UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

September 19, 2013

EA-13-129

Mr. William R. Gideon
Vice President
Carolina Power and Light Company
H. B. Robinson Steam Electric Plant, Unit 2
3581 West Entrance Rd
Hartsville, SC 29550

SUBJECT: H.B. ROBINSON STEAM ELECTRIC PLANT - FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING, NOTICE OF VIOLATION, AND ASSESSMENT FOLLOW-UP LETTER - NRC INSPECTION REPORT 05000261/2013009

Dear Mr. Gideon:

This letter provides the final significance determination of the preliminary White finding discussed in our previous communication dated July 1, 2013, in inspection report 05000261/ 2013008. The finding involved the failure to perform adequate preventive maintenance on the dedicated shutdown diesel generator (DSDG) cooling system, in accordance with vendor recommendations and as required by procedure PLP-018, "Quality Assurance Program for Non-Safety Systems and Equipment used to meet the Station Blackout rule" and Fire Safe Shutdown (SSD) equipment maintenance requirements. During surveillance testing of the DSDG on October 2, 2012, the DSDG automatically shut down on high engine temperature due to a failure of the radiator drive belts. The condition of the drive belts was significantly degraded due, in part, to a lack of adequate inspection, maintenance, and/or periodic replacement, which may have rendered the plant unable to cope for eight hours after a postulated station blackout, or provide emergency power for certain selected Fire Safe Shutdown (SSD) scenarios.

At your request, a Regulatory Conference was held on August 19, 2013, to discuss your views on this issue. A copy of the information presented at this meeting by Carolina Power and Light Company (doing business as Duke Energy Progress (DEP)) is included in the Public Meeting Summary dated August 23, 2013 (ML13238A054). During the meeting, your staff agreed with the apparent violation, and described your assessment of the significance of the finding and the corrective actions taken, including the root cause evaluation. You also presented your bases for considering the finding to be of very low safety significance (Green). You indicated that plant design and configuration of dedicated shutdown cables/high energy arc fault sources make loss of the dedicated shutdown bus unlikely. Specifically, the impact of a 480V high energy arc fault (HEAF) is smaller when compared to a 4160V high energy arc fault. In addition you indicated that recovery credit should be higher than the NRC previously considered. Specifically, the operators would have established and maintained steam generator feed and cool down rates before dedicated shutdown battery depletion and maintained the heat sink after battery depletion. You also stated that repair and recovery of the DSDG belt is straight forward with simple tools.

The differences between your characterization and NRC's characterization of the significance of the finding are discussed in detail in Enclosure 2. In summary, for the HEAF discussion, the NRC agrees that the impact of a 480V HEAF would likely be less than a 4160V HEAF. However, the likelihood of damage outside the cabinet cannot be neglected and industry guidance does not support eliminating such a consideration. For purposes of characterizing the risk associated with this finding, the risk of the E1 HEAF was not significant to the risk outcome of the finding. The NRC's final risk significance remained unchanged without including the risk associated with the E1 switchgear HEAF. Concerning a higher recovery credit, the NRC determined that recovery credit should be considered for the SBO and fire scenarios. Using a recovery event tree approach and repair probabilities taken from NUREG/CR 6890 Volume 2 and human error probabilities developed using the standardized plant analysis risk (SPAR) human reliability model, the NRC final recovery credit continued to result in a White finding.

After considering the information developed during the inspection and the information you provided at the Regulatory Conference, the NRC has concluded that the finding is appropriately characterized as White, low to moderate safety significance.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in IMC 0609, Attachment 2. An appeal must be sent in writing to the Regional Administrator, Region II, 245 Peachtree Center Avenue NE, Suite 1200, Atlanta, Georgia 30303-1257.

The NRC has also determined that the failure of the DSDG to run and provide the only source of AC power during a station blackout for the exposure period was a violation of 10 CFR 50.63 (c)(2), Loss of all Alternating Current Power, Implementation – Alternating AC Source, as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation were described in detail in inspection report 05000261/2013008. In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000261/2013009. Accordingly, apparent violation (AV) 05000261/2013008-01 is updated consistent with the regulatory positions described in this letter. Therefore, AV 05000261/2013008-01, Failure to Perform Adequate Preventative Maintenance on the DSDG In accordance with Vendor Guidelines is updated as VIO 05000261/2013008-01 in the Mitigating Systems Cornerstone with a safety significance of White and no crosscutting aspect.

The NRC determined the performance of H. B. Robinson Steam Electric Plant, Unit 2 to be in the Regulatory Response Column of the Reactor Oversight Process Action Matrix as of the second quarter of calendar year 2013. Therefore, the NRC plans to conduct a supplemental inspection in accordance with Inspection Procedure 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," to provide assurance that the root causes and contributing causes of risk-significant performance issues are understood, that the extent of cause is identified, and that your corrective action for risk-significant performance issues are sufficient to address the root and contributing causes and prevent recurrence. The NRC requests that your staff provide notification of your readiness for the NRC to conduct a supplemental inspection to review the actions taken to address the White inspection finding.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

Victor M. McCree
Regional Administrator

Docket No.: 50-261
License No.: DPR-23

Enclosures: 1. Notice of Violation
2. NRC Bases for Final Characterization and Significance Determination

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Two White Inputs in a Strategic Performance Area," to provide assurance that the root causes and contributing causes of risk-significant performance issues are understood, that the extent of cause is identified, and that your corrective action for risk-significant performance issues are sufficient to address the root and contributing causes and prevent recurrence. The NRC requests that your staff provide notification of your readiness for the NRC to conduct a supplemental inspection to review the actions taken to address the White inspection finding.

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ADAMS: ☒ Yes ACCESSION NUMBER: ML13262A069 ☒ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

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DATE	9/6/2013	9/12/2013	9/6/2013	09/18/2013	09/18/2013		
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Letter to William R. Gideon from Victor M. McCree dated September 19, 2013

SUBJECT: H.B. ROBINSON STEAM ELECTRIC PLANT - FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING, NOTICE OF VIOLATION, AND ASSESSMENT FOLLOW-UP LETTER - NRC INSPECTION REPORT 05000261/2013009

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NOTICE OF VIOLATION

Carolina Power and Light
H.B. Robinson Steam Electric Plant
Unit 2

Docket No. 50-261
License No.: DPR-23
EA-13-129

During an NRC inspection completed on March 30, 2013, one violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

10 CFR 50.63 (c)(2) states, in part, that the alternate AC power source, as defined in section 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate AC source(s) and required shutdown equipment are started and lined up to operate. Robinson Nuclear Plant Station Blackout Coping Analysis Report 8S19-P-101, identifies the Dedicated Shutdown Diesel Generator (DSDG) as its alternate AC power source and specifies that Robinson is required to cope for eight hours following a station blackout and that alternate AC power must be supplied within one hour to shut down equipment by the DSDG. Additionally, the DSDG is required to provide emergency power during selected Fire Safe Shutdown (SSD) scenarios.

Contrary to the above, from August 28, to October 3, 2012, the licensee's failure to have an alternate AC power source with acceptable capability to withstand station blackout for the required durations specified in its coping analysis. Specifically, during surveillance testing of the DSDG on October 2, 2012, the DSDG automatically shut down on high engine temperature due to a failure of the radiator drive belts. The condition of the drive belts was significantly degraded, due in part to a lack of adequate inspection, maintenance, and/or periodic replacement. Based on the failure of the DSDG and necessary repair time, this degraded condition would have prohibited the DSDG from supplying power to shutdown equipment within one hour following a station blackout and could have rendered the plant unable to cope for eight hours after a postulated station blackout or to provide emergency power for certain selected Fire SSD scenarios.

This violation is associated with a White SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Carolina Power and Light is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-13-129" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for

Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this XX day of (Month) 20(XX)

NRC Bases for Final Characterization and Significance Determination

Summary

The NRC preliminary Significance Determination Process (SDP) risk assessment estimated an increase in core damage frequency (CDF) of 9.94E-6/ year of which 3.16E-7/year represented the CDF increase associated with internal risk and 9.62E-6/year represented the CDF increase associated with fire risk. These estimates did not include any recovery credit for repair of the DSDG.

A regulatory conference was held for this finding and apparent violation on August 19, 2013, and Carolina Power and Light Company (doing business as Duke Energy Progress (DEP)) presented their assessment of the risk increase due to the finding. The total DEP risk increase was an incremental conditional core damage probability of 6.5E-7, which would characterize the finding as GREEN, a finding of very low safety significance. DEP indicated that plant design and configuration of dedicated shutdown cables / high energy arc fault (HEAF) sources make loss of the dedicated shutdown bus unlikely. Specifically, the impact of 480V HEAF is smaller when compared to a 4160V HEAF. In addition, DEP indicated that recovery credit should be higher. Specifically, the operators would have established and maintained steam generator feed and cool down rates before dedicated shutdown battery depletion and maintained the heat sink after battery depletion. Additionally, maintenance recovery of DSDG belt repair is straight forward with simple tools.

The NRC reviewed the material presented at the regulatory conference, the details of the DEP risk assessment contained in EC 92394, and the responses to the formal requests made by the NRC at the conclusion of the regulatory conference. The NRC agrees that DEP has made a case that some recovery credit is justified for the SBO and fire scenarios which would be risk significant with the DSDG performance deficiency. The NRC used the licensee recovery event tree approach and independently determined an internal events recovery factor of 2.53E-1 (74.7% success). The NRC determined a fire scenario recovery factor of 2.62E-1 (73.8% success). The NRC EDG and DSDG repair probabilities were taken from NUREG/CR 6890, " Reevaluation of Station Blackout Risk at Nuclear Power Plants," Volume 2 and the HEPs were developed using the NRC SPAR-H HRA methodology documented in NUREG/CR 6883, "The SPAR-H Human Reliability Analysis Method". The NRC final recovery credit continued to result in a white finding. The final overall significance of this performance deficiency is a White finding of low to moderate safety significance. The NRC final risk increase included an internal events CDF increase of 8.75E-8/year with a fire risk CDF increase of 2.77E-6/year, for a total CDF increase of 2.86E-6/year. Using the DEP risk estimates with the NRC recovery credit yielded a total risk increase of 2.59E-6/year, which did not include any risk for the 480V ac switchgear E1 HEAF fire scenario. In addition, the risk estimates were adjusted to remove the plant capacity reduction factor for this SDP evaluation as the plant was at full power for the entire 36 day exposure period. In conclusion, the final risk increase in CDF is 2.86E-6/year, a White finding of low to moderate safety significance.

At the regulatory conference, DEP disagreed with the NRC preliminary risk determination of White, a finding of low to moderate safety significance. DEP's risk assessment characterized the finding as Green, a finding of very low safety significance. The DEP basis for the Green risk characterization was as follows: DEP indicated that plant design and configuration of dedicated shutdown cables/HEAF sources make loss of the dedicated shutdown bus unlikely. Specifically, the impact of a 480V HEAF is smaller when compared to a 4160V HEAF. In addition, recovery credit should be higher. Specifically, the operators would have established and maintained Steam Generator (SG) feed and cool down rates before dedicated shutdown battery depletion and maintained the heat sink after battery depletion. Additionally, maintenance recovery of the DSDG belt repair is straight forward with simple tools. The following paragraphs provide a summary of the technical differences and the NRC's basis for the final risk characterization of the finding as White.

1. **DEP Input -** At the regulatory conference, DEP stated that plant design and configuration of dedicated shutdown cables, HEAF sources, make loss of dedicated shutdown bus unlikely. Specifically, the impact of 480V HEAF is smaller when compared to a 4160V HEAF.

   NRC's Position – Exclusion of the contribution of this 480V ac switchgear E1 HEAF scenario did not make a significant reduction to change the NRC's overall White finding CDF estimate. In regarding the HEAF fire scenarios, the NRC agrees that the impact of a 480V HEAF would likely be smaller than a 4160V HEAF. However, the frequency of occurrence of a HEAF in 480V switchgear would likely be higher based on industy experience. At the regulatory conference, DEP presented their contention that loss of the DS bus offsite feeders due to a switchgear E1 HEAF would not occur due to three major factors. These factors are the robust construction of Robinson 480V E1 and E2 switchgear, their outgoing load circuit breakers are protected by Amptectors, and that the 480V switchgear bus E2 cables and Dedicated Shutdown (DS) bus cable targets are outside the zone of influence (ZOI) for a switchgear E1 HEAF emanating at its supply circuit breaker. The guidance for application of fire risk due to the energetic phase of HEAFs is contained in NUREG/CR 6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities", Rev. 0. Appendix M of NUREG/CR 6850 describes the initial energy release which occurs instantaneously with the inception of the fault. Therefore, fast acting circuit breaker protective devices will not be able to isolate in time to prevent the damage since the explosive energy release will occur at time zero of the fault. The guidance for HEAF fire scenario modeling includes specific details to consider any targets in the ZOI damaged at time zero. The ZOI is clearly defined in section M4.2 of NUREG/CR 6850 Rev. 0 to include any cable tray within a 5 foot vertical distance of the top of the cabinet and to include any cable tray above the aisle way directly in front of, or behind, the faulted cabinet, provided that some part of that tray is within 0.3 meters (1 foot) horizontally of the cabinet's front or rear face panel. Using industry fire PRA guidance, it clearly shows that tray R85, which contains switchgear E2 cable and dedicated shutdown bus cable targets, is within this specified ZOI for a HEAF emanating from switchgear E1. The DEP presentation stated that the 480V HEAF ZOI is smaller than that which is characterized in NUREG/CR 6850 Appendix M for medium voltage switchgear. Appendix M of NUREG/CR 6850 is not limited to HEAFs in medium voltage switchgear, it covers switchgear, load centers and bus bars/ducts 440V and above. In their presentation, DEP contended that National Fire Protection Association (NFPA) Standard 805, Frequently Asked Question (FAQ) 06-0017 specifically limits 480V HEAFs to feeder circuit breakers.

FAQ 06-0017 exempts motor control centers from HEAF treatment if the MCC does not directly operate equipment such as load centers, however, it does not explicitly limit 480V HEAFs to feeder circuit breakers. NUREG/CR 6850 Revision 0 Appendix M does state, that in nearly all the HEAF events, the HEAF occurred in the feed circuit breaker cubicle where most of the electrical energy resides. The NRC agrees that the impact of a 480V HEAF would likely be less than that of a 4160V HEAF, however, the likelihood of damage outside the cabinet cannot be neglected. The damage from a HEAF is dependent on the fault location and the path of the energy release. FAQ 06-0017 indicates that the state of knowledge of HEAFs continues to evolve. With such a small population of reported failures there is high uncertainty associated with the resulting damage. The best estimate would consider the damage results to be representative of a probability distribution of damage with some likelihood of damage beyond the cabinet considered. The DEP position that there is no probability of damage outside the cabinet is not supported by the industry guidance. For the purposes of characterizing the risk of this finding, the risk contribution of the switchgear E1 HEAF scenario was not significant to the overall estimate of risk as a result of this finding. The NRC final risk significance remains White without the risk of the switchgear E1 HEAF. In the final analysis, the DEP risk with NRC recovery factors would yield a risk total of 2.59E-6/year without the E1 HEAF risk contribution and a risk total of 4.46E-6/year with the E1 HEAF risk included.

2. **DEP Input** - At the regulatory conference, DEP stated that operator recovery credit should be higher. Specifically, the operators would have established and maintained SG feed and cool down rates before dedicated shutdown battery depletion and maintained the heat sink after battery depletion. The significance determination portion of the DEP regulatory conference presentation stated that full recovery credit (defined as 92.5%) should be applied. DEP indicated that procedure DSP-002 establishes initial success for SG feed with the steam driven auxiliary feedwater pump (SDAFW), and they described the DEP evaluation of the Human Reliability Analysis associated with the local manual operator actions for maintaining SG feed. The DEP Human Error Probability (HEP) discussion for this action stated that maintaining feed leads to success and provided the following regarding the HEP Performance Shaping Factors (PSFs): No diagnosis required by the operator, stress was considered high, complexity was considered nominal, training was considered nominal, procedures were available, indications were available, expansive time was available, and the Technical Support Center would provide oversight.

   NRC's Position - The NRC determined that DEP made a case that recovery credit should be considered for the SBO and fire scenarios which were risk significant with the DSDG performance deficiency. The NRC used the DEP recovery event tree approach and independently determined an internal events recovery factor of 2.53 E-1 (74.7 % success). The NRC determined a fire scenario recovery factor of 2.62 E-1 (73.8% success). The result of this recovery credit was not sufficient to change the risk characterization of the finding. The difference in the SBO recovery event tree was the NRC use of an HEP of 3.62 E-1 for event REC2 (Reset TDAFW Pump following over speed trip). The difference in the fire scenario event tree was the NRC use of 3.62 E-1 for REC2 and use of 4.7E-1 for event REC1 (DSDG non-recovery probability).

The details of the DEP risk assessment were contained in Engineering Change (EC) 92394, revision 0. The operator action for establishing and continuing feed to the SGs during a SBO or fire induced SBO was reviewed. The time requirement for establishing feed to the SGs in procedure DSP-002 is 27.5 minutes. Feed must be established within this time requirement and maintained for the 24 hour SBO mission time and for the 72 hour fire safe shutdown mission time. Battery depletion would be in approximately one hour given the performance deficiency. The DEP risk assessment used an event tree approach to determine a recovery factor. The DEP recovery factor for internal events was 1.2E-1 which meant 88% success of internal events recovery. The DEP calculated fire scenario recovery was 7.71E-2 which meant 92.29% success of recovery. The DEP recovery factor event trees were developed from RCP seal LOCA leak outcomes, EDG and DSDG recovery probabilities and HEPs for 3 operator actions: (1) Implementing RCS Cool down (CD), (2) Feed SG Without SG Level Indication (FEED), and (3) Restart AFWTDP After Over speed Trip (REC2). The NRC reviewed the Performance Shaping Factors (PSFs) used in determining the HEPs associated with these three recovery operator actions. The NRC assessment determined that the Training PSF is too optimistic as the training is not done under the conditions which would be present after the loss of the DSDG during a SBO and subsequent battery depletion. Training is conducted for the procedure as written which assumes the DSDG is functional and keeping the DS batteries energized, and hence the training and procedures PSFs should not be judged as nominal. The procedure PSF for the TDAFW pump over speed trip reset is overly optimistic as the procedure cited to be used is an Operational Surveillance Test (OST) procedure and no link exists in the DSP procedures to transition to the OST, and no guidance exists in the DSP procedure to diagnose and reset the TDAFW pump once it trips on an over speed condition. For the Feed SG without SG level indication operator manual action, a component of diagnosis is required for the operator to be successful in continuing in the procedure because the primary SG level indication is lost following battery depletion. The NRC independently evaluated the risk of the three recovery event tree HEPs Cool down (CD), Feed without SG Level Indication (FEED), and Restart AFWTDP after over speed (REC2). The NRC results for the CD and FEED HEPs were not significantly different from the DEP results and the DEP results for these 2 HEPs were used in the NRC recovery event tree analysis. The NRC REC2 HEP result is different. Details of the NRC's analysis of these 3 HEPs are discussed in the section entitled NRC HRA analysis.

3. **DEP Input** - At the regulatory conference, DEP stated that DSDG belt replacement is straight forward with simple tools.

   NRC's Position - The actual diagnosis of the DSDG failure, due to high temperature trip on loss of radiator cooling belt, would not have been obvious under SBO conditions as the radiator drive motor would not have been operating with the fan stationary as was the case during the surveillance test failure. A DSDG trip on high temperature could result from different cooling system malfunctions such as, DSDG coolant pump and drive problems, etc. The DEP DSDG early recovery probability (REC1) used in the recovery event trees was overly optimistic. DEP utilized EDG recovery probabilities from NUREG/CR 6890 Volume 2 for the SBO EDG recovery probability. However, the calculated recovery probabilities for the DSDG were much more optimistic than industry average values for well-maintained safety related EDGs. The NRC utilized a value of 4.7E-1 for the DSDG early non-recovery probability (<4.25 hours) which was taken from NUREG/CR 6890 volume 2. This value was considered a lower limit since the DSDG has not received the same degree of maintenance and testing as the safety related EDGs. Additionally, the DEP analysis incorporated a plant capacity risk reduction factor in the CCDP risk and fire ignition frequency terms which inappropriately reduces the risk of this SDP risk evaluation for a PD which had a 36 day exposure period at full power. The NRC adjusted the final risk totals to remove the reduction from the plant capacity factor.

NRC Human Reliability Analysis

The NRC performed an independent assessment for the three recovery event tree operator actions, (1) Implementing RCS Cool down (CD), (2) Feed SG without SG Level Indication (FEED), and (3) Restart AFWTDP after Over speed Trip (REC2). The NRC HEP for each of these actions is discussed in the following section.

Implementing RCS Cool down

The DEP value for this HEP was 1.6 E-2. The NRC determined an HEP value for this operator action of 2.92 E-2 which is not significantly higher than the DEP value. The NRC considered this to be a case where to fail to implement the RCS cool down required in the EPP-1 and DSP-002/DSP-007 procedures the operator would have to fail to execute a procedure step requiring the cool down. The NRC determined this action to be an action only HEP. The Action PSF assignments were: time at extra time, stress at high, complexity at nominal, experience / training at low, procedures at available but poor, ergonomics at poor, fitness for duty at nominal, and work processes at nominal. The DEP value of 1.6E-2 was used for this HEP in the NRC's recovery event trees.

Feed SG without SG Level Indication

The DEP value for this HEP was 3.6E-2. The NRC determined a HEP value for this operator action of 9.23 E-2. This is only for the underfeed outcome. Overfeed is considered a success in the event tree with the potential of causing TDAFW pump trip due to SG steam line moisture carryover conditions. The NRC considers that overfeed would be a more likely outcome as operators would most likely tend to overfeed in the absence of SG level indication to ensure adequate heat sink and heat removal. The NRC Diagnosis PSFs were: time at expansive, stress at high, complexity at nominal, experience /training at low, procedures at available but poor, ergonomics at poor, fitness for duty at nominal, and work processes at nominal. The Action PSFs were: time at expansive, stress at high, complexity at nominal, experience / training at low, procedures at nominal, ergonomics at poor, fitness for duty at nominal, and work processes at nominal. The NRC HEP was higher than the DEP HEP however because the Operator training reinforces maintaining a heat sink, the NRC used the DEP HEP value in the NRC's recovery event trees.

Restart AFWTDP after over speed Trip

The DEP value for this HEP was 5.9 E-3. The NRC determined a HEP value for this HEP of 3.62E-1 which was used in the NRC recovery event trees. The NRC Diagnosis PSFs were: time at extra time, stress at high, complexity at nominal, experience / training at nominal, procedures at incomplete, ergonomics at poor, fitness for duty at nominal, and work processes at nominal. The NRC Action PSFs were: time at extra time, stress at high, complexity at moderate, experience / training at nominal, procedures at incomplete, ergonomics at poor, fitness for duty at nominal, and work processes at nominal. The DEP cites procedure OST-202 as the procedure which would be used by the operator to accomplish this action. The operator would be in procedures EPP-1, or DSP-002 and DSP-007 for implementing the recovery from the SBO or fire induced SBO scenarios. These procedures do not reference OST-202 for guidance on diagnosing a trip of the TDAFW pump on over speed conditions when operating the TDAWF pump in local manual and there is no local guidance at the pump to assist the operator. The operator would most likely be at the secondary control station and not at the TDAFW pump for these scenarios. The action would consist of determining that the TDAFW pump had tripped, isolating the steam supply, aligning the system to drain the moisture from the steam line to the AFWTDP turbine, resetting the over speed trip device , and restarting the pump and re-establishing the feed conditions.