UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

September 23, 2013

EA-13-186

Mr. Christopher Costanzo, Vice President
Nine Mile Point Nuclear Station
Constellation Energy Nuclear Group, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2 - NRC INTEGRATED INSPECTION REPORT 05000220/2013003 AND 05000410/2013003; PRELIMINARY GREATER THAN GREEN FINDING AND APPARENT VIOLATION

Dear Mr. Costanzo:

On August 13, 2013, the U.S. Nuclear Regulatory Commission (NRC) issued Integrated Inspection Report 050000220/2013003 and 05000410/2013003. This report documented a finding with an apparent violation associated with the Nine Mile Point Unit 1 loss of shutdown cooling (SDC) event which occurred on April 16, 2013, during the unit refueling outage (AV 05000220/2013003-04). Specifically, inadequate Constellation Energy Nuclear Group procedures for restoration following an unexpected loss of direct current control power resulted in an unplanned loss of all SDC when time to boil was less than 2 hours. The subject inspection report also indicated that the significance of the finding was still under evaluation by the NRC and was to be determined (TBD). We note that there was no actual safety consequence to the event, because the operators restored SDC in a timely manner, maintaining sufficient margin to boiling.

This finding has preliminarily been determined to be Greater Than Green, a finding of greater than very low safety significance, that may require additional NRC inspections, based on the best available information, using the NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)." The enclosed best estimate analysis, given that the event occurred early in the outage while no automatic injection systems were available and the primary containment was not functional, preliminarily determined that the increase in core damage frequency (CDF) and a large early release frequency (LERF) of radioactive material were in the range of one in one million years and one in ten million years, respectively. The analysis estimated the chance that operators could fail to restore SDC and then also fail to add water to the reactor coolant system to account for the amount that would have boiled, prior to core damage. The most influential input for both CDF and LERF was use of current SDP risk analysis guidance which limits the combination of human error probabilities to a chance of one in one million. Relating to LERF, assumptions concerning the point at which the evacuation of the population close to the plant would be initiated relative to the time of core damage were most influential. The finding is also an apparent violation of NRC requirements, and is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy, which can be found on the NRC's Web site at http://www.nrc.gov/aboutnrc/regulatory/enforcement/enforce-pol.html.

In accordance with IMC 0609, we intend to complete our risk evaluations using the best available information and issue our final significance determination within 90 days of the date of the subject inspection report. To refine the risks associated with this finding, additional insights concerning: the potential operator and plant staff cues and responses to the postulated scenario; the NRC guidance which limits the combination of human error probabilities; and LERF considerations concerning the timing, composition and quantity a potential radioactive material release and the timing of close in population evacuation would be beneficial to the NRC. The SDP encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before the NRC makes its final decision on this matter, we are providing you with an opportunity to: (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of IMC 0609.

Please contact Daniel Schroeder at (610) 337-5262 within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will issue our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence. Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public

Document Room or from the NRC's Agencywide Documents and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Darrell J. Roberts
Director
Division of Reactor Projects

Docket Nos. 50-220 and 50-410
License Nos. DPR-63 and NPF-69

Enclosure:
Risk Significance Determination

cc w/encl: Distribution via ListServ

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Preliminary
Phase 3 Risk Assessment
Loss of Shutdown Cooling
Nine Mile Point Unit 1

Probabilistic Risk Assessment (PRA) Analyst:	Jeff Mitman, Senior Reliability and Risk Analyst, NRR/DRA/APOB
Independent Reviewer	Jeff Circle, Senior Reliability and Risk Analyst, NRR/DRA/APOB
Region 1 Reviewer	Wayne Schmidt, Senior Reactor Analyst, DRS
Region I Inspector	Andrew Rosebrook, Senior Project Engineer, DRP

1.0 Introduction

On April 15, 2013 Nine Mile Point Unit 1 (NMP1) shutdown for a refueling outage. On April 16th the unit was in cold shutdown with the water level at the reactor flange. The reactor head vent piping had been removed and work was ongoing to detension the reactor vessel head. Simultaneous to this work, the station was conducting a Loss of Coolant Accident/Loss of Offsite Power (LOCA/LOOP) test. In addition, preparations were being made to start work on an electromatic relief valve (ERV) modification.

A contractor verifying the safety tagout on the ERV modification opened the breaker cabinet for the vital DC bus 12 (at 14:44). This was the wrong division. The vital 125 volt DC bus 12 cabinet door contains a mechanical interlock which opens battery breaker 12 and the static battery charger DC output breaker. This de-energized the DC switchgear when the door was opened. Upon opening the cabinet door and hearing breakers trip, the contractor realized he was in the incorrect cabinet and immediately contacted the control room. The loss of DC power generated an invalid SDC pump 12 high temperature pump breaker trip signal, but without DC control power the breaker did not trip and the pump continued to run and to cool the core.

Operators failed to recognize the invalid #12 SDC Pump trip signal present on the alarm log and the plant process computer displays prior to attempting to restore bus 12. The presence of the trip signal was also indicated by a control room annunciator which was locked in since the loss of battery bus 12 at 14:45.

Two unsuccessful attempts were made by operations to re-energize the bus at 15:03 and 15:05 by closing battery breaker 12. A third attempt was initiated at 15:46 utilizing a different method, which used battery charger 171. This momentarily energized the DC system allowing the previously created SDC pump trip signal to trip the running SDC pump 12. The DC system then tripped again leaving the system de-energized.
Per the licensee's time line, the operator at the controls took several minutes to identify the loss of SDC as there were no additional alarms received when the pump tripped. The operator did identify the loss of SDC by observing "lowering" reactor building closed cooling (RBCLC) temperature. Temperature was decreasing because no heat was being transferred from the SDC system to the RBCLC system.

The operators restore SDC by racking in the breakers for SDC pumps 11 and 13 and starting those pumps. SDC was restored when at 16:17 when the SDC temperature control valve (38-09) was opened.

2.0 Discussion of the Performance Deficiency

The inspectors determined that the failure of Constellation Energy Nuclear Group (CENG) to establish an adequate procedure for properly restoring the battery bus 12 was a performance deficiency that was reasonably within CENG's ability to foresee and correct and should have been prevented. The performance deficiency was determined to be more than minor because the inspectors determined it affected the configuration control aspect of the Initiating Events cornerstone and adversely affected the associated cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, CENG failed to specify the associated tripping circuits and tripping actions that could result from battery bus restoration in accordance with N1-SOP-47A.1, "Loss of DC," Revision 00101, and N1-OP-

47A, "VDC Power System," Revision 02500. This performance deficiency resulted in loss of shutdown cooling during attempted restoration of the vital DC bus 12 on April 16, 2013.

A second performance deficiency related to the loss of the #12 DC bus was also identified. Specifically, the contractor did not follow station procedures for control of maintenance by failing to verify he was on the proper equipment, and station personnel did not implement all risk management actions for protected equipment as directed by station risk management procedures. The performance deficiency was determined to be more than minor because the inspectors determined it affected the configuration control aspect of the Initiating Events cornerstone and adversely affected the associated cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

3.0 Plant Conditions Prior to the Event

Plant equipment and conditions were as follows:

- Shutdown cooling (SDC) pump 12 and associated support systems were inservice cooling the core. This was the "protected" train
- SDC pumps 11 and 13 were non-functional with their breakers racked out. However, the pumps were available with manual actions outside the control room
- Reactor water cleanup was inservice letting down at approximately 40 gpm
- Condensate system was inservice making up to the reactor at approximately 40 gpm
- Both trains (all four pumps) of core spray were non-functional, i.e., the pumps would not auto start on low reactor water level. However, the pumps could be started with manual actions outside the control room
- Both control rod drive pumps were non-functional but could be manually started with operator actions outside the control room
- Fire water system was available
- Containment spray (including torus cooling) was non-functional but could be started with actions outside the control room
- Containment Spray Raw water system was available
- Reactor water level was at the flange
- Reactor head vent piping was removed, thus ensuring that the reactor coolant system would not pressurize on a loss of shutdown cooling
- Estimated time to boil prior to the event was calculated to be less than 2 hours. The heat up from the actual event, 115 F to 145 F in approximately 30 minutes, indicates an actual time to boil of about 110 minutes
- Estimated time to core uncovery (the surrogate for core damage) was about 9 hours.
- Primary containment was open and not restorable

4.0 Licensee Event Mitigation Capability

At the time of the event, the condensate system was available from the control room. To the best of the analyst's knowledge no other core cooling or injection systems other than those actively cooling the core were in their normal status, see the list above. However, many systems were available with operator actions outside the control room. The actions needed to make these systems available differed for each system or sub-system and would require locally changing breaker and valve positions. These included:

- SDC pumps 11 and 13
- All four core spay pumps and their associated topping pumps
- Containment spray (and its associated torus cooling function)
- Fire water
- Control rod drive pumps
- Raw water injection into the RCS via the core spray system

However, it should be noted that portions of the above systems utilize DC control power from the DC bus 12, and those that are supported by DC bus 12 could not be used without first correcting the DC control power problem.

5.0 Significance Determination Process (SDP) Phase 1 and 2 Summary

Phase 1 Screening Logic:

The inspectors evaluated the finding using IMC 0609 Attachment 0609.04, "Initial Characterization of Findings," issued June 19, 2012, and IMC 0609 Appendix G, "Shutdown Operations Significance Determination Process," issued February 28, 2005. IMC 0609 Appendix G Table 1, "Losses of Control," states a quantitative analysis is required for:

- Loss of Thermal Margin (PWRs and BWRs)

(Inadvertent change in RCS temperature due to loss of RHR)/(change in temperature that would cause boiling) > 0.2 (temperature margin to boil)

In this case, RCS temperature changed 27 degrees (118 to 145 degrees Fahrenheit) and the change in temperature to boiling was 94 degrees (212 to 118 degrees Fahrenheit). Temperature margin to boil was greater than 0.2 (0.2872); thus, a quantitative analysis was required.

Phase II Screening Logic:

The Shutdown SDP proceduralized in IMC 0609, Appendix G, is used to screen shutdown findings for potential significance. This finding could not be screened as having very low significance using the Phase 2 analysis.

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 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

7.0 Development of the Model

No Low Power/Shutdown (LP/SD) SPAR model exists for NMP1. Therefore, the at-power NMP1 SPAR model was modified to allow analysis of the loss of shutdown cooling 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.

Three specific changes were made to the model to ensure that it appropriately reflected actual plant conditions. First, the failure of automatic start of the core spray pumps was set to logical TRUE to reflect that the auto start had been defeated for conduct of the LOCA/LOOP test. The significance of this is that the model now requires a manual start of one or more core spray pumps.

Second, as the loss of shutdown cooling was caused by a loss of power to 125V dc battery board 12 the representative model basic event (DCP-BDC-LP-12: Failure of Division II125vdc Bus 12) was set to logical TRUE. Although this bus was recovered in approximately one hour after the event initiation, no consequent automatic actuation would have occurred to correct the condition of a loss of shutdown cooling after recovery of 125V dc. Modeling subsequent 125V dc recovery as a sensitivity case found that this had a small effect and did not change the conclusions.

Finally, the impact of all test and maintenance unavailability events in the model was set to logical FALSE to reflect that maintenance during the outage is highly managed by the operating staff.

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 cutsets 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 1 – Internal Events*, Revision 1 Section 9.4. The RASP Manual states: “An analyst should not use a minimum joint HEP of less 1E-6 for SDP analysis. Therefore, a SDP analysis always assumes some level of dependence between HFEs even if the specific reason for that dependence cannot be identified.”

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 and Restore before Boiling to Scram Setpoint (Level 3)	30 minutes	5 hours	2.0E-5	4E.0E-4	4.2E-4
SD-XHE-XL-MINJ	Operator Fails to Diagnose and Take Action to Inject Prior to Core Damage	5 minutes to 1 hour (Note: 1)	5 hours	2.0E-5	2.0E-5	4.0E-5
OPR-XHE-XM-LTCST	Operator Fails To Replenish CST In The Long Term (after successful injection)	Several hours (Note:2)	Many hours	(Note 3)	1.0E-4	1.0E-4
SD-XHE-XL-SDC-LATE	Operator Fails to Recover Train of SDC Late (after successful injection)	Several hours	Many hours	(Note 3)	2.0E-5	2.0E-5

Note 1: Time needed to take action depends on the system used. If the condensate system is available it will take 5 minutes, if one of the systems that require racking in breakers and venting the system then it would take about 1 hour.
Note 2: This value is based on a preliminary SPAR-H re-evaluation with nominal PSF with the exception of time which is judged to be “extra.”
Note 3: Analyst assumes no diagnosis is required as operator has successfully determined that injection and makeup is required and has successfully completed this action.

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 Attachment 2 for Phase 2 analysis. 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 shutdown cooling to 1.0 and all other IEF to zero. The above described SPAR model was evaluated using the SAPHIRE code version 8.0.8.0.

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 1.05E-6; based on these results the finding is White. The top 20 cutsets for both contributing sequences (sequences 3 and 5) are in Appendix C. The analyst did not perform uncertainty analysis.

Table 3
CCDP Results

Sequence Name	**Point Estimate**	**Cut Set Count**
LOSDC:3	0	0
LOSDC:5	1.90E-8	124
LOSDC:6	1.03E-6	152
Total	**1.05E-6**	**276**

In this shutdown event analysis, as is the case with most shutdown analysis, the results are highly dependent on operator actions significantly more so than a typical at-power analysis. As summarized above, PRA practices direct that in cutsets with multiple HEPs, a justifiable minimum value for the combination of HEPs should be specified. NUREG-1792, "*Good Practices for Implementing Human Reliability Analysis*," Section 5.3.3.6 recommends a cutoff of 1E-5. The analyst did not implement this cutoff value, instead (as discussed above) the analyst used a cutoff value of 1E-6 consistent as specified by the RASP Manual for those sequences that had a duration less than 24 hours.

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 4.5 (BWR Mark I and II Containments Event Tree) from NUREG/CR-6595 is applicable to the NMP1 event.

The LERF event tree is in FigureC-1 of Appendix C. The review of the LERF event indicates a negative response for all of the event tree tops with one exception. A detailed review of the LERF event tree and each of the event tree tops is also presented in Appendix C. The questions that are answered in the negative include:

- Greater than 8 days after shutdown: It was only 1.5 days after shutdown
- Containment Isolated and not bypassed: Primary containment was open and not closable
- Containment inerted: No
- Water on drywell floor: No (but not relevant – see detailed discussion in Appendix C)
- Core damage arrested without vessel breach: Vessel head vent piping was removed therefore, by definition the vessel was breached
- No containment failure at vessel breach: Containment was open as the containment (drywell) head was removed and the head vent piping removed
- No venting after vessel breach: Containment was vented as the containment (drywell) head was removed and the head vent piping removed

The two questions that determine the LERF conclusion relate to the possibility of evacuating the near-in population before core damage and containment failure. As the containment is failed because of the containment head removal the question that needs to be resolved is the probability of evacuating the near-in population. The licensee determined that the emergency plan requires the declaration of a general emergency upon core uncovery, and that under the existing weather conditions at the time of the event it would take approximately two hours to evacuate the near-in population. The analyst has determined that the time to boil off the water to the bottom of active fuel is approximately two hours. Therefore, the time from core uncovery to core damage is less than two hours and therefore, the entire near-in population, as currently understood, cannot be evacuated.

Thus the LERF multiplier is estimated to be between 0.1 and 0.9. LERF is then calculated by multiplying the CDF by the LERF multiplier. Assuming a LERF multiplier of 0.5, and a CDF of 1.1E-6, yields a LERF of 5.5E-7 which is a White finding. It should be recognized that a closer scrutiny of LERF could alter these results.

Finally, it should be noted that with the "no cutoff" CDF of 6.1E-8 the LERF results are Green.

10.0 Sensitivity Analysis

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

Case 1: No HRA Cutoff

The base imposed an HRA HEP cuoff for multiple HEPs in a single cutset of 1E-6. This case evaluates the CCDP without a cutoff. Under these conditions the CCDP is 6.1E-8.

Case 2: HRA Cutoff of 1E-7

The base imposed an HRA HEP cutoff for multiple HEPs in a single cutset of 1E-6. If a cutoff of 1E-7 is imposed, the CCDP is 1.44E-7.

Case 3: DC Bus 12

In this sensitivity case DC Bus 12 failure probability was left in its default random failure value instead of set to logical TRUE as it was in the base case. The resultant CCDP is 1.0E-6.

11.0 Comparison with Licensee Results

As can be seen from the below table which compares the results of the licensee and the NRC the results are within about a factor of two for all available cases.

Table: Comparison of CCDP Values for Loss of SDC Event

Case	NRC CCDP	Licensee CCDP
Alignment of SDC and RPV injection HEP Values Completely Independent (no cutoff)	6.1E-8	5.6E-8
1E-6 Assigned as Joint HEP for Alignment of SDC and RPV injection	1.1E-6	1.1E-6
5E-7 Assigned as Joint HEP for Alignment of SDC and RPV injection	5.4E-7	5.6E-7
1E-7 Assigned as Joint HEP for Alignment of SDC and RPV injection	1.4E-7	1.6E-7

The licensee contends that there are numerous systems available to inject into the RCS. Accordingly, the NRC analysis takes into account the following systems:

- SDC pumps 11 and 13
- All four core spay pumps Containment spray (and its associated torus cooling function)
- Fire water
- Control rod drive pumps
- Raw water injection into the RCS via the core spray system

However, because of the chosen equipment configuration, none of these systems would automatically inject on low reactor level. Thus they are all dependent on a human intervention. This is reflected in the NRC risk analysis.

The licensee documents the numerous cues available to alert the operators of the event and of further degradation of plant conditions if operators did not diagnose and address the event early. There were many cues available and these were credited appropriately in the low diagnoses HEPs (see Table 1). The largest diagnoses HEP is 2E-5 applied in the analysis. This compares with a nominal diagnostic HEP of 1E-2. As noted in the introduction the operating crew in the control did initially miss several cues. They failed to recognize the 1) bus 12 failure signal present on the alarm log, and 2) the plant process computer displays, and 3) SDC pump 12 high temperature trip signals which were indicated by a control room annunciator. This illustrates that it is possible to miss cues.

The licensee requested credit for the additional personnel that were on site for the outage work since there were a total of 7 SROs and 7 ROs at the time of the event. It should be noted also that those other personnel are assigned to the outage unit and are given specific tasks to perform. However, NUREG/CR-1278,"Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (THERP)" talks directly to this point. It states in part in Section 18 (Page 18-7):

> "The four people (RO, SRO, SS, STA) listed above are the minimum that would be available to cope with an abnormal event, since there would be another RO available. However, he could be performing duties elsewhere in the plant, so we are assuming only the initial group of four personnel for PRA purposes. We recognize that additional qualified people may become available with time, but we do not know how to assess their influence independent of plant specifics and the characteristics of the event. Their presence may or may not help cope with the event. The Rogovin report describes instances in the TMI incident in which incorrect diagnoses were still being made more than 2 hours into the event even though several additional qualified personnel were present. For PRAs we have performed, we have given credit only for the above four persons."

This guidance is dated (1983). However, the analyst knows of no other guidance on when, how or if to credit additional personnel.

Finally, it should be noted that the event occurred because of poorly written operating and off normal procedures and that the operators tried 3 times to re-energize the DC bus 12 eventually causing the running SDC pump to trip. Clearly mistakes can and were made and more mistakes could have been made. There is a large amount of time to recover from these or other modeled mistakes and the probability of these mistakes occurring without recovery is indeed small. The analyst believes, based on his experience and judgment that this probability is on the order of one in a million in the 9 hours available.

In addition to the quantitative assessment the licensee performed a qualitative assessment of the event and their response capability. The NRC analyst performed a cursory deterministic analysis against Regulatory Guide (RG) 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2. In Section 2.1.1 this RG supplies a concise definition of defense-in-depth. Each of the defense-in-depth attributes is discussed below.

- A reasonable balance is preserved among prevention of core damage, containment failure and consequence mitigation: As the event has already initiated and the containment is open (a requirement in BWR Mark I containments to refuel) all of the reliance is on mitigation.

- Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided: The only way to mitigate this event was a reliance on programmatic activities (i.e., operator action).

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges, and uncertainties: System redundancy and diversity were maintained. However, because of the loss of DC bus 12, an entire train of safety related equipment was lost to the operators initially. This lost equipment was recoverable but it took the operators over an hour to do so. Based on operating experience, losses of shutdown cooling occur about once per shutdown year. Therefore, these are not rare events and should be anticipated.

- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed: By removing all automatic actions a common cause failure – failure of the operators – was introduced.

- Independence of barriers is not degraded: Primary containment was open and the RCS was breached. Thus two out of the three barriers were not available prior to the start of the event. Because of the configuration, that is the plant was in a refueling outage, this was unavoidable.

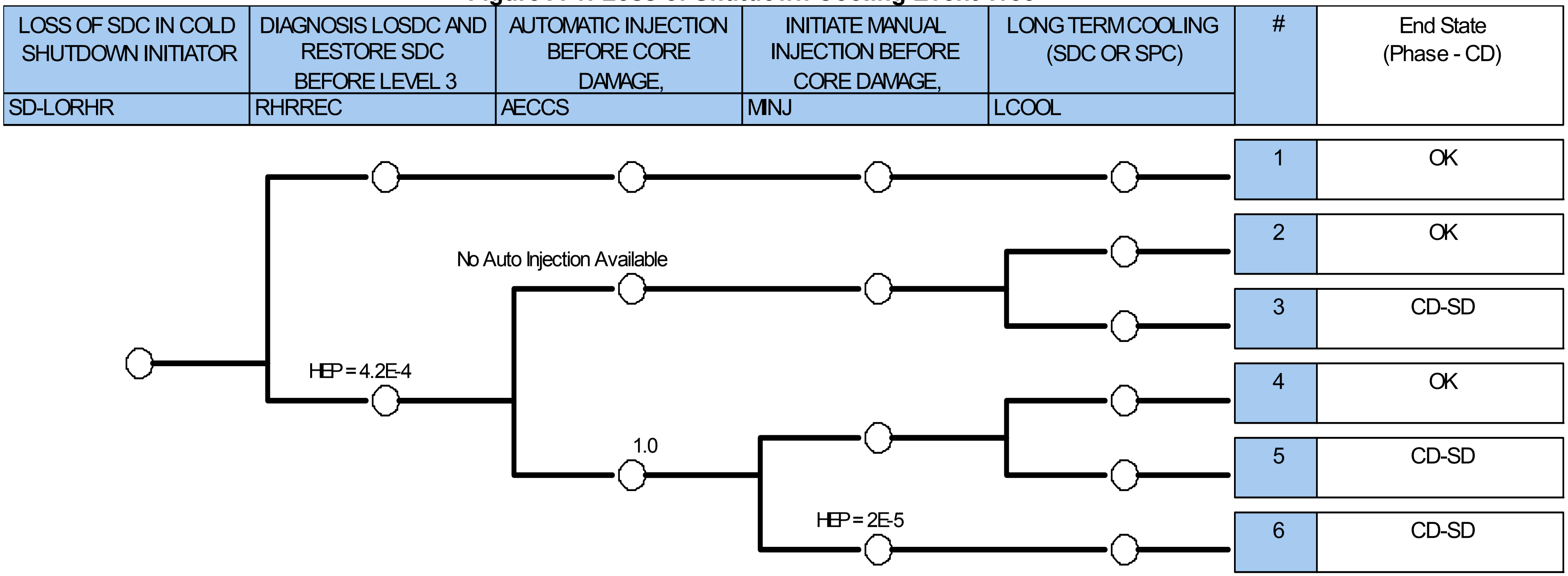
- Defenses against human errors are preserved: This was not maintained.

- The intent of the plant's design criteria is maintained: Indeterminant.

From the review above it is clear that many of the defense in depth criteria capabilities were not strong during the event. The other aspect of a proper deterministic analysis includes a safety margin review. Based on the RG 1.174 definition this event does not appear to challenge the plant's applicable safety margin.

Appendix A: Model Figures

Figure A-1: Loss of Shutdown Cooling Event Tree



Note: This event tree top MINJ for core spray auto injection always fails because the plant inhibited the auto start capability for testing

Figure A-2: Diagnoses and Restoration of SDC Fault Tree

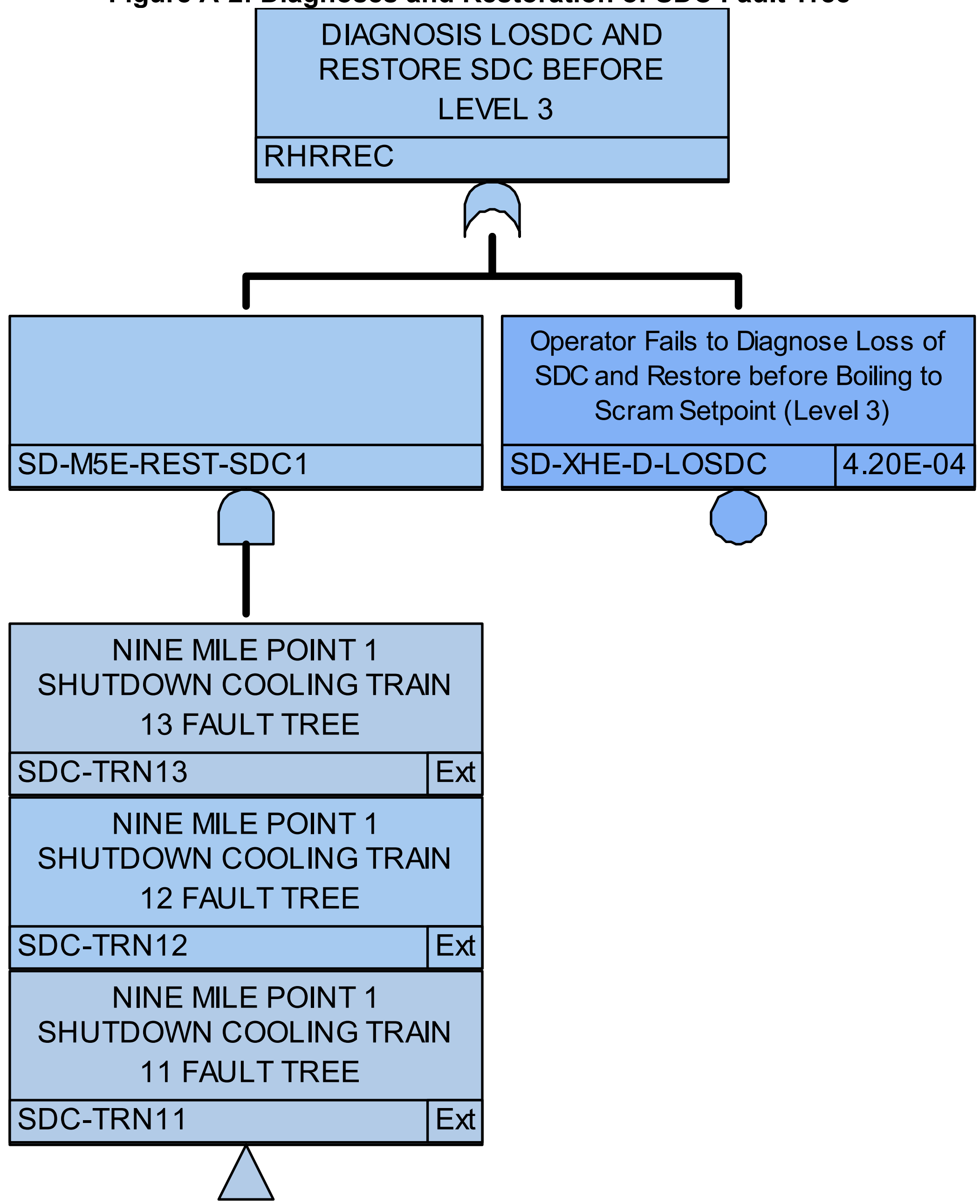


Figure A-3: Core Spray Auto Injection Fault Tree

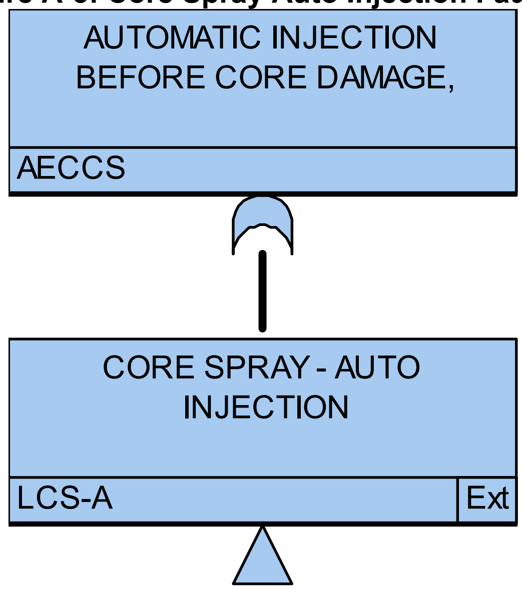
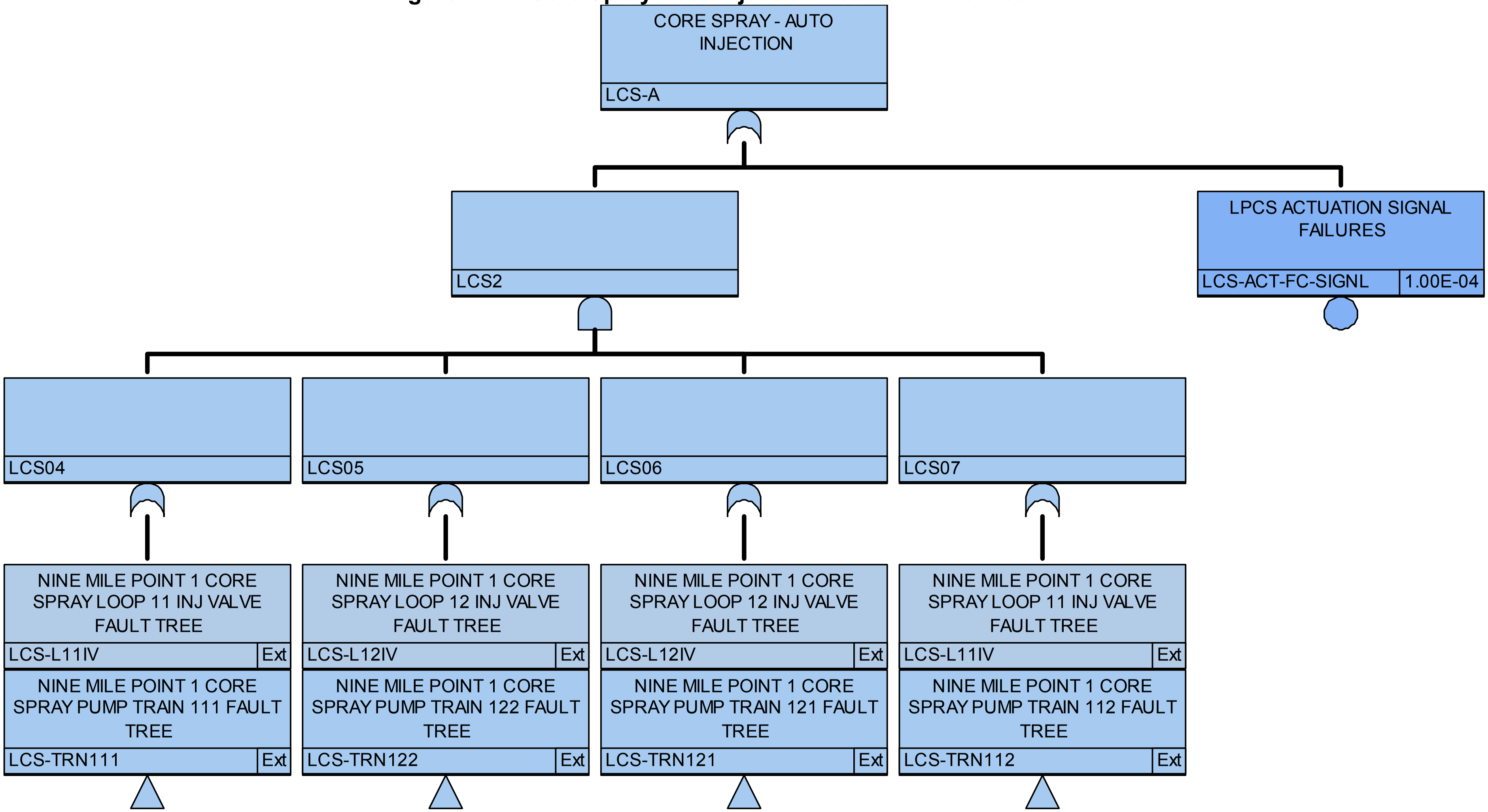


Figure A-4a: Core Spray Auto Injection Detailed Fault Tree



Note: This fault tree always fails because the plant inhibited the auto start capability for testing.

Figure A-4b: Initiate Manual RCS Injection Fault Tree

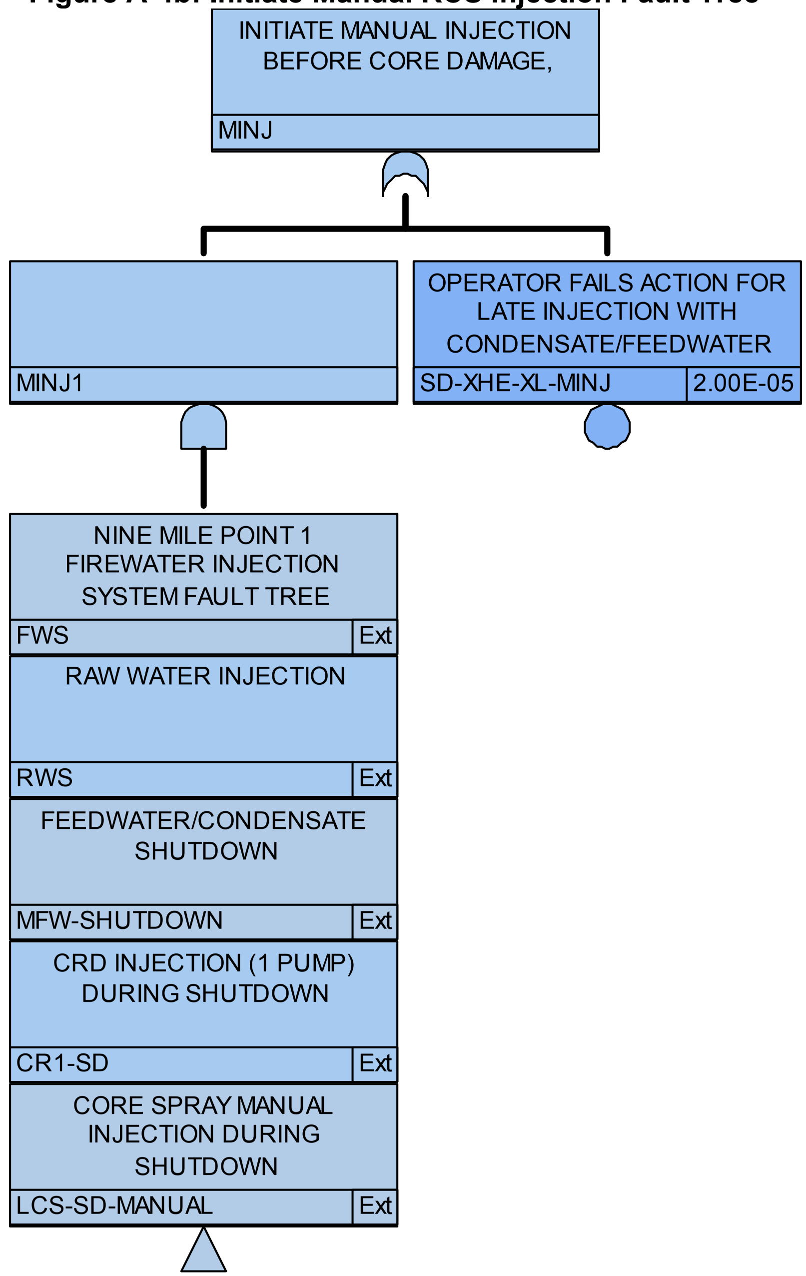


Figure A-5: Long Term Recovery after Successful Injection Fault Tree

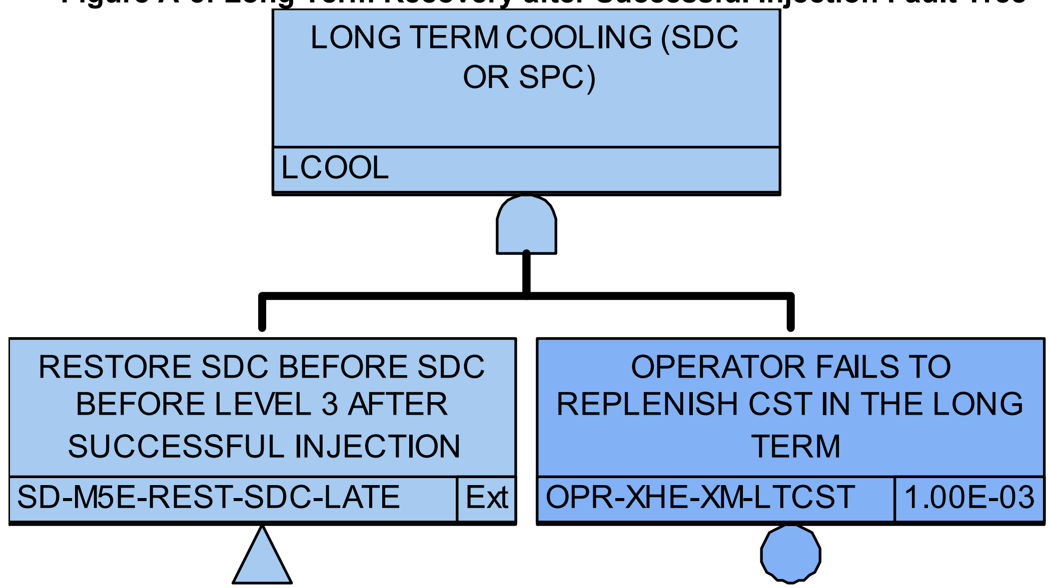
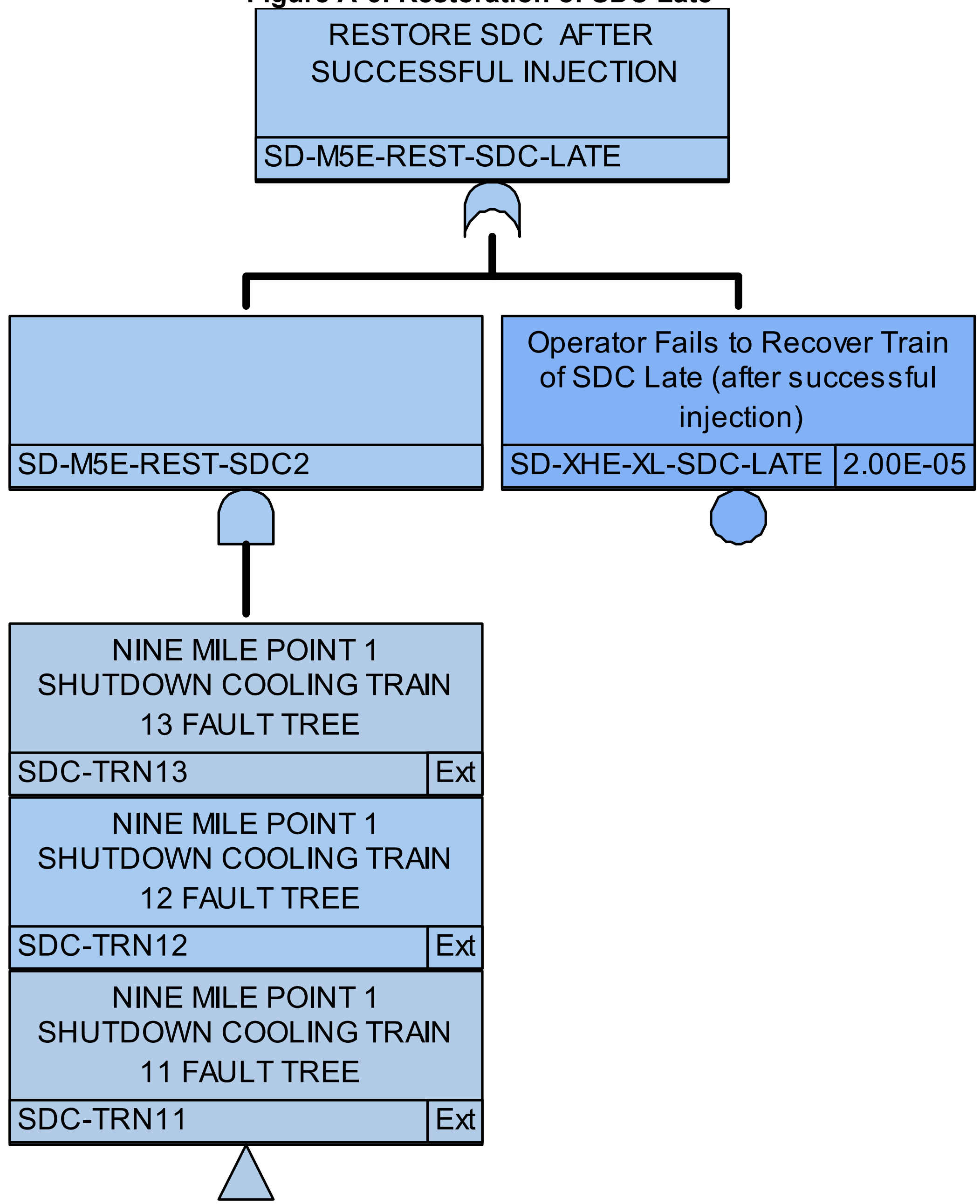


Figure A-6: Restoration of SDC Late



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.

B1a Operator Fails to Diagnose Loss of SDC before Boiling to Scram Setpoint

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: NMP1 Initiating Event: Basic Event: SD-XHE-D-LOSDC

Basic Event Description: Operator Fails to Diagnose Loss of SDC and Restore before Boiling to Scram Setpoint (Level 3)

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		Time to initiate boiling and boil down to scram setpoint. Estimated time is 5 hours. Diagnosis is assumed to take 5 or 10 minutes.
	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		Pump stop is obvious
	Moderately Complex	2		
	Nominal	1		
		0.5		
	Obvious diagnosis	**0.1**	**X**	
	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		
		Final Diagnosis HEP	**2.0E-05**	

B1b Operator Fails to Restore Loss of SDC before Boiling to Scram Setpoint

SPAR HUMAN ERROR WORKSHEET

Plant: NMP1 Initiating Event: Basic Event: SD-XHE-D-LOSDC
Basic Event Description: Operator Fails to Diagnose Loss of SDC and Restore before Boiling to Scram Setpoint (Level 3)

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 to begin boiling and boil down to scram setpoint. Estimated time is 5 hours. Action to restart RHR/SDC took the operators 30 minutes.
	Time Available is ≈ the time required	10		
	Nominal time	1		
	Time available is ≥ 5x the time required	**0.1**	**X**	
	Time available is ≥ 50x the time required	0.01		
	Insufficient information	1		
Stress	Extreme	5		
	High	**2**	**X**	
	Nominal	1		
	Insufficient information	1		
Complexity	Highly	5		Starting RHR/SDC is rarely of nominal complexity. For the NMP1 case operators had to rack in SDC pump breakers.
	Moderately	**2**	**X**	
	Nominal	1		
	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		
		Final Action HEP	4.00E-04	

B2a Operator Diagnose Need to Inject after Level Reaches Scram Setpoint and before it 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 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		There is approximately 5 hours to recognize cues and make decision should take about 5 minutes.
	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		Stress is elevated
	High	**2**	**X**	
	Nominal	1		
	Insufficient information	1		
Complexity	Highly	5		Scram setpoint is an obvious cue
	Moderately Complex	2		
	Nominal	1		
		0.5		
	Obvious diagnosis	**0.1**	**X**	
	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		
	Final Diagnosis HEP =		**2.0E-5**	

B2b Operator Fails Acton to Inject after Level Reaches Scram Setpoint and before it 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		Easiest action is to simply increase the existing condensate flow rate. This is a 5 minute action and over 4 hours are available.
	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		Stress is high
	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		
		Final Action HEP	**2.0E-05**	

B3a Operator Fails Acton to Inject after Level Reaches Scram Setpoint and before it Reaches TAF

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

Plant: NMP1 Initiating Event: Basic Event: SD-XHE-XL-SDC-LATE

Basic Event Description: Operator Fails to Recover Train of SDC Late (after successful injection)

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		This task takes place after successful injection has been accomplished. Time available is the time it will take to deplete the water that is being used for injection. This is many hours.
	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		
	Nominal	1	X	
	Insufficient information	1		
Complexity	Highly	5		This requires restoring SDC which requires both in control and ex-control room actions including racking in breakers and realigning the system
	Moderately	**2**	**X**	
	Nominal	1		
	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		
		Final Action HEP	**2.0E-05**	

Note: No diagnose worksheet is supplied as the analyst determined that because this action follows a successful HEP to diagnose and take action to inject the operators are aware that they now have to restore shutdown cooling.

Appendix C: LERF

Containment Event Tree
See Figure D-1 below for the event tree. All up branches are evaluated by answering the top event yes and all down branch by answering no. All of the quoted material in the discussion is directly from NUREG/CR-6595.

Question 1: Core damage?
Core damage is predicted at a low frequency per shutdown internal events analysis.

Question 2: No potential for large early release based on time after shutdown?

> "For accidents that occur a certain time, i.e., number of days, after shutdown, it is possible that the core inventory may have decayed to a level low enough such that releases from an accident lead to offsite doses that are below the threshold for an early fatality. This cut-off time for LERF after shutdown will, in general, depend on a number of plant, site, and accident specific factors. These factors include: (1) plant size and burnup, i.e., factors that impact the magnitude of the total core inventory at scram, (2) site weather factors affecting transport and dilution of the release, (3) accident source terms such as fractions of the core inventory of different radionuclides released in the accident, especially radionuclides such as iodine and tellurium that are relatively volatile and have a large impact on early health effects, (4) the timing and duration of the release, and (5) the energy and height of the release. Since a large number of factors affect the calculation of LERF, only some very general guidance on the LERF cut-off time can be provided here. For many releases characteristic of severe accidents, 8 days after shutdown is a reasonably conservative estimate of the LERF cut-off time; however, if releases of very large source terms are involved the time could conceivably extend over the entire outage. Alternatively, if it is desired to take credit for a LERF cut-off time of less than 8 days, justification should be provided through an appropriate Level 3 probabilistic consequence calculation. It is recognized in this context that the recent practice of the nuclear power industry has been that the duration of refueling outages is becoming shorter."

Time of initiation of the event is approximately 1.5 days after shutdown. Therefore, this is evaluated as a down branch on the event tree.

Question 3: Does core damage occur in a time frame with a potential for early fatality?

> "This question relates to the type of core damage accident that occurs. CD accident sequences that occur within a time frame such that evacuation of the close-in population is possible are assumed not to have the potential for a large early release. The time available for evacuation is the time from declaration of a general emergency** to the onset of core damage. For the purpose of screening core damage accident sequences, no credit is given for evacuation beyond the onset of core damage, regardless of the initial status of containment isolation. CD accident sequences that occur in a time frame such that an effective evacuation of the close-in population is not possible have the potential for a large early release. Due to the fact that human errors are often important contributors to core damage accidents during shutdown, and may impact evacuation timeliness, the potential of delayed evacuation has to be taken into consideration. In particular, if core damage was caused by diagnostic errors, no credit for evacuation should be taken."

At this time the evaluation of this top event cannot be evaluated. However, this does not matter as the only other top event that is not immediately evaluated in the negative in the associated branches is top event 10.

Question 4: Containment isolated or not bypassed?

Dry well head is removed in preparation for refueling; therefore, containment is open.

Question 5: Is the containment inerted?
No.

Question 6: Is the drywell floor flooded?
This question does not need to be evaluated based on the applicable paths through the event tree.

Question 7: Core damage arrest without vessel breach (VB)?
The reactor head vent piping is removed thus the vessel is breached in this scenario.

Question 8: Containment failure at vessel breach (VB)?
This question does not need to be evaluated based on the applicable paths through the event tree. However, the dry well head which is the containment head is removed in preparation of refueling therefore, containment is failed.

Question 9: No venting after vessel breach?
This question does not need to be evaluated based on the applicable paths through the event tree.

Question 10: No potential for early fatalities? Report sends the analyst to full power PWR question 7 (Section 2.1 page 2-6)

> "This question addresses whether or not early fatalities are likely given a loss of containment integrity. The potential for early fatalities depends on the magnitude and timing of the radionuclide release. The magnitude of the release is important because there is a threshold below which the doses from the early exposure pathways will be unlikely to cause an early fatality. This threshold is discussed in more detail in Appendix A to this report. The timing of release is also important because of radionuclide decay and because of its relation to the time required for evacuation of the close-in population around a nuclear power plant.
>
> Accident sequences that feed into this question have a flow path out of containment that is sufficiently large so that early health effects are likely. In order to respond to this question, the time from the declaration of a general emergency to the time of the start of the release has to be determined and compared to the time required to effectively warn and evacuate the population in the vicinity of the plant. In some accident sequences, containment failure occurs hours after the declaration of a general emergency giving time for evacuation of the population. However, for other accident sequences loss of containment integrity occurs prior to or closely after the start of core damage allowing relatively short times for evacuation."

As the dry well (containment) head is removed, this is evaluated as a containment failure. Thus radionuclide release from containment happens at the same time as core damage. The time of declaring an emergency in relationship to the predicted core damage of nine hours. The assumption of when the emergency response organization would order an evaluation of the population closest to the plant becomes a major factor is determining what LERF factor values to use. A protective action recommendation is not made by this site under General Emergency; however, the state/county may order evacuations when the Site Area Emergency level is reached.

Figure C-1: Containment Event Tree