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

August 12, 2010

EA-10-094

Mr. David A. Baxter
Site Vice President
Duke Energy Carolinas, LLC
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: FINAL SIGNIFICANCE DETERMINATION OF ONE YELLOW FINDING AND ONE WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 05000269/2010008, 05000270/2010008, AND 05000287/2010008 - OCONEE NUCLEAR STATION)

Dear Mr. Baxter:

This letter provides you with the final significance determination of three apparent violations (AVs) identified in NRC Inspection Report (IR) Nos. 05000269/2010007, 05000270/2010007, and 05000287/2010007, dated June 9, 2010.

Two AVs were assessed using the Significance Determination Process and were preliminarily characterized as Greater than Green, findings of greater than very low safety significance that may require additional NRC inspection. These findings involved: (1) the failure to identify and correct Unit 2 and Unit 3 Standby Shutdown Facility (SSF) Reactor Coolant Makeup (RCM) letdown line degradation in a timely manner after degradation was identified on Unit 1, as required by 10 CFR 50, Appendix B, Criterion XVI; and (2) the failure to ensure the SSF RCM subsystem for all three units remained operable as required by Technical Specification 3.10.1.

A third AV involving the requirements of 10 CFR 50.9 was being considered for escalated enforcement under the NRC's traditional enforcement process, and was related to materially inaccurate information provided to the NRC in the "Oconee Nuclear Station SSF RC Letdown Action Plan."

At your request, a Regulatory and Predecisional Enforcement Conference was held on July 13, 2010, to discuss your views on these issues. A meeting summary was issued on July 20, 2010, which included copies of the slide presentation made by Duke Energy Carolinas, LLC (DEC) (ADAMS Accession # ML102020020). During the conference, DEC staff discussed the circumstances surrounding the SSF RCM letdown line degradation issues, DEC's assessment of the significance of the preliminary Greater than Green findings, the AV related to inaccurate information, and the cause analyses and corrective actions taken.

At the conference, DEC agreed with the NRC's characterization of the three issues as violations of regulatory requirements. However, for the two issues assessed under the Significance Determination Process, DEC concluded that the TS violation and the 10 CFR 50, Appendix B, Criterion XVI violation stemmed from a common performance deficiency involving a failure to promptly identify and correct a degraded condition in the SSF RCM letdown line for Units 1, 2, and 3, beginning in approximately 2008. As such, DEC proposed that one performance deficiency was more appropriate to address both issues related to the SSF RCM letdown line degradation.

After further review, the NRC has concluded that these findings are best characterized by two separate and distinct performance deficiencies. As DEC acknowledged at the conference, the cause of the finding involving the TS violation was due, in part, to improper downstream strainer selection, inadequate testing, untimely and ineffective Unit 3 Emergency Sump foreign material corrective actions, inadequate design documentation, a valve manufacturing deficiency, and legacy foreign material. As discussed in NRC Inspection Manual Chapter 0612, Power Reactor Inspection Reports, a performance deficiency is defined as an issue that is the result of a licensee not meeting a requirement or standard where the cause was reasonably within the licensee's ability to foresee and correct and that should have been prevented. The NRC agrees with DEC that the factors discussed above contributed to the finding involving the TS violation. In addition, the NRC concluded that a subset of these causes (i.e., improper strainer selection, inadequate testing, and inadequate design documentation) was reasonably within DEC's ability to foresee and correct, beginning in 2008, and should have been prevented. Although ineffective corrective actions, beginning in 2008, also contributed to the TS violation, the NRC concluded that the other causal factors as discussed above should have enabled DEC to prevent the TS violation. The NRC also notes that the findings involving the TS violation and the 10 CFR 50, Appendix B, Criterion XVI violation were separated by significant periods of time. Specifically, the TS violation began as early as mid-2008 for at least one unit, while the 10 CFR 50, Appendix B, Criterion XVI violation was limited to a specific period of time for Units 2 and 3, beginning in October 2009 through mid-February 2010. For these reasons and in consideration of agency guidance regarding the definition of a performance deficiency, the NRC concluded that the two findings should be characterized as separate performance deficiencies.

DEC suggested that the exposure time related to the two findings gives the appearance of double counting since the risk analysis for Units 2 and 3 is calculated with a four month exposure time for the failure to identify and correct SSF RCM letdown line degradation in a timely manner, and a 12-month exposure time for the failure to ensure SSF RCM subsystem operability. The NRC notes that although the exposure periods for each performance deficiency overlap, the risk analyses were developed using separate and discrete exposure time periods. Consistent with the NRC's Reactor Oversight Program guidance, a one year exposure time was used for the performance deficiency involving the TS violation, while a four month exposure time was used for the performance deficiency involving the 10 CFR 50, Appendix B, Criterion XVI violation. Based on this, the NRC concluded that the exposure times for each performance deficiency were appropriately factored into the risk analyses, and this methodology does not represent double counting as suggested by DEC.

After considering the information developed during the inspection and the information provided by DEC during and after the conference, the NRC has concluded that the finding involving the failure to ensure the SSF RCM subsystem for all three units remained operable as required by

Technical Specifications should be characterized as a Yellow finding of substantial importance to safety. In addition, the finding involving the failure to identify and correct Unit 2 and Unit 3 SSF RCM letdown line degradation in a timely manner after degradation was identified on Unit 1 should be characterized as a White finding of low to moderate significance with regard to safety. These findings will require additional NRC inspections. The bases for the NRC's significance determination for these findings, and the differences in the licensee's characterization of the findings, are discussed in Enclosure 2. The key differences between the licensee's characterization and the NRC's significance determination involved consideration of electrical bus duct fault frequency and propagation.

The NRC also has determined that the finding involving the failure to ensure the SSF RCM subsystem for all three units remained operable is a violation of Oconee Nuclear Station Technical Specification 3.10.1, and the finding involving the failure to identify and correct Unit 2 and Unit 3 SSF RCM letdown line degradation in a timely manner is a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violations are described in detail in IR 05000269/2010007, 05000270/2010007, and 05000287/2010007. In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with one Yellow and one White finding.

Regarding the AV involving inaccurate information that was assessed under the NRC's traditional enforcement process, based on the information developed during the inspection and the information provided at the conference, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation and the circumstances surrounding it are described in detail in IR 05000269/2010007, 05000270/2010007, and 05000287/2010007. The cited violation involved the requirements of 10 CFR 50.9, which state, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects. On December 18, 2009, DEC provided information to the NRC that was not complete and accurate in all material respects. In this case, the violation impacted the regulatory process, in that the "Oconee Nuclear Station SSF RC Letdown Action Plan," was used, in part, by the NRC staff as the basis for determining whether DEC's response to the degraded condition was adequate and whether additional compensatory actions or NRC review would be necessary. Had the information been complete and accurate at the time provided, it likely would have resulted in a reconsideration of a regulatory position or substantial further NRC inquiry. Based on the above, the NRC has concluded that the violation of 10 CFR 50.9 is appropriately characterized at Severity Level III, in accordance with the NRC Enforcement Policy.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $70,000 is considered for each Severity Level III violation (i.e., the 10 CFR 50.9 violation). Because your facility has not been the subject of escalated enforcement action within the past two years, the NRC considered whether credit was warranted for *Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. As DEC presented at the conference, corrective actions regarding the 10 CFR 50.9 violation included, in part: (1) reinforcement of accountability for administrative procedure compliance; (2) reinforcement of expectations regarding communications with NRC; (3) conduct of a table top session and issuance of written expectations to the Regulatory Compliance group; and (4) issuance of a Duke Energy Fleet wide communication by the Senior Vice President for Oconee. Based on the above, credit is warranted for the factor of *Corrective Action* regarding the 10 CFR 50.9 violation. The NRC notes that the 10 CFR 50, Appendix B, Criterion XVI violation and the TS violation were dispositioned in accordance with the NRC's Reactor Oversight Process and are not subject to the civil penalty assessment process of the Enforcement Policy.

Therefore, to encourage prompt and comprehensive correction of violations, I have been authorized, after consultation with the Director, Office of Enforcement, to propose that a civil penalty not be assessed in this case. However, significant violations in the future could result in a civil penalty.

You have 30 calendar days from the date of this letter to appeal the staff's significance determination for the Yellow and White findings or the Notice of Violation. An appeal of the Yellow and White findings will be considered to have merit only if it meets the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

Because plant performance for this issue has been determined to be beyond the licensee response band of the NRC Action Matrix, we will use the Action Matrix to determine the most appropriate NRC response for this event. We will notify you, by separate correspondence, of that determination. In addition, issuance of this Severity Level III violation constitutes escalated enforcement action that may subject you to increased inspection effort.

The NRC has concluded that information regarding the reason for the violations, the corrective actions taken and planned to correct the violations and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in this letter and in the information presented by DEC at the conference. Therefore, you are not required to respond to this letter unless the description herein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

For administrative purposes, this letter is issued as NRC Inspection Report No. 05000269/2010008, 05000270/2010008, and 05000287/2010008. Accordingly, AVs 05000270, 287/2010007-01, 05000270, 287/2010007-02, and 05000269, 270, 287/2010007-03 are updated consistent with the regulatory positions described in this letter. Therefore: (1) AV 05000270, 287/2010007-01, Failure to Promptly Identify and Correct an Adverse Condition Affecting Operability of the Unit 2 and Unit 3 Standby Shutdown Facility, is updated as VIO 05000269, 270, 287/2010007-01 with a safety significance of White and a cross-cutting aspect in the area of Human Performance, H.1(b); (2) AV 05000270, 287/2010007-02, Materially Inaccurate Information Provided to NRC Regarding SSF Event Mitigation Capability, is updated as Severity Level III violation VIO 05000270, 287/2010007-02; and (3) AV 05000269, 270, 287/2010007-03, SSF Reactor Coolant Makeup Subsystem Inoperable for Greater than Allowed by Technical Specifications, is updated as VIO 05000270, 287/2010007-03 with a safety significance of Yellow and no cross-cutting aspect.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response, if you choose to submit one, 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. If personal privacy or proprietary information is necessary to provide an acceptable response, 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 information, 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). The NRC also includes significant enforcement actions on its Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcement/actions.

Should you have any questions concerning this letter, please contact Mr. Jonathan Bartley at 404-997-4607.

Sincerely,

/Victor McCree RA for/

Luis A. Reyes
Regional Administrator

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55

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

cc w/Encl: (see page 6)

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, 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 information, 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). The NRC also includes significant enforcement actions on its Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcement/actions.
Should you have any questions concerning this letter, please contact Mr. Jonathan Bartley at 404-997-4607.

Sincerely,

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Regional Administrator

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DATE	08/10/2010	07/30/2010	08/04/2010	08/04/2010	08/11/2010	08/04/2010	08/04/2010
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Letter to David Baxter from Luis A. Reyes dated August 12, 2010

SUBJECT: FINAL SIGNIFICANCE DETERMINATION OF ONE YELLOW FINDING AND ONE WHITE FINDING AND NOTICE OF VIOATION (NRC INSPECTION REPORT 05000269/2010008, 05000270/2010008, AND 05000287/2010008 - OCONEE NUCLEAR STATION)

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NOTICE OF VIOLATION

Duke Energy Carolinas, LLC
Oconee Nuclear Station
Units 1, 2, and 3

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55
EA-10-094

During an inspection and in-office review completed on May 27, 2010, violations of NRC requirements were identified. In accordance with the NRC Enforcement Policy, the violations are set forth below:

A. Technical Specification 3.10.1 requires the Standby Shutdown Facility (SSF) to be operable in Modes 1, 2 and 3 and Condition C allowed the SSF Reactor Coolant Makeup (RCM) subsystem to be inoperable for up to seven days without additional actions being taken.

Contrary to the above, the SSF RCM system was inoperable for greater than seven days without additional actions being taken. Specifically, the SSF RCM subsystem was inoperable whenever the unit was in Modes 1, 2, or 3 from May 30, 2008, until October 9, 2009, for Unit 1; from December 10, 2008, until February 20, 2010, for Unit 2; and from May 19, 2009, until February 23, 2010, for Unit 3 because the letdown line could not pass the required flow due to the licensee's failure to implement design reviews, inspection activities, or adequate testing to ensure operability.

This violation is associated with a Yellow Significance Determination Process finding.

B. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, deficiencies, and defective material are promptly identified and corrected.

Contrary to the above, from October 19, 2009, to February 20, 2010, (Unit 2) and February 23, 2010, (Unit 3) the licensee failed to promptly identify and correct a condition adverse to quality involving foreign material on the Unit 2 and 3 SSF letdown line filters. In this case, after identification of a condition adverse to quality on Unit 1, the licensee failed to identify and correct a similar condition adverse to quality on Unit 2 and Unit 3. The condition would have adversely affected operator ability to control reactor coolant system inventory during a postulated event involving the use of the SSF.

This violation is associated with a White Significance Determination Process finding.

C. 10 CFR 50.9(a) requires, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

Contrary to the above, on December 18, 2009, the licensee provided information to the NRC that was not complete and accurate in all material respects. The information provided described compensatory actions for controlling pressurizer level during an SSF event which was not available due to a closed manual valve inside containment. This information, combined with an evaluation that showed flow rates on Unit 2 and Unit 3 were greater than the required value for level control in the last as-tested condition, was material to the NRC because it was used, in part, as the basis for determining whether the licensee's response to the degraded condition was adequate and whether additional compensatory actions or NRC review would be necessary.

This is a Severity Level III violation (Supplement VII).

The NRC has concluded that information regarding the reasons for the violations, the corrective actions taken or planned to correct the violations and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket, and in the information presented by DEC at the July 13, 2010, Regulatory and Predecisional Enforcement Conference. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response "Reply to a Notice of Violation EA-10-094," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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.

Should you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS). 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. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. 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 that such material be withheld from public disclosure, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim (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.

Dated this 12th day of August 2010

EA-10-094

NRC Bases for Final Significance Determination

On July 13, 2010, the NRC held a regulatory conference with representatives of Duke Energy Carolinas, LLC (DEC) Oconee Nuclear Plant, to discuss two preliminary Greater than Green inspection findings documented in NRC Inspection Report (IR) Nos. 05000269/2010007, 05000270/2010007, and 05000287/2010007, dated June 9, 2010. These findings involved: (1) the failure to identify and correct Unit 2 and Unit 3 Standby Shutdown Facility (SSF) Reactor Coolant Makeup (RCM) letdown line degradation in a timely manner after degradation was identified on Unit 1; and (2) the failure to ensure the SSF RCM subsystem for all three units remained operable as required by Technical Specifications.

At the regulatory conference, DEC highlighted its assumptions used in determining the risk associated with the two findings, and the differences between its risk calculations and those performed by the NRC as part of the Significance Determination Process (SDP). The paragraphs below provide a summary of the technical differences and the NRC's bases for incorporating DEC's assumptions into the NRC's final SDP.

1. DEC Input – At the conference DEC noted that the NRC used the fire ignition frequencies developed in NUREG/CR 6850 (EPRI TR 1011989), EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, in its preliminary SDP analysis. However, DEC developed refinements that resulted in fire ignition frequencies that were lower than those used in the NUREG. In addition, DEC cited other industry efforts that are underway to refine the frequencies of NUREG/CR 6850, including Electric Power Research Institute's (EPRI) TR-1016735, "Fire PRA Methods Enhancements," the Pressurized Water Reactor Owners Group (PWROG) and the Nuclear Energy Institute (NEI). DEC presented information at the conference showing that the NUREG/CR 6850 results over estimate low voltage electrical cabinets by a factor of ten when compared to actual operating experience. DEC also noted that the Advisory Committee on Reactor Safety (ACRS) has recommended the need for refinement of NUREG/CR 6850 fire ignition frequencies.

   NRC Consideration – The NRC concluded that the best estimate risk assessment would utilize fire ignition frequencies from NUREG/CR-6850 (EPRI TR 1011989), as modified by National Fire Protection Association (NFPA) Standard 805, Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, Frequently Asked Question (FAQ) 08-0048. As DEC is aware, NUREG/CR 6850 was a collaborative effort of the NRC and the nuclear industry that provided consensus methods, tools, and data for conducting fire Probabilistic Risk Assessments. Additional refinements of this basic document have been made through resolution of FAQs via the pilot program to revise the fire protection licensing basis of a facility under NFPA 805. The NRC generally accepted the fire ignition frequencies in EPRI's TR-1016735 and incorporated this into FAQ 08-0048. In the final SDP, the NRC used the frequencies accepted under this FAQ resulting in reduction in the fire ignition frequencies by approximately 50% with a comparable core damage frequency reduction for those categories where it was used.

   The NRC acknowledges industry efforts to refine fire ignition frequencies; however, the NRC concluded that the frequencies used in the final SDP represent the best available information. As for the possibility that actual operating experience indicates an over estimation by a factor of ten for low voltage electrical cabinet fires (when compared to the NUREG/CR 6850 database), the NRC notes that the NUREG/CR 6850 database does not segregate low voltage electrical cabinet fires from other electrical cabinet fires. Instead, the NUREG database includes electrical cabinet fires, which include switchgear, motor control centers, direct current distribution panels, relay cabinets, control cabinets and switch panels. Within this bin, the database indicates 109 actual fires that had the potential to become self sustaining over approximately 2500 reactor years. This information was utilized in the NRC's final risk assessment.

2. DEC Input – In DEC's analysis, electrical cabinets were modeled in the Probabilistic Risk Assessment (PRA) using an approach different from that discussed in NUREG/CR 6850. This approach involved different treatments for low and high voltage cabinets and a different treatment based upon whether the cabinet was sealed or ventilated, and includes an empirically based severity factor for low voltage cabinets.

   NRC Consideration – The NRC concluded that the best estimate risk assessment would utilize electrical cabinet modeling from NUREG/CR-6850, as modified by NFPA FAQ 08-0042. In the final SDP, the NRC assumed that fires in well-sealed cabinets, as defined by NUREG/CR 6850 and amplified in FAQ 08-0042, did not propagate. This assumption was consistent with DEC's input.

   As for poorly ventilated cabinets, the reduction of oxygen would affect the fire behavior. However, extensive analysis or test results would be needed to understand how such information should be factored into the risk assessment. Even if such information had been available and factored into the risk assessment, the NRC concluded that the effect on the final result would be minimal.

   DEC's input also included an empirically based severity factor for low voltage cabinets. The NRC reviewed this information and does not agree with DEC's methodology for calculating this factor. Specifically, DEC's empirical method assigned the whole bin of 109 fires in all types of cabinets to the denominator without partitioning for the appropriate fraction of fires in low voltage control panels. In addition, the database for this category of fires appeared to contain potentially severe low voltage fires that could traverse outside the cabinet. DEC's analysis did not account for these fires as an input into the numerator which resulted in an artificially low severity factor.

   Although the NRC does not preclude the use of an alternate analysis, in this case, the NRC concluded that the use of DEC's severity factor was not warranted. Nevertheless, the NRC performed a sensitivity analysis as part of the final SDP by varying the severity factor, and concluded that the safety significance determination was insensitive to DEC's severity factor treatment of electrical cabinet fire ignition sources.

3. DEC Input – For cabinets with internal wire ways that allow cables to pass between cabinets, DEC's best estimate approach assumes that fires do not propagate between cabinets or to adjacent equipment since there is a limited oxygen source and no polyvinyl chloride material inside the cabinets.

NRC Consideration – The NRC disagrees with DEC's assertion that fires would not propagate between cabinets or to adjacent equipment. The cabinets in question were generally not well sealed, and targets that were physically located above such cabinets were also within the zone of influence, even for fires exhibiting low heat release rates. In addition, unshielded cabling was present in the wire ways, which would allow fire propagation internal to the cabinets. Also, each cabinet was somewhat inter-connected at the functional level via the wire ways.

Consequently, as an assumption of the final SDP, the NRC concluded that fire propagation between cabinets or to adjacent equipment was credible for each discrete cabinet.

4. DEC Input – DEC noted that for cabinets with internal wire ways, if propagation is assumed ("severe" fires), the NFPA FAQ on "super cabinets" provides an alternate and more appropriate way to count the cabinets and apply the proper fire ignition frequency for Oconee. In addition, DEC requested that the NRC reconsider giving appropriate credit for fire brigade response.

   NRC Consideration – The NRC determined that individual cabinet counting was appropriate, and manual suppression could not be credited for "super cabinets." As discussed in NFPA FAQ 06-0016 on cabinet counting, the disposition of example 3 on page 4 was used in establishing the fire ignition frequency for cabinets with internal wire ways. As an assumption of the final SDP, each cabinet was counted separately and not viewed as a "super cabinet." The NRC fully considered crediting manual suppression of a fire due to fire brigade response. However, because damage to the cabinets due to a fire would likely occur over a short duration, the effectiveness of any manual suppression would be limited. As such, the NRC concluded that credit was not warranted for fire brigade response. The NRC also notes that the compartments containing these types of cabinets do not have an automatic fire suppression system.

5. DEC Input – The NUREG/CR 6850 methodology has been a topic of industry discussion. Consequently, EPRI calculated new frequencies (older data receives less weight) which included a bus duct fault frequency. Inclusive in the data was that one plant experienced three of the seven events. As such DEC concluded that the EPRI frequency should be used with a plant-specific Bayesian update.

   NRC Consideration – The NRC acknowledges that the NUREG/CR 6850 methodology has been a topic of industry discussion. However, we note that in response to NFPA FAQ 07-0035, the NRC did not accept the EPRI calculation and provided a new frequency that was based upon data up to the year 2000. In the NRC's final SDP, the bus duct fault frequency through 2009 was updated using a Bayesian analysis and plant-specific data. As an assumption to the final SDP, the mean fault frequency used was 2.4E-3 bus duct faults per year per unit. When applied to the 4509 linear feet of segmented bus ducts at Oconee, a result of 1.6E-6 fault/year/linear foot was calculated. This value was used in the NRC's final risk assessment.

6. DEC Input – At the conference, DEC noted that the Oconee bus ducts are physically robust. The thickness of the enclosure offers better containment relative to sheet metal and the circular construction provides better protection. Also, the armored cables were less vulnerable targets (galvanized steel).

   NRC Consideration – Given the energy expended via a bus duct fault, the NRC concluded that there was no basis to exclude a fault from breaching the enclosure. Neither testing nor historical evidence was provided to support DEC's contention that the enclosures at Oconee would not breach in the event of a bus duct fault. In fact, the NRC notes that historical evidence indicates that the metallic enclosure itself has become a part of the faulting behavior and was one source of the molten slag ejected during such events.

   Incorporating guidance from FAQ 07-0035, the NRC gave consideration to armored cabling associated with the targets, by assuming that only the first tray within the zone of influence would be damaged.

7. DEC Input – In loss of normal letdown scenarios, the Oconee emergency procedures provide multiple options to ensure a continuous suction path for the charging pumps.

   NRC Consideration – The NRC recognized that the accident sequence modeling in DEC's PRA included multiple options for charging pump suction. Consequently, DEC's modeling assumptions were incorporated into the final SDP.

8. DEC Input – At the conference, DEC suggested that the NRC's preliminary SDP used a 16 month cumulative exposure time for Unit 2 and 3 for the two performance deficiencies. However, risk results are calculated on a per year basis. Therefore, DEC proposes a 12 month exposure time for all three units.

   NRC Consideration – As the NRC clarified at the conference, the NRC's preliminary SDP identified two separate performance deficiencies with discrete exposure times. An exposure time of 12 months was used for the performance deficiency involving the failure to ensure that the SSF RCM subsystem for all three units remained operable as required by Technical Specifications. A four month exposure time was used for the performance deficiency involving 10 CFR 50, Appendix XVI (Units 2 and 3). This methodology is consistent with the NRC's Reactor Oversight Program guidance.

9. DEC Input – In conclusion, DEC determined that the total delta core damage frequency (CDF) for the SSF filter plugging issue was approximately 8E-6, based upon a 12 month duration time. NUREG/CR 6850 does not provide "best estimate" results, and a 12 month exposure time instead of a 16 month cumulative exposure time should be used.

   NRC Consideration – As stated above, the NRC used a 12 month exposure time in calculating the delta CDF for the SSF filter issue for the Technical Specifications violation performance deficiency. After incorporating the fire risk contribution into the delta CDF calculation, and using an exposure time of 12 months, the numerical result varied between 1.6 and 1.8E-5, depending upon which unit was being calculated. The dominant accident sequences all involved bus duct faults that ultimately led to a station blackout. In the fire analysis portion of the final SDP, the NRC used elements of NUREG/CR 6850, augmented with issued FAQs.

Conclusion:

Based on best-estimate assumptions and in consideration of the above, the NRC concluded that the delta CDF increase for the finding involving the failure to ensure that the SSF RCM subsystem for all three units remained operable as required by Technical Specifications, is approximately 1.6E-5 (Yellow), using a 12 month exposure time. In addition, the NRC determined that the delta CDF increase for the finding involving the failure to identify and correct Unit 2 and Unit 3 SSF RCM letdown line degradation in a timely manner after degradation was identified on Unit 1, is approximately 5.0E-6 (White), using a four month exposure time.