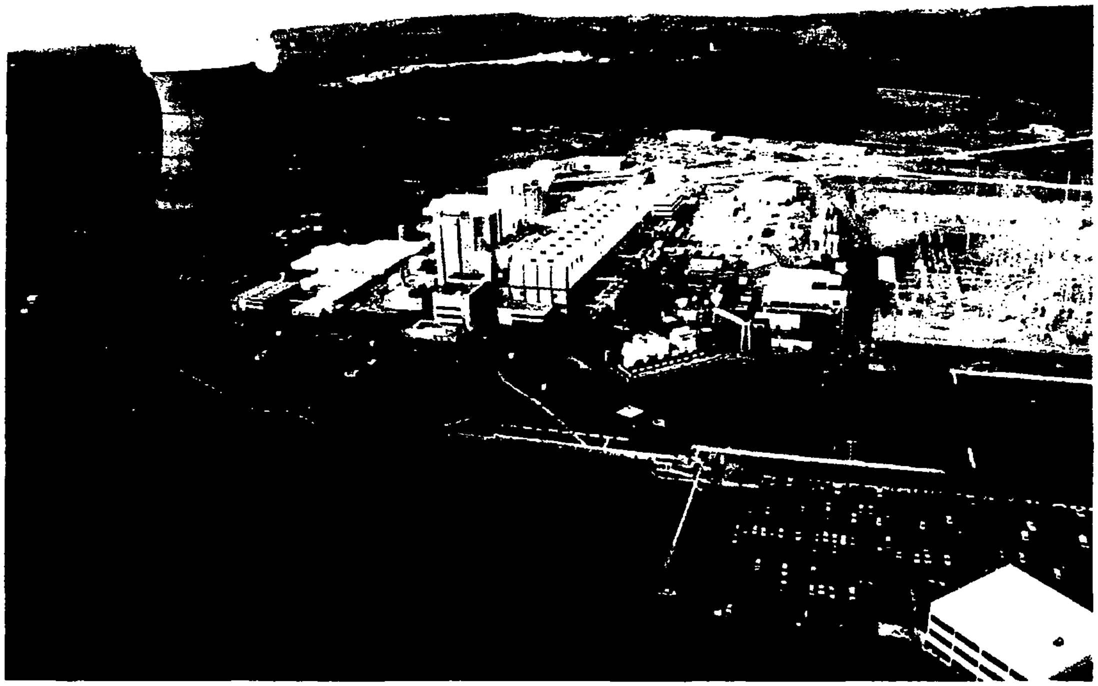
Summary of Charter Items

- **Plant Response to the Event**
  - **Design Control of Steam Generator Nozzle Dams**
  - **Main Feedwater Regulating Valve Maintenance**
  - **Flood Barrier Effectiveness**
- **Compensatory Measures**
  - **Fire Water Compensatory Actions**
- **Event Classification and Reporting**
  - **Timeliness of Emergency Declaration**

Summary of Charter Items

- **Heavy Lift Preparations**
  - Shutdown Reactor Equipment Risk
  - Implementation of Material Control Procedure
- **Status of Cause Evaluation Efforts**
  - Review of Causes/Corrective Actions
- **Operating Experience**
- **Independent Risk Assessment Data**

Licensee Response and Remarks



Contacting the NRC

- Report an emergency
  - (301) 816-5100 (call collect)
- Report a safety concern
  - (800) 695-7403
  - Allegation@nrc.gov
- General information or questions
  - www.nrc.gov
  - Select "What We Do" for Public Affairs

Electronic Distribution

- To receive a summary of this meeting and begin receiving other plant-specific e-mail distributions, subscribe to the Operating Reactor Correspondence electronic distribution via http://www.nrc.gov/public-involve/listserver/plants-by-region.html.

- To discontinue receiving electronic distribution, you may unsubscribe at any time by visiting the same web address above.

Latta, Robert

From: MOSHER, NATALIE B <NMOSHER@entergy.com>
Sent: Thursday, August 22, 2013 9:58 AM
To: Willoughby, Leonard; Melfi, Jim; Latta, Robert
Subject: FW: ANO-1 LER 2013-001-01 Main Generator Stator Temporary Lift Assembly Failure
Attachments: 0CAN081301.pdf

Thought you all would like a copy of the revised LER.

Sent: Thursday, August 22, 2013 8:42 AM
Subject: ANO-1 LER 2013-001-01 Main Generator Stator Temporary Lift Assembly Failure

Outgoing NRC Correspondence

0CAN081301 -- dated 8/22/2013 - Subject: Licensee Event Report 50-313/2013-001-01 - ANO-1 Main Generator Stator Temporary Lift Assembly Failure

Release

Latta, Robert

From: Willoughby, Leonard
Sent: Thursday, September 05, 2013 1:52 PM
To: Loveless, David; Allen, Don
Cc: Latta, Robert
Subject: RE: ONE LAST TIME

I can live with the changes. Let's send it to HQ and tell them we their evalution done and back to us by COB Oct 4, 2013.

From: Loveless, David
Sent: Thursday, September 05, 2013 10:31 AM
To: Willoughby, Leonard; Allen, Don
Cc: Latta, Robert
Subject: ONE LAST TIME

I added a couple more comments I received. Please take one last look.

Thanks,

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

David P. Loveless
Senior Reactor Analyst

(817) 200-1161

Latta, Robert

From: Willoughby, Leonard
Sent: Thursday, September 26, 2013 1:15 PM
To: Melfi, Jim
Cc: Okonkwo, Nnaerika; Latta, Robert
Subject: RE: AIT Followup Report

Jim,

At this time we are waiting on the SRA evaluation. Did you or Dave come to a conclusion on the recently discovered geological features of the area that may affect flooding?

Please send me what you have on the SERP package.

Have fun in Chattanooga.

Leonard

From: Melfi, Jim
Sent: Thursday, September 26, 2013 7:01 AM
To: Willoughby, Leonard
Cc: Okonkwo, Nnaerika; Latta, Robert
Subject: AIT Followup Report

Leonard

FYI, I will be in training the next 2 weeks in Chattanooga. (Oh Joy!!)

I will still be monitoring email, etc., on the ANO Followup report, and will review it when issued.

I have started SERP packages for the issues, but have not gotten very far.

What do you need from me for the report, or what can I assist you with from the region?

JIM

Werner, Greg Release this page

From:	Weerakkody, Sunil
Sent:	Tuesday, November 19, 2013 11:47 AM
To:	Loveless, David
Cc:	Werner, Greg
Subject:	ANO LOOP Cut Set Report - M6-LOOP2 (ET) 2013_11_15 (2).rtf
Attachments:	ANO LOOP Cut Set Report - M6-LOOP2 (ET) 2013_11_15 (2).rtf

Release

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICE OF THE
GENERAL COUNSEL

April 30, 2013

MEMORANDUM FOR: Reginald W. Mitchell
Assistant for Operations, OEDO

FROM: Andrew P. Averbach, Solicitor
Office of the General Counsel

RE: "LITIGATION HOLD" ON ALL MATERIALS PERTAINING TO ACCIDENT RESULTING IN DEATH AT ARKANSAS NUCLEAR ONE

The Office of the General Counsel has been advised that the estate and family of Wade Walters are likely to commence litigation related to the death of Wade Walters following a crane accident at the Arkansas Nuclear One plant on March 31, 2013. We are also advised that a request has been filed pursuant to the Freedom of Information Act for information pertaining to the NRC's investigation of this accident, including any investigation reports, photographs, and inspection reports. Therefore, it is appropriate to implement a "Litigation Hold" on any documents related to the accident.

The implementation of a Litigation Hold requires NRC employees to:

1. Preserve any records related to the accident at Arkansas Nuclear One on March 31, 2013, including any documents generated as part of the investigation of that accident; and

2. E-mail the name and contact information of any staff member likely to have discoverable information to Andrew P. Averbach of OGC.

Preservation Duties:

1. NRC employees should take measures to preserve any materials relating to the subject matter of the contemplated litigation. This obligation includes preserving "electronically stored information" or "ESI." NRC employees must preserve any electronically stored or written material, whether final or in draft form, such as memoranda, e-mails, photographs, maps, diagrams, handwritten notes, databases, letters, presentation materials, recordings, microfilm, scanned photographs or documents. Working files may be kept in place, but must be identified on the inventory and readily retrievable

2. NRC employees may not delete, destroy, overwrite or throw away potentially relevant materials, including any relevant information in personal files, home computers or personal e-mail accounts. Even privileged materials must be preserved because a court may need to review documents to evaluate claims of privilege.

3. Any office identifying its possession of records subject to this Litigation Hold should designate a contact person to facilitate coordination. A list of records inventoried should

be maintained and updated by this contact person. Each office should review inventoried records for any claims of privilege or for the presence of Safeguards information, proprietary information, or classified information.

Staff Likely to Have Materials:

OGC is also requesting that the name and contact information of any staff member likely to have information relevant to the accident or any associated inspection or investigation be emailed to Andrew P. Averbach of OGC.

Please direct any questions – and provide all information – to Andrew Averbach. You may reach Mr. Averbach at 301-415-1956 or andrew.averbach@nrc.gov.

Miller, Geoffrey

From: Scott, Michael
Sent: Thursday, May 09, 2013 4:08 PM
To: Huyck, Doug; Rosales-Cooper, Cindy
Cc: Kennedy, Kriss; Allen, Don; Blount, Tom; Sanchez, Alfred; Azua, Ray; Howell, Art; Lewis, Robert; Hay, Michael; Melfi, Jim; Fairbanks, Abin; Schaup, William; Leeds, Eric; Uhle, Jennifer; Dorman, Dan; McGinty, Tim; Hiland, Patrick; Nieh, Ho; Croteau, Rick; Roberts, Darrell; Reynolds, Steven; Clark, Jeff; Vegel, Anton; Pruett, Troy; Lund, Louise; Evans, Michele; Markley, Michael; Kalyanam, Kaly; Weil, Jenny; Howell, Linda; Miller, Geoffrey
Subject: RE: ANO Weekly Status Report for week ending May 10, 2013
Attachments: ANO Update Week Ending May 10 2013 Rev 1.docx

Revised to add key massages from the AIT exit.

From: Scott, Michael
Sent: Thursday, May 09, 2013 2:17 PM
To: Huyck, Doug; Rosales-Cooper, Cindy
Cc: Kennedy, Kriss; Allen, Don; Blount, Tom; Sanchez, Alfred; Azua, Ray; Howell, Art; Lewis, Robert; Hay, Michael; Melfi, Jim
Subject: ANO Weekly Status Report for week ending May 10, 2013

Subject report attached. Per OEDO, this report is to be provided every other week, so next update will be for week ending May 24, 2013.

Michael L (Mike) Scott
Deputy Director (Acting)
Division of Reactor Projects
Region IV
(817) 200-1462

Arkansas Nuclear One Dropped Stator Event

Week Ending May 10, 2013

Background

At 7:50 a.m. (CDT) on March 31, 2013, while lifting and transferring the Arkansas Nuclear One Unit 1 main generator stator to the train bay, the lift system collapsed, causing the 525-ton stator to fall on and extensively damage portions of the turbine deck, and subsequently to fall over 30 feet into the train bay. At the time of the event, Unit 1 was in a refueling outage. The reactor vessel head had been removed, fuel was in the reactor vessel, and the refueling cavity was flooded up with water level greater than 23 feet above the fuel. Unit 2 was operating at 100% power.

The failure of the lifting device and the dropped stator damaged Unit 1 electrical busses, resulting in a loss of offsite power to Unit 1. Emergency diesel generators started and restored power to the vital busses. On Unit 2, the event caused a reactor coolant pump breaker to open, resulting in a Unit 2 reactor trip from 100% power. Later, due to fire water intrusion into Unit 2 switchgear (the fire main was damaged during the event), offsite power was lost to one of the Unit 2 vital busses due to the failure of a breaker. The associated emergency diesel generator started and restored power to the bus. The licensee declared a Notification of Unusual Event due to the failure (explosion) of the breaker.

Unit 1 Current Status

- Defueled, reactor vessel head removed, RCS drained for outage work
- Core offload completed on April 26.
- Both spent fuel pool cooling pumps are in service
- Stator and all debris have been removed from train bay
- Offsite power is available via a temporary modification from Startup Transformer 1 to vital 4160V and 480V busses.
- Work is ongoing to connect a second temporary modification power line to the vital buses of Unit 1 from Startup Transformer 2.
- Power to non-vital 480V busses is being supplied by a combination of offsite power sources and skid-mounted diesel generators
- Fire main is pressurized with damaged sections isolated. Fire watches are in place as needed.

Unit 2 Current Status

- Mode 1, Power Operations. Plant started up on April 28th.
- The Unit 2 electrical distribution system is operating in its normal technical specification-required mode for reactor power operations. Vital and non-vital busses are energized via the unit auxiliary transformer. Startup Transformer 3 is operable for a fast transfer.
- Alternate AC Diesel Generator (Blackout Diesel) supply bus has been repaired and can now supply Unit 2 if needed
- Emergency diesel generators are in standby.
- Fire main is pressurized with damaged sections isolated. Fire watches are in place as needed.

Licensee Actions

Unit 1

- Restored offsite power to the vital busses on April 6 via a temporary modification from Startup Transformer 1.
- Licensee's root cause team is on site and working on the evaluation.
- The licensee shored up various damaged parts of the Unit 1 turbine building. The licensee has removed the crane debris from the turbine building.
- The licensee is currently removing damaged concrete from the Unit 1 side of the turbine building, and is finalizing a schedule and plan to repair the Unit 1 turbine building floor.

Unit 2

- Two offsite power sources and the alternate AC diesel generator have been restored.
- Post Trip/Transient Report and repair/ resolution of items damaged in the event were reviewed by the Onsite Safety Review Committee on April 19.
- The licensee and the NRC participated in a Unit 2 phone call on April 24 to discuss the licensee's restart plan and related concerns.
- The licensee performed plant walkdowns, inspections, equipment testing, and evaluations to support reactor startup. Items reviewed included structural damage, flood barriers, firefighting strategies, electrical switchgear, and risk.
- The licensee received an outside review of the Unit 2 restart plan from Entergy fleet, INPO, and the Onsite Safety Review Committee.

NRC Actions

- The resident inspectors continue to monitor licensee actions.
- Augmented Inspection Team completed on site inspection and is documenting findings in a report. A public AIT exit meeting was held on May 9. Key messages for the AIT meeting included that the reactor plant safety systems responded as designed, that the licensee's outage planning did not address the stator drop as a credible event, that the licensee took appropriate actions to recover plant equipment, and that the root cause of the event has not yet been determined. The AIT continues to evaluate several unresolved issues.
- Residents, Region, and NRR reviewed the licensee's 50.59 evaluation to verify that the temporary offsite power source satisfies the Unit 1 Technical Specifications requirements prior to defueling the reactor.
- Region IV developed and implemented an inspection plan for oversight of restart of Unit 2. Resident inspectors walked down the firewater, instrument air, hydrogen, carbon dioxide, and electrical switchgear systems. Additionally, inspectors monitored key systems during and after startup and reviewed the post-trip actions and risk assessments for debris removal.
- NRC continues to respond to questions from the media. A press release was issued to announce the beginning of the augmented inspection and another to announce the public exit meeting.

NRC/OSHA Coordination

- NRC staff and OSHA staff continue to coordinate activities and share information.
- Interactions with OSHA are being conducted consistent with guidance provided in Inspection Manual Chapter 1007 and the NRC/OSHA Memorandum of Understanding dated October 21, 1988.

Refer to NRC/DSE/SBPB

Miller, Geoffrey

From: Jones, Steve
Sent: Monday, April 29, 2013 2:51 PM
To: Miller, Geoffrey
Subject: RE: Preservation of Information Related to Death of Wade Walters at ANO

Geoff,

I recommend using "temporary overhead crane" because the definition of "overhead crane" in ASME B30 2 is:

A crane with a multiple-girder movable bridge carrying a movable or fixed hoisting mechanism and traveling on an overhead fixed runway structure.

In this case, the moveable bridge was the trolley on which the stand-jack hoists were mounted as a fixed hoisting mechanism. The overhead runway structure, although temporary, was fixed by attachments to the turbine building structure. A gantry crane is similar to an overhead crane, except the bridge is rigidly supported on legs running on fixed rails. The ANO temporary crane had no legs between the girders and the wheel trucks. The same standard applies to either design (ASME B30.2), but the legs on a gantry crane allow generation of much larger moments at the ends of the bridge. Since ASME B30.2 is cited in OSHA regulations, it would be better to just use overhead crane.

I received the FOIA earlier today. I do have a three procedures and the outage schedule, which I was holding in case there was a need to discuss my input with Region IV management. Is the report in concurrence now?

Steve

From: Miller, Geoffrey
Sent: Monday, April 29, 2013 3:25 PM
To: Jones, Steve
Subject: FW: Preservation of Information Related to Death of Wade Walters at ANO
Importance: High

Steve,

Please see the below.

Thank you for getting me your timely input. We are planning an exit meeting on May 9 (you need not travel out for that).

OSHA is referring to the lift device as a "temporary overhead gantry crane." I know we had a discussion that it was a crane, but not a gantry crane. Would it be incorrect to use this term? Would it be better to go more generic (e.g., 'temporary overhead crane')?

Thanks,

Geoff

From: Tannenbaum, Anita **On Behalf Of** Fuller, Karla
Sent: Monday, April 29, 2013 2:20 PM
To: Howell, Art; Lewis, Robert; Kennedy, Kriss; Scott, Michael; Blount, Tom; Clark, Jeff; Farnholtz, Thomas; Miller, Geoffrey; Haire, Mark; Drake, James; Watkins, John; Kellar, Ray; Gaddy, Vincent; Allen, Don; Azua, Ray; Bradley, Dan; Melfi, Jim; Sanchez, Alfred; Schaup, William; Fairbanks, Abin; Hatfield, Gloria; Jones, Steve; Ahern, Gregory; Alexander,

Ryan; Alferink, Beth; Andrews, Tom; Diederich, Karl; Livermore, Dan; Makris, Nestor; Rodriguez, Jaime
Cc: Quayle, Lisa; Fuller, Karla; Lusk, Rustin; Lackey, Dana; Harrison, Deborah; Karl, Tracy
Subject: Preservation of Information Related to Death of Wade Walters at ANO
Importance: High

On April 23, 2013, we received a letter from a law firm representing the estate and family of Wade Walters related to the recent industrial accident at the Arkansas Nuclear One plant. The firm requested that we preserve all our findings, reports, evidence, data, videos, and all information about the event. We have also received a request pursuant to the Freedom of Information Act for such material. In the near future, we will be circulating instructions concerning the collection and preservation of this material. In the meantime, however, please make sure you preserve all information that is relevant to this event.

If any NRC employees were omitted from the distribution/addressee list who should receive this message, please share it with them immediately.

If you have any questions, please contact me. Thank you for your assistance in this matter.

Karla

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Region IV
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OOS. Releasable

Miller, Geoffrey

From: Jones, Steve
Sent: Tuesday, May 07, 2013 10:17 AM
To: Miller, Geoffrey
Cc: Sanchez, Alfred
Subject: RE: REVIEW: ANO AIT Key Messages

Refer to NRR/DSS/SBPB

Geoff,

I agree the statements are accurate. On the second bullet, multiple members of the licensee's organization stated what you have in brackets.

Steve

From: Miller, Geoffrey
Sent: Tuesday, May 07, 2013 9:38 AM
To: Jones, Steve; Sanchez, Alfred
Subject: REVIEW: ANO AIT Key Messages
Importance: High

Fred/Steve,

Below are some key messages I'd like to use for the AIT exit meeting. Could you please take a look and tell me if the statements are accurate?

Thank you!

Geoff

- Reactor Plant Safety Systems Responded as Designed [point: safety-related stuff worked]

- No Documented Ties between the Heavy Load Lift and Other Outage Activities in the Station Outage Risk Plan [point: rig failure not considered credible risk]

- Entergy Took Appropriate Actions to Recover Plant Equipment on Units 1 and 2 and Has Initiated an Extensive Root Cause Evaluation Effort [point: we don't have a failure cause yet]

Miller, Geoffrey

From: Pannier, Stephen
Sent: Tuesday, July 16, 2013 3:14 PM
To: Miller, Geoffrey
Subject: RE: Potential Generic Comms from ANO AIT

Refer NRR

Hi Geoff,

I am still thinking about how to proceed. As you know... pushing IN's into the pipeline is a daunting task. I had thought about bundling all three and putting this out as one IN. I am not sure licensees read inspection reports, but I know that they at least acknowledge receipt of an IN.

Thanks for asking. I'll be in touch.

Steve

From: Miller, Geoffrey
Sent: Friday, July 12, 2013 4:14 PM
To: Pannier, Stephen
Subject: RE: Potential Generic Comms from ANO AIT

Release

Steve,

Have you received word on whether or not any of the below topics would be suitable subjects for a new or updated generic comm.?

Thank you for your help,

Geoff

From: Miller, Geoffrey
Sent: Wednesday, June 05, 2013 8:40 AM
To: Pannier, Stephen
Subject: Potential Generic Comms from ANO AIT

Release

Steve,

Below are some topics flagged as potential subjects for generic communications during the ANO AIT. Could you please take a look and let me know if there would be benefit in pursuing any of these based on existing comms/OpE?

Thank you very much for your help,

Geoff

- Regarding the impact of trolley on Unit 2 turbine deck. potential generic communication associated with assessing scope of area when evaluating load paths (i.e., impact of heavy loads ***outside*** of intended load path).

- Potential generic communication associated with integration of major project schedules for outage risk assessment (IF confirmed to be an issue by AIT follow up inspection).

- Potential generic comm for diverse methods of supply for nozzle dams (N2 bottles) Release

Refer NRR

Werner, Greg

From:	Weerakkody, Sunil
Sent:	Tuesday, November 19, 2013 11:46 AM
To:	Loveless, David; Miller, Geoffrey; Werner, Greg
Cc:	Mitman, Jeffrey
Subject:	prelimianry draft of ANO Stator Drop SDP Draft Revison 0
Attachments:	ANO1 LOOP SDP Analysis Rev 0.0.docx

From: Mitman, Jeffrey
Sent: Thursday, November 14, 2013 4:32 PM
To: Weerakkody, Sunil
Subject: ANO Stator Drop SDP Draft Revison 0

Sunil, attached is the subject analysis and a zip file containing the SPAR model for review and comment. The zip file may be too big to email to Region IV. I'm still working on the SDP on the loss of SDC at midloop.

Jeff Mitman

Refer to NRR

Cut Set Report - M6-LOOP2 (ET)

Nov 15, 2013

PWR D SPAR MODEL FOR ARKANSAS NUCLEAR ONE UNIT 1

#	Prob/ Freq	Total %	Cut Set	Description
Total	1.51E-4	100	Displaying 50 of 43171 Cut Sets.	
1	6.46E-5	42.7	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
2	2.92E-5	19.3	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	4.09E-4		EPS-DGN-CF-DG12R	CCF OF DIESEL GENERATORS DG1&DG2 TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
3	6.21E-6	4.11	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	2.89E-3		EPS-DGN-FS-DG2	DIESEL GENERATOR 2 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
4	6.21E-6	4.11	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
5	5.14E-6	3.4	M6-LOOP2 . 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A308	4160V AC BREAKER 152-308 FAILS TO CLOSE
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
6	5.14E-6	3.4	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A408	4160V AC BREAKER 152-408 FAILS TO CLOSE
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
7	2.91E-6	1.92	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	9.93E-1		SWS-4C-RUNNING	SWS MDP P4C IS RUNNING; 4B ALIGNED TO RED TRAIN
	1.36E-3		SWS-MDP-FS-P4C	SERVICE WATER MDP P4C FAILS TO START
8	2.58E-6	1.71	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.61E-5		EPS-DGN-CF-DG12S	CCF OF DIESEL GENERATORS DG1&DG2 TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
9	2.15E-6	1.42	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	1.00E-3		EPS-XHE-XR-DG2	OP FAILS TO RESTORE DIESEL GENERATOR 2
10	2.15E-6	1.42	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS

	1.00E-3		EPS-XHE-XR-DG1	OP FAILS TO RESTORE DIESEL GENERATOR 1
11	2.07E-6	1.37	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	9.63E-4		EPS-MOV-CC-CV3807	SWS SUPPLY MOV CV-3807 TO DGN 2 COOLING FAILS TO OPEN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
12	2.07E-6	1.37	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	9.63E-4		EPS-MOV-CC-CV3806	SWS SUPPLY MOV CV-3806 TO DGN 1 COOLING FAILS TO OPEN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
13	1.69E-6	1.12	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.37E-5		EPS-MDP-CF-P16ABS	CCF of EDG Fuel Oil Pump to Start
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
14	1.33E-6	0.88	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	1.86E-5		EPS-MOV-CF-SWS	CCF OF SWS SUPPLY MOVs 3806 AND 3807
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
15	1.22E-6	0.8	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	9.63E-4		LPI-MOV-CC-CV1400	LPI DISCHARGE MOV CV-1400 FAILS TO OPEN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
16	1.22E-6	0.8	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	9.63E-4		LPI-MOV-CC-CV1401	LPI DISCHARGE MOV CV-1401 FAILS TO OPEN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
17	1.20E-6	0.79	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
	9.51E-4		SWS-AOV-CC-CV3841	FAILURE OF SWS MOV CV-3841 TO PMP P34A TO OPEN
18	1.20E-6	0.79	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
	9.51E-4		SWS-AOV-CC-CV3840	FAILURE OF SWS AOV CV-3840 TO PMP P34A TO OPEN
19	1.20E-6	0.79	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	9.47E-4		LPI-MDP-FS-P34A	LPI MDP P34A FAILS TO START
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
20	1.20E-6	0.79	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	9.47E-4		LPI-MDP-FS-P34B	LPI MDP P34B FAILS TO START
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
21	1.04E-6	0.69	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6

	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
	2.48E-5		SWS-AOV-CF-CV38401	CCF OF SWS AOVs CV-3840/3841 TO PUMPS P34A/B TO OPEN
22	9.95E-7	0.66	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.37E-5		LPI-MDP-CF-STRT	LPI PUMP COMMON CAUSE FAILURES TO START
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
23	9.03E-7	0.6	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	1.26E-5		EPS-MDP-CF-P16ABR	CCF of EDG Fuel Oil Pump to Run
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
24	7.70E-7	0.51	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	1.83E-5		LPI-ACX-CF-VC1XR	Common Cause failure of DHR Unit Coolers VUC-1A,1B, 1C & 1D to RUN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
25	5.97E-7	0.39	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START
	2.89E-3		EPS-DGN-FS-DG2	DIESEL GENERATOR 2 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
26	5.31E-7	0.35	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	1.26E-5		LPI-MDP-CF-RUN	LPI PUMP COMMON CAUSE FAILURES TO RUN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
27	4.94E-7	0.33	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A308	4160V AC BREAKER 152-308 FAILS TO CLOSE
	2.89E-3		EPS-DGN-FS-DG2	DIESEL GENERATOR 2 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
28	4.94E-7	0.33	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A408	4160V AC BREAKER 152-408 FAILS TO CLOSE
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
29	4.58E-7	0.3	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	3.62E-4		LPI-MDP-FR-P34A	LPI MDP P34A FAILS TO RUN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
30	4.58E-7	0.3	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	3.62E-4		LPI-MDP-FR-P34B	LPI MDP P34B FAILS TO RUN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
31	4.09E-7	0.27	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A308	4160V AC BREAKER 152-308 FAILS TO CLOSE
	2.39E-3		ACP-CRB-OO-1A408	4160V AC BREAKER 152-408 FAILS TO CLOSE
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
32	2.79E-7	0.18	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START

	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	9.93E-1		SWS-4C-RUNNING	SWS MDP P4C IS RUNNING; 4B ALIGNED TO RED TRAIN
	1.36E-3		SWS-MDP-FS-P4C	SERVICE WATER MDP P4C FAILS TO START
33	2.63E-7	0.17	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	9.93E-1		SWS-4A-RUNNING	SWS MDP P4A IS RUNNING; 4B ALIGNED TO RED TRAIN
	9.93E-1		SWS-4C-RUNNING	SWS MDP P4C IS RUNNING; 4B ALIGNED TO RED TRAIN
	3.73E-6		SWS-MDP-CF-STRT4ABC	CCF OF SERVICE WATER MDPS 4A,4B & 4C TO START
34	2.31E-7	0.15	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A308	4160V AC BREAKER 152-308 FAILS TO CLOSE
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	9.93E-1		SWS-4C-RUNNING	SWS MDP P4C IS RUNNING; 4B ALIGNED TO RED TRAIN
	1.36E-3		SWS-MDP-FS-P4C	SERVICE WATER MDP P4C FAILS TO START
35	2.06E-7	0.14	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	1.00E-3		EPS-XHE-XR-DG2	OP FAILS TO RESTORE DIESEL GENERATOR 2
36	2.06E-7	0.14	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG2	DIESEL GENERATOR 2 FAILS TO START
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	1.00E-3		EPS-XHE-XR-DG1	OP FAILS TO RESTORE DIESEL GENERATOR 1
37	1.99E-7	0.13	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START
	9.63E-4		EPS-MOV-CC-CV3807	SWS SUPPLY MOV CV-3807 TO DGN 2 COOLING FAILS TO OPEN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
38	1.99E-7	0.13	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG2	DIESEL GENERATOR 2 FAILS TO START
	9.63E-4		EPS-MOV-CC-CV3806	SWS SUPPLY MOV CV-3806 TO DGN 1 COOLING FAILS TO OPEN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
39	1.90E-7	0.13	M6-LOOP2 : 08	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.33E-5		ACP-BAC-LP-LCCB5	FAILURE OF LCC B5 BUS
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	1.90E-1		LTREC-DHR-3D	Late Recovery of SDC/DHR (3 Days)
40	1.81E-7	0.12	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	9.93E-1		SWS-4C-RUNNING	SWS MDP P4C IS RUNNING; 4B ALIGNED TO RED TRAIN
	8.47E-5		SWS-MDP-FR-P4C	SERVICE WATER MDP P4C FAILS TO RUN
41	1.71E-7	0.11	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A308	4160V AC BREAKER 152-308 FAILS TO CLOSE
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	1.00E-3		EPS-XHE-XR-DG2	OP FAILS TO RESTORE DIESEL GENERATOR 2

42	1.71E-7	0.11	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A408	4160V AC BREAKER 152-408 FAILS TO CLOSE
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
	1.00E-3		EPS-XHE-XR-DG1	OP FAILS TO RESTORE DIESEL GENERATOR 1
43	1.65E-7	0.11	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A308	4160V AC BREAKER 152-308 FAILS TO CLOSE
	9.63E-4		EPS-MOV-CC-CV3807	SWS SUPPLY MOV CV-3807 TO DGN 2 COOLING FAILS TO OPEN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
44	1.65E-7	0.11	M6-LOOP2 : 19	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.39E-3		ACP-CRB-OO-1A408	4160V AC BREAKER 152-408 FAILS TO CLOSE
	9.63E-4		EPS-MOV-CC-CV3806	SWS SUPPLY MOV CV-3806 TO DGN 1 COOLING FAILS TO OPEN
	7.14E-2		EPS-XHE-XL-NR72H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 72 HOURS
45	1.29E-7	0.09	M6-LOOP2 : 08	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.27E-5		ACP-TFM-FC-X5	4160V/480V TRANSFORMER X5 FAILS
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	1.90E-1		LTREC-DHR-3D	Late Recovery of SDC/DHR (3 Days)
46	1.29E-7	0.09	M6-LOOP2 : 08	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.27E-5		ACP-TFM-FC-X6	4160V/480V TRANSFORMER X6 FAILS
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	1.90E-1		LTREC-DHR-3D	Late Recovery of SDC/DHR (3 Days)
47	1.22E-7	0.08	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-6		LPI-ACX-CF-VC1XS	Common Cause failure of DHR Unit Coolers VUC-1A,1B, 1C & 1D to Start
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
48	1.20E-7	0.08	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG2	DIESEL GENERATOR 2 FAILS TO RUN
	9.50E-5		LPI-ACX-CF-VC1ABR	Common Cause failure of DHR Unit Coolers VUC-1A and 1B to Run
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
49	1.20E-7	0.08	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	3.01E-2		EPS-DGN-FR-DG1	DIESEL GENERATOR 1 FAILS TO RUN
	9.50E-5		LPI-ACX-CF-VC1CDR	Common Cause failure of DHR Unit Coolers VUC-1C and 1D to Run
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)
50	1.17E-7	0.08	M6-LOOP2 : 04	
	1.00E+0		IE-M6-LOOP	LOOP Event Occurs during Mode 6
	2.89E-3		EPS-DGN-FS-DG1	DIESEL GENERATOR 1 FAILS TO START
	9.63E-4		LPI-MOV-CC-CV1400	LPI DISCHARGE MOV CV-1400 FAILS TO OPEN
	4.20E-2		LTREC-DHR-5D	Late Recovery of SDC/DHR (5 Days)

(NRR) Refer in Entirety + Attachment

Werner, Greg

From: Weerakkody, Sunil
Sent: Tuesday, November 19, 2013 11:46 AM
To: Loveless, David; Miller, Geoffrey; Werner, Greg
Cc: Mitman, Jeffrey
Subject: prelimianry draft of ANO Stator Drop SDP Draft Revison 0
Attachments: ANO1 LOOP SDP Analysis Rev 0.0.docx

From: Mitman, Jeffrey
Sent: Thursday, November 14, 2013 4:32 PM
To: Weerakkody, Sunil
Subject: ANO Stator Drop SDP Draft Revison 0

Sunil, attached is the subject analysis and a zip file containing the SPAR model for review and comment. The zip file may be too big to email to Region IV. I'm still working on the SDP on the loss of SDC at midloop.

Jeff Mitman

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Phase 3 Risk Assessment
Loss of Offsite Power
Arkansas Nuclear One Unit 1

Revision 0.0

Probabilistic Risk Assessment (PRA) Analyst:	Jeff Mitman, Senior Reliability and Risk Analyst, NRR/DRA/APOB
Independent Reviewer	
Region IV Reviewer	

1.0 Introduction

On March 31st 2013 at 7:50 am Arkansas Nuclear One Unit 1 (ANO1) experienced a loss of offsite power. While lifting and transferring the Unit 1 main generator stator to the train bay, the lift system failed, falling on to the turbine deck and into the train bay. This resulted in damage to the turbine building including damage to electrical buses supplying offsite power to Unit1 and damage to the fire suppression piping.

At the time of the event, Unit 1 was in a refueling outage. It had been shutdown approximately 7 days. Fuel in the reactor vessel, the reactor cavity was flooded up, and both trains of decay heat removal system were in service. With the loss of offsite power, both Unit 1emergency diesel generators started and loaded their respective buses. Decay heat removal (DHR) was quickly restored. Once DHR was restored the unit was quasi stable with no offsite power available due to damage to the non-vital electrical buses, with EDGs powering the vital busses and the decay heat removal system operating and providing decay heat removal to the reactor vessel.

Dropping the generator stator caused the following damage:

- Offsite power was lost – it took six days to recover
- The station blackout diesel generator's (called the AAC) connection to the plant was severed rendering the ACC non-functional
- Fire watering piping was damaged requiring shutdown of the fire protection system. It also caused flooding the Unit 1 and 2 structures with tens of thousands of gallons of water challenging critical equipment
-

2.0 Discussion of the Performance Deficiency

The licensee failed to properly implement Engineering Procedure EN-MA-119, "Material Handling Program." The following two examples are presented:

- <u>The licensee failed to adequately review and approve Bigge Calculation 27619-C1 as required by Section 5.2[7](a)</u>

  Engineering Procedure EN-MA-119, Section 5.2[7] requires temporary hoisting assemblies to be designed or approved by Engineering Support Personnel (ESP). On September 12, 2012 Siemens Energy transmitted to Entergy, Bigge Calculation 27619-C1, "ANO Stator Replacement Project." The design calculation did not adequately consider the loads that would be experienced by the lift. Entergy's review and approval process failed to identify the calculation deficiencies and the weak component in the north tower structure. Specifically, Entergy's ESP failed to adequately review and identify the flaw in Calculation 27619-C1 consistent with the requirements of procedure Section 5.2[7](a) which states that temporary hoisting assemblies are required to be designed or approved by ESP. Had ESP's approval process identified the deficiency and eliminated it prior to use of the assembly at ANO, the event would have been avoided.

- The licensee failed to ensure that a load test of the assembly to at least 125 percent of the projected hook load or to another approved standard was performed as required by Section 5.2[7](b) and associated note

  Engineering Procedure EN-MA-119, Section 5.2[7](b) requires assembly's to be load tested and held for at least five minutes at 125 percent of actual load rating before initial use. However, the note in Section 5.2[7] allows specially designed devices, for specific applications to be designed and tested to other approved standards. On February 14. 2013 Entergy's supplemental Project Civil Engineer reviewed the letter from Bigge to Siemens Energy, dated February 8, 2013, and included that letter into Engineering Change Notice ECN-39028. The Entergy engineer failed to identify that the upper columns and intermediate header were listed as new, negating Bigge's assertion that, "This hoist assembly has been used at other electric power stations to lift components that exceed the anticipated weight of the unit 1 stator." This erroneous information was then used in lieu of a load test or testing in accordance with other approved standards. Had engineering personnel identified the erroneous information and a load test, or testing to other approved standards, been performed, the deficiencies in the design would have been detected prior to use at ANO and the event would have been avoided.

3.0 Plant Conditions Prior to the Event

Plant equipment and conditions were as follows:

- *Unit was shutdown with fuel in the reactor, head removed and refueling canal flooded*
- Estimated time to boil (TTB) was 8 hours
- Estimated time to core uncovery was 3 days
- Both trains of SDC were in service
- Plant electrical lineup was in a plant shutdown configuration to support maintenance and testing as follows:
  - 6900 Volt Bus H2 was de-energized.
  - 6900 Volt Bus H1 was energized.
  - 4160 Volt Bus A2 was de-energized.
  - Safety related 4160 Volt Buses A3 and A4 were cross tied and supplied power via Non-safety related 4160 Volt bus A1.
  - 480 Volt buses B5 and B6 were cross tied.
  - Green Train battery D06 had been disconnected from D02 bus.
  - D04 battery charger was supplied from Swing MCC B56 to provide power to Green Train DC bus D02.
  - B56 was aligned to B5.

4.0 Plant Conditions after Initiating Event Initiated

Time to boil was estimated at eight hours and time to core uncovery without mitigation was estimated at three days.

The following equipment was unavailable after event initiation:

- Offsite power
- Station blackout diesel generator - ACC
- Fire water
- All balance of plant equipment
- Gravity feed from the BWST as water level in the BWST was lower than water level in RCS
- Instrument air (IA) was unavailable – the analyst assumed that all air operated valve failed in a safe direction, i.e.. the systems IA supported remained available
- Starting air compressors for the emergency generators
- Normal lighting

The following equipment was available after the event initiation to mitigate the event:

- Both emergency diesel generators and their respective electrical distribution systems
- Both decay heat removal trains (two pumps)
- Both high pressure injection (HPI) trains (three pumps)
- Reactor building spray systems – note these were not credited in the analysis, however, the non-crediting had no effect on the quantitative results

5.0 Significance Determination Process (SDP) Phase 2 Summary

No Phase 2 was conducted.

6.0 Initiation of a Phase 3 SDP Risk Assessment

A Phase 3 SDP risk assessment was performed by the Office of Nuclear Reactor Regulation (NRR).

The analysts used the following generic references in preparing the risk assessment:

- NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991
- NUREG/CR-6883, "The SPAR-H Human Analysis Method," August 2005
- NUREG-1842, "Good Practices for Implementing Human Reliability Analysis," April 2005
- NUREG/CR-6595 Revision 1, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," October 2004
- INL/EXT-10-18533 Revision 2, "SPAR-H Step-by-Step Guidance," May 2011
- "RASP Manual Volume 1 – Internal Events," Revision 2.0 date January 2013
- NUREG/CR-1278, "Handbook of HRA with Emphasis on Nuclear Power Plant Applications," August 1983

The analyst used the following plant specific references:

- EOP: 1202.007, Degraded Power
- AOPs:
  - 1203.024, Loss of Instrument Air
  - 1203.028, Loss of Decay Heat Removal
  - 1203.050, Unit 1 Spent Fuel Pool Emergencies
- Calculation: 89-E-0017-01, Time to Boiling and Time to Core Uncovery after Loss of Decay Heat Removal, Unit 1, Revision 7
- Procedure: 1103.018, Maintenance of RCS Water Level

7.0 Development of the Model

No Low Power/Shutdown (LP/SD) SPAR model exists for ANO1. Therefore, the at-power ANO1 SPAR model was modified to allow analysis of the loss of offsite power (LOOP) event. A new event tree (ET) was created to analyze the event.

This ET is shown in Figure A-1 of Appendix A. The ET was linked to a mix of existing at-power fault trees (FT) and new FTs, as applicable. The existing FTs were modified as necessary to appropriately describe system dependencies during shutdown conditions and the different success criterion. The ET and high level FTs are shown in Appendix A.

HRA Analysis

Shutdown operation is highly dependent on operator actions as most of the required actions are manual (e.g., initiating feed of the RCS). HRA analysis was conducted to properly characterize the required manual actions. The human error probabilities (HEPs) were calculated using the Low Power Shutdown SPAR-H worksheets from NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method" and INL/EXT-10-18533 and SPAR-H Step-by-Step." Consideration was given to the available time to perform the action, the stress levels of the crew during the event, complexity of the diagnoses and actions, crew experience and applicable and relevant training, quality and thoroughness of procedures, ergonomics, fitness of duty issues, and the available work processes. Table 1 shows a summary of the dominant HEPs, a detailed discussion of the HEPs is given in Appendix B.

In addition to the calculation of specific HEPs for this condition, sequences or cutsets which involved multiple operator actions were examined for human action dependency. For the dominant HEPs no dependent couplets were found.

In addition, the cutsets were reviewed to find those that contained two or more HEPs in a single sequence of cutset. For those cutset with multiple HEPs, the HEPs were reviewed to determine if the product of the HEPs was less than 1E-6. For those cutsets a floor, or cutoff, was applied as directed by *RASP Manual Volume 4 – Shutdown Events*, Revision 1 Appendix B. A because of the long times to core damage, a cutoff of 1E-7 was applied. This conservative assumption did not materially affect the results

Normal lighting was impacted by the LOOP. This could have an impact on the ability of the equipment operators to perform tasks outside of the main control. This impact was not assessed.

A detailed description of the HEPs is given in Appendix B.

Table 1
Summary of Dominant HRA Results

Human Error Event	Description	Time Needed	Time Available	Mean Diagnosis HEP	Mean Action HEP	Total Mean HEP
SD-XHE-D-LOSDC	Operator Fails to Diagnose Loss of SDC before boiling	5 minutes	8 hours	2E-5	n/a	2E-5
SD-XHE-XL-LOSDC	Operator Fails to Recover Loss of SDC before Boiling	30 minutes	8 hours	n/a	4E-4	4E-4
SD-XHE-XL-MINJ	Operator Fails to Inject (AC power available) before Level Reaches TAF	30 minutes	3 days	n/a	2E-5	2E-5
SD-XHE-XL-LPR	Operator Fails to Initiate Low Pressure Recirc	1 hour	4 days	2E-5	2E-4	2.2E-4
SD-XHE-XM-BWST	Operator Fails to Refill BWST during Shutdown	10 hour	4 days	n/a	2E-5	2E-5

8.0 Conditional Core Damage Probability (CCDP) Assessment Results

A detailed Phase 3 Significance Determination Process risk analysis was performed consistent with NRC Inspection Manual Chapter (IMC) 0609 Appendix G. Step 4.3.8 of this procedure directs the analyst to assess the significance of shutdown events by calculating an instantaneous conditional core damage probability (ICCDP). (Throughout this assessment, the analyst has used the terminology of CCDP instead of ICCDP for simplicity.) This assessment was performed by setting the initiating event frequency (IEF) for loss of offsite power to 1.0 and all other IEF to zero. The above described SPAR model was evaluated using the SAPHIRE code version 8.0.9.0.

As this SDP evaluates an actual event in which no external events occurred, there was no risk from external events. As discussed in the above paragraph, this would include setting any external event IEF to zero also.

The truncation limit was set at 1E-16.

The result of the CCDP analysis is 1.6E-4; based on these results the finding is Red. The top cutsets for are in Appendix C. The analyst did not perform uncertainty analysis.

Table 2
CCDP Results

Sequence	Point Estimate	Cut Set Count
4	1.5E-5	8368
6	1.3E-7	3370
8	1.1E-6	25193
11	1.0E-7	834
13	1.0E-7	134
15	1.0E-7	915
19	1.4E-4	4357
Total	**1.6E-4**	43171

The results are dominated by two sequences. The largest contributor is from Sequence 19 which comprises a failure of the emergency diesel generators (EDG) without recovery. Both the EDG and EDG non-recovery failure probabilities were calculated using the standard SPAR methods and models. Sequence 4 is also a significant contributor. Sequence 4 cutsets are dominated by combinations of equipment and failure to recover DHR.

The numeric results above quantify to a Red finding. However, with such a long time to core damage, recovery is possible with temporary systems such as B.5.b equipment. The analyst is unaware of procedures or training to cool the RCS during these conditions. In addition, condition in the reactor building will become difficult if not life threatening once boiling begins. In conclusion, some credit for these type of actions is warranted using a SPAR-H approach (note neither SPAR-H nor any other HRA method were ever intended to quantify these type of scenarios) would quantify this with a failure probability between 0.1 and 0.5. If such credit were given, this would reduce the finding into the Yellow range.

9.0 Conditional Large Early Release Probability (CLERP) Assessment

The figure of merit for this analysis is incremental conditional large early release probability (ICLERP). This ICLERP analysis is based on the method for shutdown described in NUREG/CR-6595 Revision 1, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," dated 10/2004. This report supplies simplified containment event trees (CET) to determine if the core damage sequence contributes to LERF. NUREG/CR-6595 presents its analysis in terms of LERF, which is interpreted here as ICLERP.

NUREG/CR-6595 defines LERF as "... the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects." This is identical to the definition of LERF in IMC 0609 Appendix H. Figure 4.2 (PWR Large Dry and Sub-atmospheric Containment Event Tree) from NUREG/CR-6595 is applicable to the ANO1 event.

This event occurred seven days after shutdown. The earliest core damage could occur would be three days after event initiation. Thus core damage would not occur until 10 days after shutdown. Based on this time and the recommended approach given by NUREG/CF-6595 no large early release could occur.

10.0 Sensitivity Analysis

Several sensitivity cases were conducted to further understand the event risk significance. The cases are described below.

Case 1: Loss of Instrument Air

The LOOP event on Unit 1 in combination with the partial LOOP in Unit 2 combined to cause a loss of instrument air on Unit. There does not appear to be any impact on Unit 1 from the loss of air. However, instrument air was being supplied to the steam generator nozzle dams. If the nozzle dams had failed, water level could have drained to the bottom of the steam generator openings. The nozzle dam design appears to preclude a significant inventory on loss of air. The design limits the leakage to a couple of gallons per minute on each nozzle dam. With several hundred thousand gallons of water above the nozzle dams this leakage rate is insignificant.

Case 2: HRA No Cutoff

A case was conducted to verify the sensitivity of the results to the cutoff value. This case was run with truncation level of 1E-16. The calculated CCDP was 1.6E-4. This indicates that the cutoff implementation is a second order effect only.

Sequence	Point Estimate	Cut Set Count
4	1.5E-05	8368
6	2.5E-08	3370
8	1.0E-06	25193
11	5.7E-10	834
13	9.5E-13	134
15	4.6E-10	915
19	1.4E-04	4357
Total	**1.6E-04**	43171

Case 3: DC Flooding

The stator drop severed a fire water header pipe. It took approximately 45 minutes to stop this leakage. Before the leakage was stopped, water accumulated into the Unit 1 and 2 turbine buildings where it caused a small Unit 2 kV fire/explosion. This caused a loss of offsite power to one division of Unit 2 AC power which was mitigated by the associated emergency diesel generator. Water also started to accumulate into the Unit 1 SDC/DHR B pump vault. If this accumulation continued it could have failed the pump. Potentially it could have impacted other Unit 1 equipment. Sensitivity cases were conducted with various flooding probabilities and various combinations of impacted equipment. Those combinations and their impacts are presented in the below table. These analyses assume that the flooding could not impact the Unit emergency diesel generator or their associated 4kV switch gear and 480 v MCCs.

This analysis shows that if the flooding had not been terminated in a timely manner it could have had a significant impact on plant safety.

Impacted Equimpment	**CCDP**	
	Flood Probability = 0.1	**Flood Probability = 1.0**[1]
One LPI/SDC/DHR pump (either pump A or B)	3E-4	2E-3
Both LPI/SDC/DHR pumps	1E-3	5E-2
A single HPI pump (either A, B or C)	no impact	1.8E-4
Two HPI pumps (A & B)	no impact	1.8E-4
Two HPI pumps (A & C)	no impact	2.2E-4
All three HPI pumps	no impact	2.2E-4
All of HPI and SDC/DHR	1.1E-3	2.5E-1

Notes: 1) If the associated basic events are set to True instead of 1.0 the CCDPs are somewhat lower as would be expected.
2) These sensitivity cases were run with truncation set to 1E-8.

Case 4: Impact of Loss of EDG Starting Air Compressors

The LOOP caused a loss of normal EDG starting air. If multiple starts of the EDG were required this could impact the restoration of the emergency power. While it is difficult to quantify the change in the EDG non-probability that changes effect on the CCDP is easily assessed. The analyst assumed that the non-recovery probability was double from 7.14E-2 (for 72 hours) to 1.4E-1. The new CCDP is 2.9E-4. Because the risk results are dominated by Sequence 19 which is the only sequence effected by the EDG non-recovery probability, the change in CCDP is directly proportional to the change in the non-recovery probability on Sequence 19.

11.0 Comparison with Licensee Results

At this time the analyst has seen no licensee results to compare.

Appendix A: Model Figures

Figure A-1: Loss of Offsite Power Event Tree

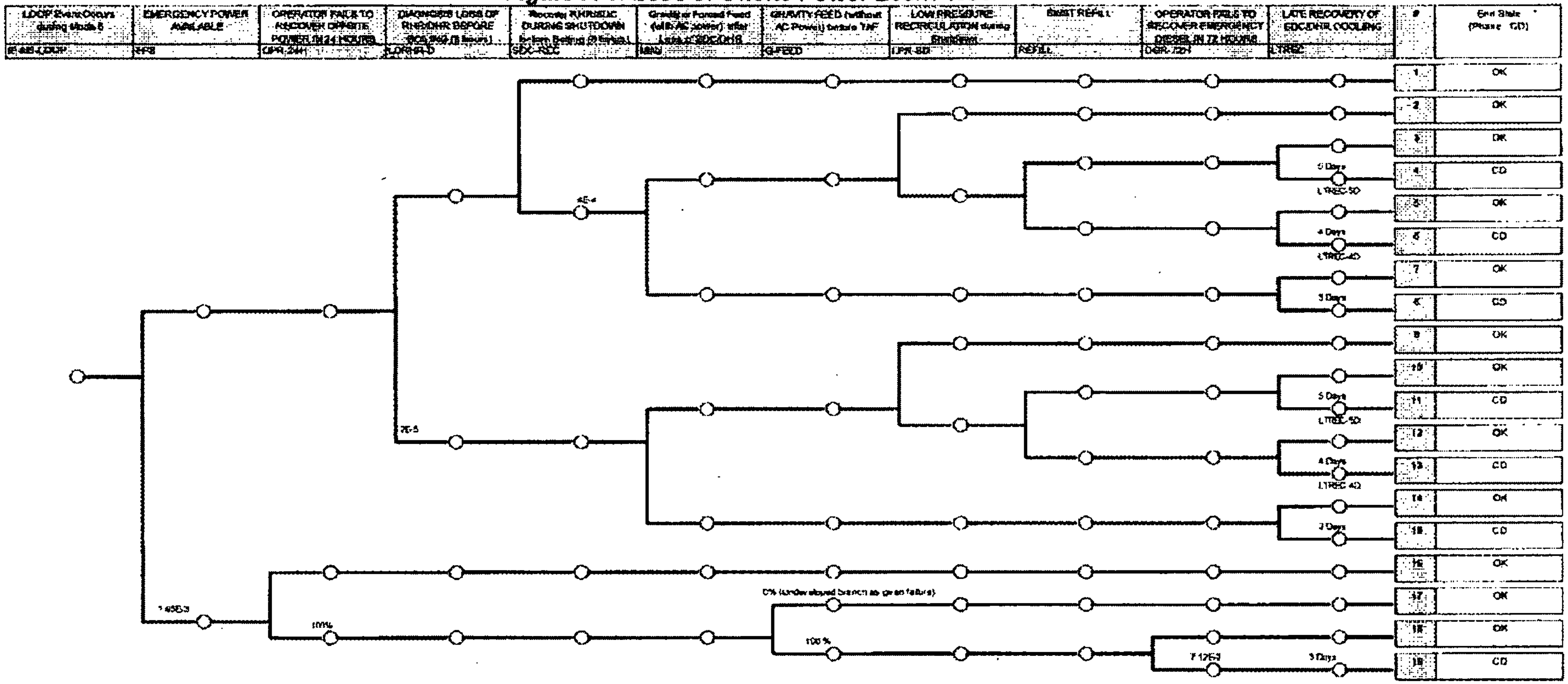


Figure A-2: Emergency Power Failure Fault Tree

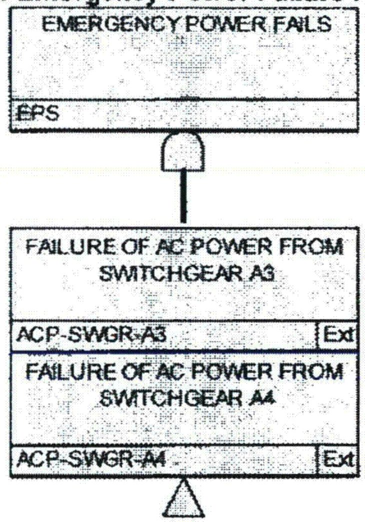
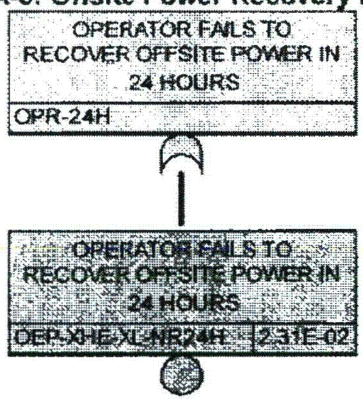


Figure A-3: Offsite Power Recovery Fault Tree



Note that the non-recovery probability was set to one in a change set

Figure A-4: Diagnose Loss of RHR/DHR Fault Tree

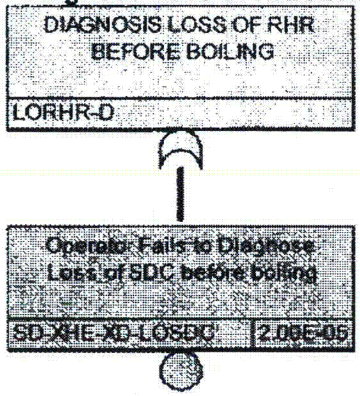


Figure A-5: Recovery RHR/SDC Fault Tree

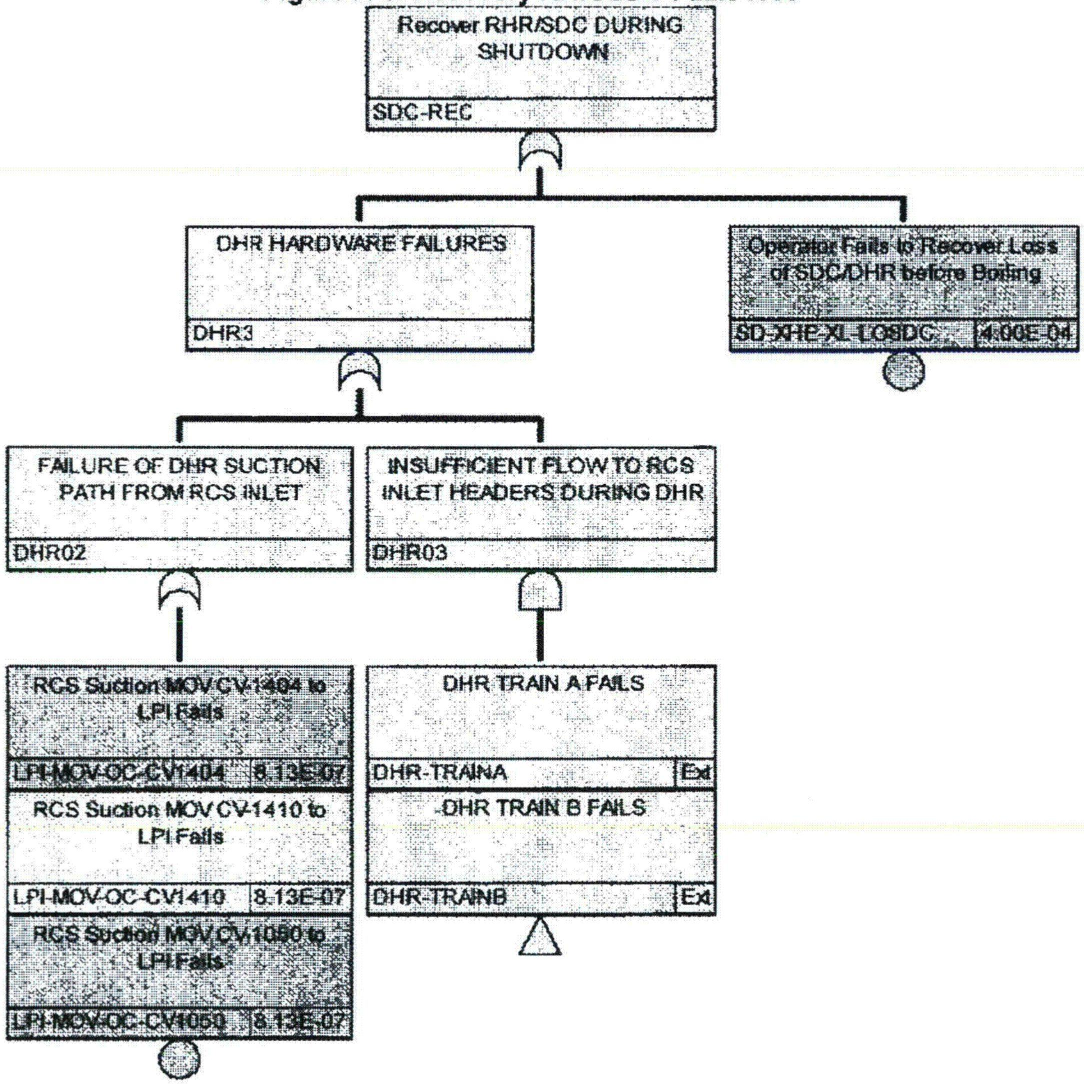
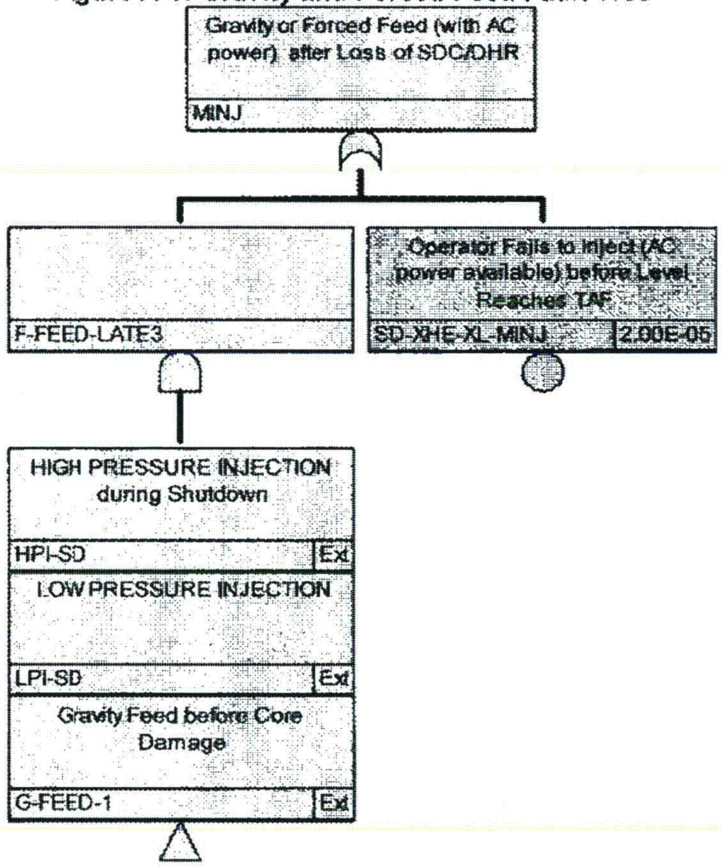
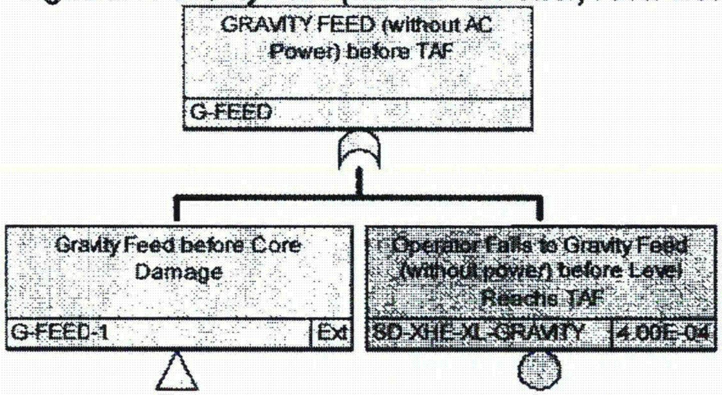


Figure A-6: Gravity and Forced Feed Fault Tree



Note the gravity feed portion of this FT is set to fail as gravity feed will not work because the physical level of the BWST is lower than the refueling canal

Figure A-7: Gravity Feed (without AC Power) Fault Tree



Note this FT is set to fail as gravity feed will not work because the physical level of the BWST is lower than the refueling canal

Figure A-8: Low Pressure Recirculation Fault Tree

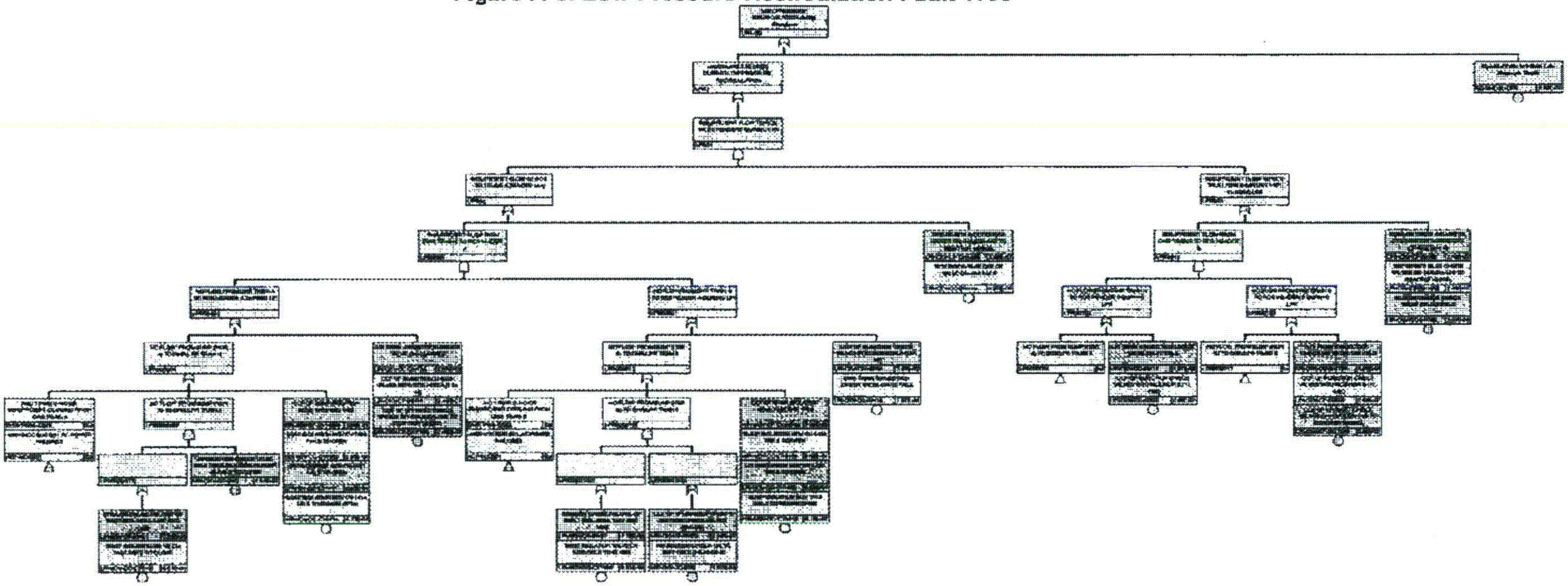


Figure A-9: BWST Refill Fault Tree

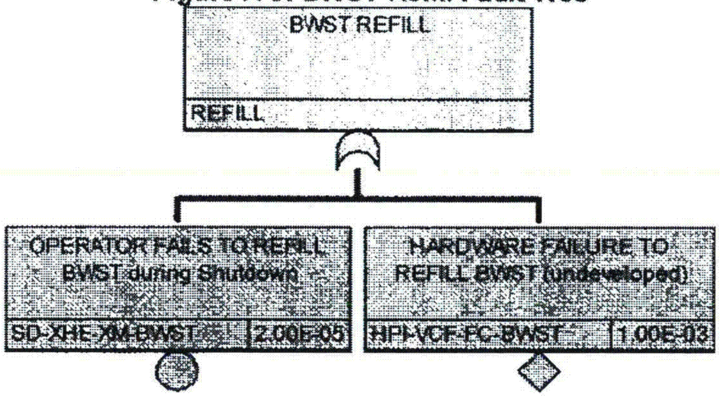


Figure A10: Diesel Generator Recovery Fault Tree

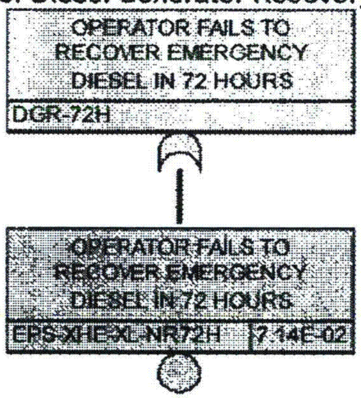
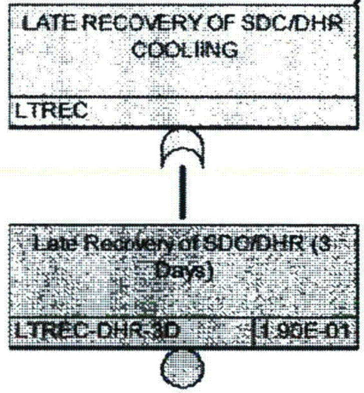


Figure A-11: SDC/DHR Late Recovery Fault Tree



Note the value of the late recovery basic event varies with the time available

Appendix B: HRA Analysis

Human Error Probabilities

A high level discussion of the Human Reliability Analysis (HRA) is presented above in Section 7 on Model Development. Also included above is a summary of the HRA results. The following discusses the Human Failure Events (HFE), the derivation of the in individual Human Error Probabilities (HEP). This HRA analysis was done consistent with the guidance of NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method," dated August 2005.

The Human Error Probabilities (HEPs) for this analysis were calculated using the Low Power Shutdown SPAR-H worksheets from NUREG/CR-6883. Consideration was given to the available time to perform the action, the stress levels of the crew during the event, complexity of the action, crew experience and applicable and relevant training, quality and thoroughness of procedures, ergonomics, fitness of duty issues, and the available work processes.

B1 Operator Fails to Diagnose Loss of SDC before Boiling

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: ANO1 **Initiating Event:** **Basic Event:** SD-XHE-D-LOSDC
Basic Event Description: Operator Fails to Diagnose Loss of SDC before boiling

Part I. DIAGNOSIS WORKSHEET

PSFs	PSF Levels	Multiplier for Diagnosis	Selected PSF	Please note specific reasons for PSF level selection in this column.
Available Time	Inadequate time	P(failure) = 1.0		5 minutes required, 8 hours available
	Barely adequate time (≈2/3 Nominal)	10		
	Nominal time	1		
	Extra time (between 1 and 2 x nominal and > than 30 min)	0.1		
	Expansive time (> 2 x nominal and > 30 min)	0.01	X	
	Insufficient information	1		
Stress	Extreme	5		
	High	2	X	
	Nominal	1		
	Insufficient information	1		
Complexity	Highly	5		
	Moderately Complex	2		
	Nominal	1		
		0.5		
	Obvious diagnosis	0.1	X	Pump stop with loss of power is obvious
	Insufficient information	1		
Experience/ Training	Low	10		
	Nominal	1	X	
	High	0.5		
	Insufficient information	1		
Procedures	Not available	50		
	Incomplete	20		
	Available, but poor	5		
	Nominal	1	X	
	Diagnostic/symptom oriented	0.5		
	Insufficient information	1		
Ergonomics/HMI	Missing/Misleading	50		
	Poor	10		
	Nominal	1	X	
	Good	0.5		
	Insufficient information	1		
Fitness for Duty	Unfit	P(failure) = 1.0		
	Degraded Fitness	5		
	Nominal	1	X	
	Insufficient information	1		
Work Processes	Poor	2		
	Nominal	1	X	
	Good	0.8		
	Insufficient information	1		

NHEP = **2.00E-05**

Negative PSFs adjustment (≥3 negative PSFs) NA

Final Diagnosis HEP **2.00E-05**

B2 Operator Fails to Recover Loss of SDC/DHR before Boiling

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: ANO1 **Initiating Event:** **Basic Event:** SD-XHE-XL-LOSDC
Basic Event Description: Operator Fails to Recover Loss of SDC before boiling

Part II. ACTION WORKSHEET

PSFs	PSF Levels	Multiplier for Action	Selected PSF	Please note specific reasons for PSF level selection in this column.
Available Time	Inadequate time Time Available is ≈ the time required Nominal time Time available is ≥ 5x the time required Time available is ≥ 50x the time required Insufficient information	P(failure) = 1.0 10 1 0.1 0.01 1	 X 	30 minutes required, 8 hours available. SDC/DHR pumps are located in the containment one boiling occurs into containment operation of pumps will be effected
Stress	Extreme High Nominal Insufficient information	5 2 1 1	 X 	
Complexity	Highly Moderately Nominal Insufficient information	5 2 1 1	 X 	
Experience/Training	Low Nominal High Insufficient information	3 1 0.5 1	 X 	
Procedures	Not available Incomplete Available but poor Nominal Insufficient information	50 20 5 1 1	 X 	
Ergonomics/HMI	Missing/Misleading Poor Nominal Good Insufficient information	50 10 1 0.5 1	 X 	
Fitness for Duty	Unfit Degraded Fitness Nominal Insufficient information	P(failure) = 1.0 5 1 1	 X 	
Work Processes	Poor Nominal Good Insufficient information	5 1 0.5 1	 X 	
		Final Action HEP	4.00E-04	

B3 Operator Fails to Inject (AC power available) before Level Reaches TAF

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: NMP1 **Initiating Event:** **Basic Event:** SD-XHE-XL-MINJ
Basic Event Description: Operator Fails to Inject after Level Reaches Scram Setpoint and before it Reaches TAF

Part II. ACTION WORKSHEET

PSFs	PSF Levels	Multiplier for Action	Selected PSF	Please note specific reasons for PSF level selection in this column.
Available Time	Inadequate time	P(failure) = 1.0		
	Time Available is ≈ the time required	10		
	Nominal time	1		
	Time available is ≥ 5x the time required	0.1		
	Time available is ≥ 50x the time required	0.01	X	
	Insufficient information	1		
Stress	Extreme	5		
	High	2	X	
	Nominal	1		
	Insufficient information	1		
Complexity	Highly	5		This assumes that condensate continues to run on loss of DC. If racking in core spray is required this would be moderate.
	Moderately	2		
	Nominal	1	X	
	Insufficient information	1		
Experience/Training	Low	3		
	Nominal	1	X	
	High	0.5		
	Insufficient information	1		
Procedures	Not available	50		
	Incomplete	20		
	Available but poor	5		
	Nominal	1	X	
	Insufficient information	1		
Ergonomics/HMI	Missing/Misleading	50		
	Poor	10		
	Nominal	1	X	
	Good	0.5		
	Insufficient information	1		
Fitness for Duty	Unfit	P(failure) = 1.0		
	Degraded Fitness	5		
	Nominal	1	X	
	Insufficient information	1		
Work Processes	Poor	5		
	Nominal	1	X	
	Good	0.5		
	Insufficient information	1		

NHEP = 2.00E-05

Negative PSFs adjustment (≥3 negative PSFs) NA

Final Action HEP 2.00E-05

B4a Operator Fails to Diagnose Need for Low Pressure Recirc

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: ANO1 **Initiating Event:** **Basic Event:** SD-XHE-XL-LPR
Basic Event Description: Operator Fails to Initiate Low Pressure Recirc

Part I. DIAGNOSIS WORKSHEET

PSFs	PSF Levels	Multiplier for Diagnosis	Selected PSF	Please note specific reasons for PSF level selection in this column.
Available Time	Inadequate time	P(failure) = 1.0		Feed has been started therefore there is at least 24 hours to restart SDC
	Barely adequate time (≈2/3 Nominal)	10		
	Nominal time	1		
	Extra time (between 1 and 2 x nominal and > than 30 min)	0.1		
	Expansive time (> 2 x nominal and > 30 min)	0.01	X	
	Insufficient information	1		
Stress	Extreme	5		
	High	2	X	
	Nominal	1		
	Insufficient information	1		
Complexity	Highly	5		
	Moderately Complex	2		
	Nominal	1	X	
		0.5		
	Obvious diagnosis	0.1		
	Insufficient information	1		Scram setpoint is an obvious cue
Experience/Training	Low	10		
	Nominal	1	X	
	High	0.5		
	Insufficient information	1		
Procedures	Not available	50		
	Incomplete	20		
	Available, but poor	5		
	Nominal	1	X	
	Diagnostic/symptom oriented	0.5		
	Insufficient information	1		
Ergonomics/HMI	Missing/Misleading	50		
	Poor	10		
	Nominal	1	X	
	Good	0.5		
	Insufficient information	1		
Fitness for Duty	Unfit	P(failure) = 1.0		
	Degraded Fitness	5		
	Nominal	1	X	
	Insufficient information	1		
Work Processes	Poor	2		
	Nominal	1	X	
	Good	0.8		
	Insufficient information	1		

NHEP = 2.00E-4

Negative PSFs adjustment (≥3 negative PSFs) NA

Final Diagnosis HEP = 2.00E-4

B4b Operator Fails Action for Low Pressure Recirc

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: ANO1 **Initiating Event:** **Basic Event:** SD-XHE-XL-LPR
Basic Event Description: Operator Fails to Initiate Low Pressure Recirc

Part II. ACTION WORKSHEET

PSFs	PSF Levels	Multiplier for Action	Selected PSF	Please note specific reasons for PSF level selection in this column.
Available Time	Inadequate time	P(failure) = 1.0		
	Time Available is ≈ the time required	10		
	Nominal time	1		
	Time available is ≥ 5x the time required	0.1		
	Time available is ≥ 50x the time required	0.01	X	
	Insufficient information	1		
Stress	Extreme	5		
	High	2	X	
	Nominal	1		
	Insufficient information	1		
Complexity	Highly	5		
	Moderately	2		
	Nominal	1	X	
	Insufficient information	1		
Experience/Training	Low	3		
	Nominal	1	X	
	High	0.5		
	Insufficient information	1		
Procedures	Not available	50		
	Incomplete	20		
	Available but poor	5		
	Nominal	1	X	
	Insufficient information	1		
Ergonomics/HMI	Missing/Misleading	50		
	Poor	10		
	Nominal	1	X	
	Good	0.5		
	Insufficient information	1		
Fitness for Duty	Unfit	P(failure) = 1.0		
	Degraded Fitness	5		
	Nominal	1	X	
	Insufficient information	1		
Work Processes	Poor	5		
	Nominal	1	X	
	Good	0.5		
	Insufficient information	1		

NHEP = 2.00E-05

Negative PSFs adjustment (≥3 negative PSFs) NA

Final Action HEP 2.00E-05

Appendix C: Cutsets

Top 20 Cutsets:

Fairbanks, Abin

From: Tindell, Brian
Sent: Tuesday, November 19, 2013 2:21 PM
To: Young, Matt; Fairbanks, Abin
Subject: FW: prelimianry draft of ANO Stator Drop SDP Draft Revison 0
Attachments: ANO1 LOOP SDP Analysis Rev 0.0.docx

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FYI – I read through this. Preliminary Yellow just for the stator drop.

From: Werner, Greg
Sent: Tuesday, November 19, 2013 3:03 PM
To: Bloodgood, Michael; Melfi, Jim; Tindell, Brian
Subject: FW: prelimianry draft of ANO Stator Drop SDP Draft Revison 0

FYI

From: Weerakkody, Sunil
Sent: Tuesday, November 19, 2013 11:46 AM
To: Loveless, David; Miller, Geoffrey; Werner, Greg
Cc: Mitman, Jeffrey
Subject: prelimianry draft of ANO Stator Drop SDP Draft Revison 0

From: Mitman, Jeffrey
Sent: Thursday, November 14, 2013 4:32 PM
To: Weerakkody, Sunil
Subject: ANO Stator Drop SDP Draft Revison 0

Refer to NRR/DRA/APOB

Sunil, attached is the subject analysis and a zip file containing the SPAR model for review and comment. The zip file may be too big to email to Region IV. I'm still working on the SDP on the loss of SDC at midloop.

Jeff Mitman

Torbey, Andrea

From: Telson, Ross *Reactor Systems Engineer NRR/DIRS/IRIB*
Sent: Friday, April 05, 2013 5:49 PM *2515 - Supplemental Inspections lead*
To: Kobetz, Timothy
Cc: Lewin, Aron; Cauffman, Christopher; Klett, Audrey; Levasseur, Gabriel; Isom, James; Campbell, Stephen; Cartwright, William; Gamberoni, Marsha
Subject: FW: ANO MD 8.3
Attachments: MD 8 3 for ANO stator drop Rev3.docx

FYI – ANO AIT

From: Pannier, Stephen *Reactor Systems Engineer, NRR/DIRS/IOEB*
Sent: Friday, April 05, 2013 9:36 AM
To: Nieh, Ho; Howe, Allen
Cc: King, Mark; Markley, Michael; Telson, Ross; Sigmon, Rebecca
Subject: FW: ANO MD 8.3

Attached is a copy of the ANO MD 8.3 determination.

Thanks

Steve

From: Allen, Don
Sent: Friday, April 05, 2013 9:25 AM
To: Wang, Alan; Chernoff, Harold; Weerakkody, Sunil; Balazik, Michael; Jones, Steve; Mendiola, Anthony; Pannier, Stephen; Loveless, David; Clark, Jeff; Blount, Barbara; Garmon, David
Subject: FW: ANO MD 8.3

For the discussion at 9:00 central time today

From: Lusk, Rustin
Sent: Friday, April 05, 2013 8:18 AM
To: Markley, Michael
Cc: Allen, Don; Miller, Geoffrey
Subject: ANO MD 8.3

Good Morning,

Please see the attached report. Thank you.

Respectfully,

Rustin "Russ" Lusk
Division of Reactor Projects, Division Admin Assistant
U.S. Nuclear Regulatory Commission, Region IV
Phone: (817) 200-1184
Fax: (817) 200-1278

"Faith is being sure of what we do not see & certain of what we hope for."
R.I.P. Rick "Rypper" Rypien

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

April 5, 2013

Release

MEMORANDUM TO: Arthur T. Howell III, Regional Administrator

THRU: Kriss M. Kennedy, Director, Division of Reactor Projects

FROM: Donald B. Allen, Chief, Reactor Projects Branch E

SUBJECT: MANAGEMENT DIRECTIVE 8.3 EVALUATION FOR ARKANSAS NUCLEAR ONE

Pursuant to Regional Office Policy Guide 0801, "Documenting Management Directive 8.3 Reactive Team Inspection Decisions," the enclosed table provides the Management Directive 8.3 evaluation of the March 31, 2013, event at Arkansas Nuclear One involving the failure of the Unit 1 main generator stator lifting rig. Based on the results of the MD 8.3 evaluation (attached), I recommend that we conduct an augmented inspection at Arkansas Nuclear One.

Concur with Recommendation:______________________________________

Arthur T. Howell III, Regional Administrator

Enclosure:
MD 8.3 Decision Documentation Form

Electronic distribution by RIV:
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Deputy Regional Administrator (Robert.Lewis@nrc.gov)
DRP Director (Kriss.Kennedy@nrc.gov)
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DRP Director, Region III (Steven.Reynolds@nrc.gov)

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R:_MD 8.3 Decisions\ ADAMS ML

SUNSI Rev Compl.	☑ Yes ☐ No	ADAMS	☑Yes ☐ No	Reviewer Initials	DBA
Publicly Avail Public Avail Date	☐ Yes ☑No	Sensitive Keyword	☑Yes ☐ No MD 3.4/A.7	Sens. Type Initials	DBA

RIV:C/DRP/E	D:DRS	D:DRP	
DAllen	TBlount	KKennedy	

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

MD 8.3/IMC 0309
DECISION DOCUMENTATION FORM
(Deterministic and Risk Criteria Analyzed)

PLANT	Arkansas Nuclear One	**EVENT DATE**	3/31/2013
RESPONSIBLE BRANCH CHIEF	Don Allen	**EVALUATION DATE**	4/1/2013

BRIEF DESCRIPTION OF THE SIGNIFICANT OPERATIONAL EVENT OR DEGRADED CONDITION

On March 31, 2013, at approximately 7:50 a.m. (CDT), Arkansas Nuclear One (ANO), Unit 1 which was in a refueling outage, lost offsite power and ANO Unit 2 experienced a reactor trip after a 600 ton generator stator fell onto the turbine deck and then approximately 30 feet onto the train bay floor. The electrical non-vital busses supplying offsite power to Unit 1 were damaged, and some of the fire suppression system piping was damaged. The falling stator and crane components caused the supply breaker to Unit 2 reactor coolant pump B to open. The loss of reactor coolant pump B resulted in a Unit 2 reactor trip, which had been operating at 100 percent power. Both units are stable and remain shutdown.

The licensee reported that one worker was killed and eight others were injured when the main generator stator fell. Seven workers have been treated and released from a hospital, while one remains hospitalized.

With the loss of offsite power to Unit 1, both Unit 1 emergency diesel generators (EDGs) started and loaded onto their electrical busses. Decay heat removal was quickly restored. The Unit 1 emergency diesel generators continue to supply power to the vital electrical busses.

At 9:22 a.m. (CDT), offsite power to Unit 2 from startup transformer 3 was lost because water from a fire main caused a short circuit. ANO Unit 2 EDG 2 started and energized the train B vital electrical bus, while train A vital and non-vital electrical busses were re-energized from startup transformer 2. The supply breaker from startup transformer 3 failed because of water intrusion stemming from damaged fire suppression system piping. Operators cooled down Unit 2 to hot shutdown.

At 10:44 a.m. (CDT), the licensee declared a Notification of Unusual Event because the failure of the supply breaker may have been caused by an explosion in the breaker cubicle. The event was terminated at 6:21 p.m. (CDT) because the affected electrical bus was not energized and there was no other damage. The fire suppression system to ANO Unit 1 is shutdown due to the damage to the fire water system piping. Damaged portions of the ANO Unit 2 fire protection system have been isolated. Additional fire water pumps have been positioned to provide fire water if necessary. The licensee has established fire watches in the auxiliary buildings of both units.

The deterministic criteria described in MD 8.3 and MC 0309 was reviewed, and the criteria listed below were determined to be applicable to this event.

Y/N	DETERMINISTIC CRITERIA
	d. Led to the loss of a safety function or multiple failures in systems used to mitigate an actual event
Y	Remarks- The failure of the lift system resulted in the main generator stator damaging the electrical buses supplying offsite power to Unit 1. The damage resulted in Unit 1 losing both trains of offsite power. Both EDGs for Unit 1 started automatically and are supplying power to their buses. The Unit 1 fire suppression system was damaged and during the event and portions of the system are secured. A portion of Unit 2's fire water system was damaged and caused the feeder breaker from a startup transformer to open. This resulted in a partial loss of offsite power to Unit 2.
	e. Involved possible adverse generic implications
Y	Remarks- Nuclear power plants conduct lifts of heavy equipment from time to time. Although unknown at this time, the cause(s) of the failure of the lifting rig could have adverse generic implications.

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CONDITIONAL RISK ASSESSMENT

IF IT IS DETERMINED THAT A RISK ANALYSIS IS NOT REQUIRED - ENTER NA BELOW AND CONTINUE TO THE DECISION BASIS BLOCK

RISK ANALYSIS BY- David Loveless	**DATE-** April 1, 2013

Brief description for the basis of the assessment:

The senior reactor analyst evaluated the risk of this event using the Arkansas Nuclear One, Unit 2, Standardized Plant Analysis Risk (SPAR) model, Revision 8.21, Inspection Manual Chapter 0609, Appendix G, Attachment 2, and other qualitative assessment tools.

The analyst assumed that the event in Unit 2 was similar to an uncomplicated reactor transient with Switchgear 2A2 out of service. The resulting conditional core damage probability (CCDP), 1.1 x 10 -6, indicated the lower bound of the risk from the drop. Assuming that the risk could be bounded on the high side by modeling the event as a plant-centered loss of offsite power, the CCDP was quantified as 1.3 x 10-5.

The analyst used Figure 6 from Appendix G, Attachment 2, to assess the risk of the drop event on Unit 1. The licensee informed the analyst that one of the breakers required to power the vital busses from the alternate ac diesel generator was not available because of potential damage from the event. Therefore, the analyst calculated the probability of an emergency power supply system demand failure at 4.49 x 10-3, assuming that only Diesel Generators 1 and 2 were available to supply vital loads. Given that offsite power had not been restored within 36 hours and was not expected to be returned for some time, the analyst set the probability of failure to restore offsite power to 1.0. The probability of not recovering a postulated diesel generator failure within 18 hours was derived using the SPAR as 3.63 x 10-1. The analyst used a screening value of 0.1 for the probability of alternative strategies failure leading to core damage. The resulting CCDP was 1.6 x 10-4.

The analyst noted that there were several unknown aspects of the event that could affect the risk. These issues are listed below:

Unit 1:

- The failure probability for the emergency diesel generators was set using a probability for failure-to-run for 24 hours. As the diesels are demanded for longer periods of time, the probability of failure to run increases faster than the probability of recovery. Therefore, this probability would suggest an increased CCDP.
- Configuration and operation of the instrument air system would have an impact on the probability of a loss of inventory from the refueling pool.

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CONDITIONAL RISK ASSESSMENT

- Condition Report indicates that there was a loss of all instrumentation related to the nozzle dam seals.
- The ability of operators to vent the containment following boiling in the refueling pool would have an impact on the success of alternative strategies.

Unit 2:

- The condition and structural integrity of the walls and ceiling of the Turbine Building 362 – foot elevation switchgear area can affect the probability of continued offsite power to the unit

Peer Review:

- The analyst's results were peer reviewed and concurred upon by analysts from NRR/DRA/APOB.

Licensee:

- The analyst discussed this analysis with the licensee's analysts. The licensee does not have a shutdown risk model. Their qualitative risk tool indicates that a loss of offsite power while shutdown is a Red condition. During the initial discussion, the licensee stated that the evaluation seemed reasonable for Unit 1 risk, but requested that we give them more credit for the available time to recover power.

- The licensee provided a conditional core damage probability for the Unit 2 event of 2 x 10-7. The licensee did not consider Bus 2A2 to be unavailable, nor did they account for the condition it was in at the start of the event. Additionally, the licensee did not account for the secondary equipment that was not powered because of Diesel 2-2 supplying Bus 2A4.

Note: description may include assumptions, calculations, references, peer review, or comparison with licensee results.

THE CONDITIONAL CORE DAMAGE PROBABILITY (CCDP) IS	**Unit 1: 1.6E-4** **Unit 2: 1.1E-6 - 1.3E-5**
WHICH PLACES THE RISK IN THE RANGE OF	

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RESPONSE DECISION AND BASIS

USING THE ABOVE INFORMATION AND OTHER KEY ELEMENTS OF CONSIDERATION AS APPROPRIATE, DOCUMENT THE RESPONSE DECISION TO THE EVENT OR CONDITION, AND THE BASIS FOR THAT DECISION

Response to the event or condition	Augmented Inspection

BASIS FOR THE RESPONSE

Based on meeting the two deterministic criteria, and the results of the conditional risk assessment (including the uncertainties associated with assessment), I have concluded that the NRC should conduct an augmented inspection at Arkansas Nuclear One. Information gathered during the inspection will be evaluated to determine if an augmented inspection is the appropriate response to this event.

COMPLETED BY	Donald B. Allen	**DATE**	
BRANCH CHIEF REVIEW	Donald B. Allen	**DATE**	
DIVISION DIRECTOR APPROVAL	Kriss M. Kennedy	**DATE**	

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