UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-8931

February 6, 2006

EA-05-195

Florida Power and Light Company
ATTN: Mr. J. A. Stall, Senior Vice President
Nuclear and Chief Nuclear Officer
P. O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: EXERCISE OF ENFORCEMENT DISCRETION – TURKEY POINT NUCLEAR PLANT (NRC Inspection Report Nos. 05000250,251/2005013)

Dear Mr. Stall:

The purpose of this letter is to provide you with the results of our review of findings involving certain fire response procedures at the Turkey Point Nuclear Plant that would not be effective in ensuring a safe shutdown of Units 3 and 4 should a severe fire develop in certain fire zones. This issue was discussed at a regulatory conference held in the NRC's Region II Office on November 16, 2005.

To summarize our review of this matter, the NRC has concluded that the significance of the findings was very low for Unit 3 and low to moderate for Unit 4. In addition, the NRC has concluded that the exercise of enforcement discretion is appropriate for these findings in accordance with the NRC Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48). The NRC will also refrain from including the findings in the Agency Action Matrix in accordance with NRC Inspection Manual Chapter 0305, Operating Reactor Assessment Program. The bases for our conclusions regarding the significance of the findings is described below and in Enclosure 1.

As background, NRC Inspection Report No. 05000250,251/2005010, was issued on October 7, 2005, and it documented a finding (one apparent violation of 10 CFR Part 50, Appendix R) based on the NRC's review of three separate unresolved items (URIs 05000250,251/2004007-001, -006, and -007). The collective significance of the finding was assessed under the significance determination process as a preliminary "greater than Green" issue (i.e., an issue of at least low to moderate safety significance which may require additional NRC inspection). The cover letter to NRC Inspection Report No. 05000250,251/2005010 informed Florida Power and Light Company (FPL) of the NRC's preliminary conclusion, provided FPL an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for the finding.

On November 15, 2005, FPL submitted a letter to the NRC documenting its intent to adopt the risk-informed, performance-based fire protection program under 10 CFR 50.48(c), which includes approaches in National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating

Plants," 2001 Edition. In accordance with the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48), FPL requested enforcement discretion for existing identified noncompliances, noncompliances identified during its NFPA 805 transition process, and the noncompliance that was the subject of the November 16, 2005, regulatory conference.

At FPL's request, an open regulatory conference was conducted on November 16, 2005, to discuss FPL's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference and material presented by FPL and NRC.

During the conference, FPL presented the results of its estimate of the increase in CDF including influential assumptions and risk analysis methodology. FPL's presentation addressed the three aspects of the finding separately (i.e., URI 05000250,251/2004007-001, -006, and -007), and provided the following conclusions on the matter. In summary, FPL agreed with the NRC's view that URI 05000250,251/2004007-001 was a violation of 10 CFR Part 50, Appendix R, but concluded that the significance of this issue was very low. FPL disagreed with that portion of the finding consisting of URIs 05000250,251/2004007-006 and -007, concluding that procedural requirements were acceptable, a performance deficiency did not exist, and thus this finding did not represent a violation. Furthermore, FPL concluded that the significance of URIs 05000250,251/2004007-006 and -007, even if considered to be a finding, was very low. At the conference, the NRC requested that FPL provide additional information to support its risk analysis. This information was subsequently submitted by letter dated November 23, 2005.

FPL's presentation and subsequent correspondence provided 15 inputs on the safety significance of the finding. The following three of the inputs were incorporated into the NRC's final significance determination: (1) reducing the main control room fire ignition frequency, (2) modifying the time at which the operating charging pump would be stopped during a main control room fire, and (3) reducing the time that reactor coolant pump (RCP) leakage could have created conditions for a small break loss of coolant accident (LOCA). A detailed discussion of the NRC's basis for acceptance or rejection of all inputs is provided in Enclosure 1.

After considering the information developed during the inspection, the information presented at the conference, and the supplemental information provided by FPL after the conference, the NRC has concluded that the final inspection findings are appropriately characterized as very low for Unit 3 and low to moderate for Unit 4.

The NRC also concluded that a violation of 10 CFR 50.48(b)(1) and 10 CFR Part 50, Appendix R, Sections III.G.2 and III.G.3, occurred. In particular, Appendix R, Section III.G.2 states, in part, that where cables or equipment (including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground) of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of three means of ensuring that one of the redundant trains is free of fire damage shall be provided. Section III.G.3 states that alternative shutdown capability should be provided where the protection of systems (whose function is required for hot shutdown) does not satisfy the requirements of Section III.G.2.

In this case, the following three instances were identified where FPL failed to satisfy the above Appendix R requirements: (1) FPL failed to protect control circuits and cables that could cause maloperation of motor-operated valve (MOV) MOV-4-626, "RCP Thermal Barrier Component Cooling Water System Return Isolation Valve," in fire zone (FZ) 67. This condition existed since at least September 9, 2003, when it was first identified by FPL. (2) FPL failed to protect control circuits and cables that could cause maloperation of necessary RCP thermal barrier component cooling system valves MOV-3-716B and MOV-3-626 in FZ 63 and MOV-4-716B valve in FZ 67. This condition existed since at least February 9, 2001, when the applicable procedure page was last revised FPL. (3) FPL failed to protect control circuits and cables that could cause maloperation of necessary RCP thermal barrier component cooling system valves in FZ 106 and did not meet the alternative shutdown capability requirements. Specifically, under certain scenarios, FPL's procedure may not have mitigated a spurious closure of valves MOV-3-716A and MOV-4-716A in a timely manner, possibly resulting in an RCP seal LOCA and pressurizer level dropping below the indicating range. This condition existed since at least April 24, 2002, when the applicable procedure pages were last revised.

The NRC also reviewed FPL's request that the violation be considered for enforcement discretion. On June 16, 2004, the NRC published a final rule revising its regulations in 10 CFR 50.48, which governs fire protection at operating nuclear power plants. This revision became effective on July 16, 2004, and it added a new paragraph (c) to 10 CFR 50.48 that allows reactor licensees to voluntary comply with the risk-informed, performance-based fire protection approaches in NFPA 805 (with limited exceptions stated in the rule language) as an alternative to complying with 10 CFR 50.48(b) or the requirements in their fire protection license conditions. As part of the transition to 10 CFR 50.48(c), licensees will establish the fundamental fire protection program identified in NFPA 805 and will perform a plant-wide assessment to identify fire areas and fire hazards and to evaluate compliance with their existing fire protection licensing basis. This fire protection assessment is beyond the normal licensee review of their fire protection program. With regard to applicable noncompliances, licensees are required to adopt compensatory measures until compliance is either restored to 10 CFR 50.48(b) or achieved per 10 CFR 50.48(c).

In order to provide incentives for licensees initiating efforts to identify and correct subtle violations that are not likely to be identified by routine efforts, the NRC issued Enforcement Policy Statement revisions on June 16, 2004, and January 14, 2005, allowing enforcement discretion for certain fire protection noncompliances identified as part of the transition to 10 CFR 50.48(c) as well as existing identified compliances which could reasonably be corrected under 10 CFR 50.48(c). This interim enforcement discretion policy is consistent with the long-standing policy included in Section VII.B.3, "Violations Involving Old Design Issues," of the Enforcement Policy, which addresses discretion when licensee undertake a comprehensive review and assessment. This exercise of discretion provides appropriate incentives for licensees to initiate efforts to identify and correct subtle violations that are not likely to be identified by routine efforts. As such, the NRC Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48) provides enforcement discretion for licensees who wish to take advantage of the new rule to resolve existing noncompliances (i.e., implement corrective actions until the licensee has transitioned to 10 CFR 50.48(c)) provided that certain criteria are met.

In this case, the NRC concluded that (1) the noncompliance was not associated with a finding that the reactor oversight process significance determination process would evaluate as Red, or it would not be categorized at Severity Level I; (2) FPL submitted a letter of intent before December 31, 2005, stating its intent to transition to 10 CFR 50.48(c), which includes approaches in NFPA 805; and (3) the licensee entered the noncompliance into its corrective action program and implemented appropriate compensatory measures (including an operability evaluation to demonstrate that safety will be maintained during operation and shutdown). Regarding compensatory measures, FPL incorporated acceptable manual operator actions into applicable procedures to address the finding at the time of the original inspection. Based on the above, in accordance with the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48), and in consultation with the Director, Office of Enforcement; I have been authorized to exercise enforcement discretion such that the above violation will not be cited. In addition, based on the above corrective actions and in accordance with NRC Inspection Manual Chapter 0305, Operating Reactor Assessment Program, the NRC will also refrain from including the finding in the Agency Action Matrix.

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

Should you have any questions regarding this letter, please contact Mr. D. Charles Payne, Chief, Engineering Branch 2, Division of Reactor Safety, at (404)562-4669.

Sincerely,

Loren R. Plisco

William D. Travers
Regional Administrator

Docket Nos. 50-250, 50-251
License Nos. DPR-31, DPR-41

Enclosures:
1. NRC Evaluation of Licensee's
   Risk Significance Inputs
2. List of Attendees
3. Material Presented by Licensee
4. Material Presented by NRC

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E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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NRC EVALUATION OF LICENSEE'S RISK SIGNIFICANCE INPUTS

At the November 16, 2005, regulatory conference, Florida Power and Light Company (FPL) provided numerous inputs to the NRC's preliminary significance determination. FPL's letter of November 23, 2005, provided additional details and documented other inputs that were not discussed at the regulatory conference. In all, FPL provided 15 specific inputs to the NRC's preliminary significance determination. The 15 inputs and the NRC's disposition/alteration of the preliminary significance determination process (SDP) are provided below:

1. FPL, through an MPR Associates Thermal-Hydraulic analysis, indicated that after 20 minutes and provided nominal seal leakage remained below 3 gallons per minute (gpm), the fluid in the reactor coolant pump (RCP) cavity would not increase to the critical temperature that produces the possibility of seal failures. Using this analysis, only 1% and 6% of the time on the two units would leakages exceed 3 gpm.

   NRC Disposition - This information was insufficient to justify deviating from the vendor's recommendations to use the "plug" model. Consequently, the critical leakage rate was assumed to be 2.4 gpm for 20 minutes to perform the restoration of cooling to the seals.

2. FPL indicated that only the B charging pump would be tripped (33% of the time) during initial control room evacuation and that the other pumps would be de-energized 5 minutes later in actions outside the control room. Therefore, 66% of the time the A or C charging pump would be in-service providing seal cooling for an additional 5 minutes. This reduced the exposure period that RCP seal leakage could have been high enough to allow elevated temperatures in the critical volume and establish possible seal failure conditions.

   NRC Disposition - After reviewing procedure ONOP-105 and the operating experience of charging pump operation, a modification of the analysis was warranted in this area. Specifically, for Unit 3, 57% of the time charging pump B will be assumed in service and 55% of the time on Unit 4. When charging pump B is in service, the critical leakage rate is assumed to be 2.4 gpm. When the other two charging pumps are in service 50% of the time, the assumed critical leakage will be 2.4 gpm and 3.2 gpm for the other 50% of the time. The basis for the partitioning was due to the uncertainty associated with the possibility that fire effects would initially cause the loss of the operating charging pump. All walk downs and analyses did not focus on the location of charging pump power/controls/interlocks. Therefore, 0.5 was inserted to account for this. These changes were included in the final significance determination.

3. FPL indicated that, due to the unique design of the Turkey Point RCPs, the critical volume of the seal cavity was 48 gallons. This information should be used when determining the critical leakage rate assuming 20 minutes was necessary to restore seal cooling. In addition, FPL provided the leakage data for all RCPs.

   NRC Disposition - The NRC accepted the critical volume as 48 gallons. Upon reviewing the seal leakage information provided by FPL, for a given year, it was assumed 67% of the time for Unit 4 and 14% of the time for Unit 3 that leakage rates were sufficient for the conditions of an RCP seal loss of coolant accident (LOCA) to potentially exist. These exposure periods were factored into the final significance determination.

4. FPL, through a Kleinsorg Group, L.L.C., report, provided an analysis that control room evacuation due to fire would occur after 15 minutes, rather than the 10 minutes used in the preliminary significance determination. This evacuation time was based upon results indicating that the visibility and heat in the control room was not severe enough to cause evacuation until at least 15 minutes.

   NRC Disposition - This analysis did not address the status of the reactor controls, how this situation related to procedural direction for control room evacuation, or the other physical factors contained in NUREG/CR-6850, Chapter 11, Section 11.5.2.11, Step 11.b. All of these aspects were taken into account in the original NRC analysis when selecting the 10-minute evacuation time. Also, any fire modeling should include different ventilation conditions which were not evident from a review of the analysis. Therefore, no alteration of this portion of the preliminary significance determination was appropriate.

5. FPL provided an analysis of post-LOCA operator actions based upon information contained in WCAP-16141.

   NRC Disposition - The staff has reviewed the licensee's proposed time-estimates for time to core uncovery given various RCP seal LOCA sizes. Specifically, assuming a 182-gpm leak rate for each RCP, the licensee suggested times greater than 7 hours to core uncovery. Other times were presented assuming different combinations of charging and AFW availability including a 17-hour case. In general, time frames greater than 5 hours are atypical for a Westinghouse three-loop plant when compared to the NRC's previous fire significance determination phase-3 analysis experience. Normally, the staff has used approximately 5 hours to core uncovery with an 182-gpm leak rate per RCP. The boundary conditions for the 5-hour window assumes the RCPs are tripped, the turbine-driven auxiliary feedwater pump is successful, and there is no RCS inventory makeup (except cold leg accumulators). The 5-hour best-estimate (for PRA modeling) is based on assumptions used to close out Generic Safety Issue 23, "Reactor Coolant Pump Seal Failures," and confirmed under plant-specific analyses during subsequent significance determination cases. The 5-hour time frame is consistent with WCAP-16396, "Westinghouse Owners Group Reactor Coolant Pump Seal Performance for Appendix R Assessments (2005)." WCAP-16396, Table 7-1, cites an estimated time of 5.2 hours to core uncovery when only the turbine-driven auxiliary pump is used for cooldown. WCAP-16396 indicates that a 17-hour time frame may be achievable with an RCS cooldown and depressurization.

   The Turkey Point fire procedure that would have been used for alternate shutdown panel operations appears to assume that a natural circulation cooldown would occur with an intact reactor coolant system. This is not the appropriate context for modeling fire PRA sequences involving a consequential RCP seal LOCA. Specifically, procedure ONOP-105 instructs the operators to halt the RCS cooldown at approximately 1200 pounds per square inch. The procedural hold point specifies an approximate 9-hour time frame to attempt to maintain RCS pressure in order to reduce the likelihood of void formation in the reactor vessel head area. The procedural hold directly conflicts with use of the 17-hour estimate case provided in WCAP-16396 (which is based on WCAP-16141). Therefore, the staff believes the 5-hour time to core uncovery estimate is a more appropriate and realistic time frame for fire PRA sequences of interest and associated human actions. The NRC considers the 5-hour estimate provided in WCAP-16396, which was issued in 2005, as the best available information for significance determination risk assessment purposes and is appropriate for assessing the Turkey Point fire protection issue for RCP seal LOCA rates of 182 gpm per RCP.

6. FPL, through a Scientech report, provided an analysis that the additional actions/equipment needed for operators being able to mitigate a small break LOCA (SBLOCA) from outside the control room would have a failure probability of approximately 7E-2. The NRC used 1.0 in the preliminary significance determination. The actions considered in the FPL analysis included recognition that a SBLOCA was happening, placing a high head safety injection (HHSI) pump into service, opening a crosstie valve, and opening the containment sump recirculation valve to provide suction to an residual heat removal (RHR) pump. All these actions were local.

   NRC Disposition - There were a number of aspects of this mitigation evaluation which the NRC viewed differently. The success criteria used in the FPL analysis was HHSI followed by low pressure recirculation (LPR). The standard success criteria for a SBLOCA is HHSI followed by high pressure recirculation (HPR). There was not sufficient justification for deviating from the standard success criteria. This was especially true given the small break size involved. None of the diagnostic considerations or actions associated with HPR were discussed or considered appropriate. Even for the LPR actions included in the analysis, there was no diagnostic aspect as to when to transition to LPR. This action is predicated on process indications and must be discerned from the appropriate section of the emergency operating procedures (EOP). The key process indication does not exist at the auxiliary shutdown panel and may be damaged by the fire or in subsequent suppression efforts. The HHSI pump portion of the analysis only involved starting an opposite unit HHSI and opening one valve. The actual procedure directed the manipulation of five valves and would require dependency considerations to establish the collective failure probability of accomplishing the critical task. There was no method that related to how operators would ensure adequate HHSI flow was discussed. (No HHSI flow indication is available at the auxiliary shutdown panel.) The FPL analysis relied upon the technical support center when establishing the procedure portion of the performance shaping factor. However, the ramifications of the fire/suppression efforts in the control room were not discussed or considered for equipment/indications in the technical support center. Also, for the equipment being credited, there was not sufficient discussion as to whether the equipment could have been damaged (line item 4 of NUREG/CR-6850, Section 12.5.5.3). In addition, there would be an extra distraction in transferring from the prescribed procedure (ONOP-105) to the EOP, especially since the ONOP clearly stated that the EOP was not applicable. This is important since there would need to be a transition in command and control to the technical support center very early in accident mitigation. The actual evaluation provided FPL's perspective on the increase in the failure probability without discussing the baseline mitigation strategy for shutdown from the remote shutdown panel. Consequently, the actions considered part of the "baseline" risk could not be determined. These were some of the factors coupled with no experience/training of operating personnel to mitigate a SBLOCA from outside the control room and the sub-optimal environmental conditions to diagnose and carry out the mitigation strategy that precluded successful mitigation credit. Therefore, there was no alteration of the preliminary significance determination in this area.

7. In Attachment 7 of its November 23, 2005, letter, FPL presented information to lower the non-suppression probability of low voltage cabinet fires to 0.08. This was derived by gleaning of the fire events provided in NUREG/CR-6850 and extrapolating a non-suppression probability based upon those fires deemed to be "challenging" and those that spread to other cabinets.

   NRC Disposition - NUREG/CR-6850 included an analysis of each of the nominally 1500 fire events in the fire event database to assess each event's individual relevance to fire PRA analysis. Those fire events that had no potential for growth or spread were already removed from the fire frequency calculations. The fundamental criterion for retaining an event was that the reported fire was such that some active intervention was needed to prevent a growing and potentially damaging fire. This approach was explicitly designed to preserve the independence of the credit given to fire suppression efforts. Any alterations to the event screening would also have strong implications for the treatment of fire intensity, fire growth, and fire suppression. This event screening analysis was conducted on a consensus basis. The consensus team included representatives of both Electric Power Research Institute (EPRI) and NRC, and the results were peer reviewed by industry representatives (including FPL). The explicit objective was to resolve past inconsistencies introduced through the application of individualized fire event data reviews and analyses such as that put forward by the licensee for the non-suppression probability of low voltage cabinet fires. Furthermore, the independent analysis of a sampling of the fire events being put forward is fundamentally inconsistent with NUREG/CR-6850 and NRC Manual Chapter 0609, Appendix F, fire frequencies. Based on the information in these documents, the severity factor is tied explicitly to the characteristics of the fire (e.g., the fire heat release rate). Other severity factors should not be applied. Therefore, there was no alteration of the preliminary significance determination in this area.

8. In Attachment 7 of its November 23, 2005, letter, FPL presented information to lower the probability of non-suppression for control room low voltage electrical cabinet fires. Part of that information stated:

   > "The generic duration curve used in NUREG/CR-6850 is conservative for control room fires since plant specific attributes are not considered. Specifically, the derivation of these curves does not include plant specific items such as:
   >
   > a. Continuously occupied locations
   > b. Time to [a] control fire is much smaller than the time to suppress the fire
   >
   > Using these factors[,] it is judged that the probability of non-suppression will be a factor of [two] lower than that determined in NUREG/CR-6850."

NRC Disposition - NRC Manual Chapter 0609, Appendix F, main control room non-suppression curves only included data from control room fire events and assume all control rooms are continuously manned at all plants. Hence, the data inherently reflected the fire fighting response for continuously manned areas. In addition, the NRC observed the control room at Turkey Point and classified it as typical of a nuclear power plant control room. The licensee's input that fire control times were "much smaller" than fire suppression times was without support in the explicit context of fires, and in particular, main control room fires. The Requantification Study Team, which produced NUREG/CR-6850, considered the possibility of drawing such distinctions in its analysis of event data. However, the team found that the distinction could not be supported by the available fire event data either for fires in general or any specific subset of fire events. There was no technical basis other than the analyst's judgment provided for the input that the NRC Manual Chapter 0609, Appendix F, fire suppression curve was conservative by a factor of two or more. Therefore, this input was rejected, and the non-suppression curves used in the preliminary SDP analysis remained applicable.

9. In Attachment 7 of its November 23, 2005, letter, FPL indicated that a value of 2.5E-3/yr instead of 4.8E-3/yr should be used for main control board fire ignition frequency in the SDP analysis. This was based upon the information located in NUREG/CR-6850, Appendix C, Table C-3.

   NRC Disposition - The NRC agreed that the lower ignition frequency was appropriate for the main control boards in the main control room. This reduction was incorporated into the final significance determination.

10. In Attachment 7 of its November 23, 2005, letter, FPL stated that an additional 0.50 factor should be used in the SDP to bound the uncertainty and conservative assumptions associated with the horizontal spread of the postulated control room fires.

    NRC Disposition - The fact that fire spread was limited in past fires was reflected elsewhere in the analysis. In particular, this behavior was reflected in two companion assumptions. Of greatest importance in the context of the main control room analysis was the substantial credit to the disruption of main control room fires within a very short time period present in the fire suppression response curve. This was consistent with the experience base which showed that prompt suppression will limit fire spread. The second assumption was tied to fire intensity. Manual Chapter 0609, Appendix F, assumed that 90% of all fires remain small even if left unsuppressed for an extended period. (The NUREG/CR-6850 approach further refined the fire intensity assumptions.) It would be inappropriate to apply an additional unsupported factor of two reduction independent of fire suppression credits applied elsewhere. This would simply be another type of fire severity factor that was not consistent with the SDP approach.

11. In Attachment 7 of its November 23, 2005, letter, FPL stated that the probability that an MOV will close due to a hot short-induced spurious operation is 0.50. That is, the spurious operation can either open or close the valve with equal likelihood. The NRC's SDP analysis assumed, for example, a spurious operation probability of 0.30 without considering the failure mode. For this case, a more realistic probability would have been

0.15 (0.30 * 0.50) since the concern only deals with the closure of the valve. Spurious opening of the valve does not lead to a loss of RCP seal cooling event.

NRC Disposition - The input that the spurious actuation probability could be reduced by a factor of two based on the fact that there was only one undesired end state (closure of a normally open valve) was contradicted by the available experimental data. Hot shorts impacting multiple conductors were the predominant failure mode noted in the Nuclear Energy Institute (NEI) tests. In a number of tests, both the open and close circuit target conductors in the surrogate MOV circuit experienced hot shorts. Hence, this was not a simple question of one versus two potential end states, rather it involved complex dependencies that are currently poorly understood. Even under the most optimistic possible interpretation of the experimental data, minor reductions in the spurious actuation likelihood (far less than a factor of two) would be expected given the case of only one undesired end state. It would be inappropriate for an analyst to speculate as to the magnitude of any such reductions without the benefit of directly applicable test data.

To explain further, the NEI surrogate MOV control circuit configuration allowed for hot shorts to impact either of the two control "direction" target conductors (i.e., hot shorts to the target conductors for either the "open" or "close" coils on the motor controller circuit would be indicated as a spurious operation). In reality, with an actual valve in the "open" position, a hot short to the "open" coil target conductor would have no impact on the circuit. That is, with the valve open, the open coil target conductor was effectively relegated to the status of a spare conductor because it was isolated by limit switches which opened once the valve reached its travel limit. A hot short to the "open" target conductor would not reposition the valve, nor would it de-energize the circuit by causing a fuse or breaker to open. Most importantly, it would not prevent a subsequent hot short to the "close" coil target conductor from spuriously closing the valve. The close direction target conductor's exposure to hot shorts and spurious operation would not change.

While future research may reveal that some minor reduction in probability for this type of case is warranted, the contention that a factor of two reduction applies is excessively optimistic and cannot be supported given the available data. Pending actual data that reflects a more representative control circuit configuration, no reductions in the spurious actuation probabilities will be applied. It is also noteworthy that this issue was a point of discussion during the EPRI expert panel deliberations (see EPRI Technical Report 1006961). The expert panel was aware of the configuration issues and explicitly chose not to modify the spurious actuation likelihood values on this basis. The expert panel spurious actuation values used in NRC Manual Chapter 0609, Appendix F, (and in NUREG/CR-6850) were used without modification in the final significance determination. Finally, this input was not consistent with FPL's input 12 below which indicated the wrong methodology was used to determine the probability of valve closure due to hot short at 0.27.

12. In Attachment 7 of its November 23, 2005, letter, FPL stated that an alternative approach as described in NUREG/CR-6850, Appendix J, is justified. The example cited was Unit 4 MOV-626. It uses 12 conductor cables which includes spare conductors mounted in a vertical configuration with no tray or conduit. This configuration is different from that listed

in Table 10-3. Therefore, the alternate approach is appropriate. A re-calculation of the closure probability through a hot short was 0.27.

NRC Disposition - NUREG/CR-6850 stated that the formula method was recommended only for cases that deviate substantially from the configurations tested by NEI. The only apparent difference for the FPL case compared to the NEI tests was the presence of additional spare conductors in their control cables (12 versus 7 conductors). This was not a substantive deviation from the tested configurations because the spare conductors play a small role in the spurious actuation behavior given that they can neither prevent nor cause a spurious actuation. (The key factors are the number of grounded conductors, the number of source conductors, and the number of target conductors.) The licensee's "lower bound" estimate was also flawed. The NEI test data clearly showed that hot shorts did not typically impact only one conductor, but rather, impacted multiple conductors. Hence, simple models do not apply. The contention that the "lower bound" hot short probability is 1/12 was without merit and contradicted by the actual experimental data. If a "lower bound" spurious actuation likelihood value was desired, the EPRI expert panel report which provided upper bound, lower bound, and best estimate values for each configuration should have been referenced. Overall, the FPL circuits appeared substantially identical to the circuits tested by NEI. Hence, the tabulated spurious actuation likelihood values were the appropriate values to apply, and this portion of the preliminary significance determination was not altered.

13. FPL stated that a common cause factor for hot shorts of adjacent cables should be considered. If this factor is considered, the probability of multiple cables having a hot short due to the same fire should be lower – not more than the maximum of the hot short of any cables involved.

    NRC Disposition - The preliminary NRC SDP analysis did not involve the probability of two cables in any fire scenario having a hot short for the event of interest to happen. The analysis was based upon the possibility that any one of a group of cables could cause the event of interest. The analysis explicitly removed multiple hot shorts from the calculation. Therefore, to apply some sort of common cause factor was inappropriate and was not incorporated into the final significance determination.

14. FPL stated that the MPR thermal hydraulic analysis demonstrated that complete seal injection seal cooling could be lost for up to 60 minutes, component cooling water seal cooling could be lost for 40 minutes, and component cooling water seal cooling could be reestablished without seal failure.

    NRC Disposition - The thermal hydraulic analysis provided was a computer simulation that did not include sufficient test validation to support the input. In addition, the historical evidence was very limited in this area. Given the uncertainty associated with the probability of the cold shock seal LOCA, this additional aspect was captured within that quantification. Therefore, the analysis assumptions associated with this area remained unchanged.

15. FPL stated that the Haddam Neck event of July 15, 1969, was not applicable when considering RCP seal performance without forced cooling.

    NRC Disposition - There are methods available for bounding or considering the failure discussed in the Haddam Neck event with varying degrees of importance assumed in a risk-informed context. To eliminate data would require complete information that demonstrated the event was not applicable. The information provided about the Haddam Neck event was not sufficient for its exclusion.

In addition to the FPL input, the NRC identified an omission in the preliminary significance determination. Specifically, the event tree portion that led directly to the RCP seal LOCA prior to restoring seal cooling was not included in the preliminary significance determination. This accident sequence should have been included. However, after including this event tree portion in the final significance determination, it had no material affect on the outcome. The performance deficiency related to Unit 4 continued to be of low to moderate safety significance, and the performance deficiency for Unit 3 remained very low safety significance.

LIST OF ATTENDEES

NUCLEAR REGULATORY COMMISSION:
H. Christensen, Deputy Director, Division of Reactor Safety (DRS), RII
C. Payne, Chief, Engineering Branch 2, DRS, RII
W. Rogers, Senior Reactor Analyst, DRS, RII
C. Evans, Regional Counsel and Enforcement Officer, RII
B. Desai, Acting Chief, Reactor Projects Branch 3, DRP, RII
S. Sparks, Senior Enforcement Specialist, RII
L. Trocine, Senior Enforcement Specialist, Office of Enforcement
R. Perch, APOB, Office of Nuclear Reactor Regulation (NRR) via teleconference
M. Reinhart, APOB, NRR via teleconference
S. Wong, APOB, NRR via teleconference
M. Franovich, APOB, NRR via teleconference
D. Harrison, APOB, NRR via teleconference
C. Liang, SBWB, NRR via teleconference

FLORIDA POWER AND LIGHT COMPANY (FPL):
T. Jones, Site Vice President, Turkey Point
R. Kundalkar, Vice President, Nuclear Engineering, FPL
J. Garcia, Chief Nuclear Engineer, FPL
S. Greenlee, Engineering Manager, Turkey Point
W. Parker, Licensing Manager, Turkey Point
C. Guey, Reliability and Risk Assessment Supervisor, FPL
V. Rubano, Engineering Project Manager, FPL
M. Ross, Assistant General Counsel, FPL
J. Lambright, Fire Risk Consultant for FPL
J. Julius, Scientech HRA Expert Consultant for FPL
T. Greene, MPR Consultant for FPL

OTHER ATTENDEES:
C. Morgan, Miami Herald (via teleconference)

Enclosure 2

MATERIAL PRESENTED BY LICENSEE

Enclosure 3

FPL
Nuclear Division

Regulatory Conference
NRC Region II

Turkey Point Station Units 3 and 4

NRC Triennial Fire Protection Inspection Reports

05000250/2005010 and 05000251/2005010

November 16, 2005

Agenda

- Introductions - Terry Jones
- Topics of Discussion – Jose Garcia
  - FPL Position on the Apparent Violation
  - Deterministic Analysis
    - Westinghouse Plug Model
    - Thermal Hydraulic Analysis
    - Refined Analysis
    - Model 93 Industry Experience
  - Significant Determination Process (SDP) Analysis
    - NRC Phase 2 Results
    - FPL Phase 2 ++ Results
- Closing Remarks – Terry Jones

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NRC Inspection Report

- NRC Triennial Fire Protection Inspection (follow up) report dated October 7, 2005
- Apparent Violation (AV) 2005010-01 cited against 10CFR50, Appendix R, Sections III.G.2 and III.G.3

NRC Apparent Violation

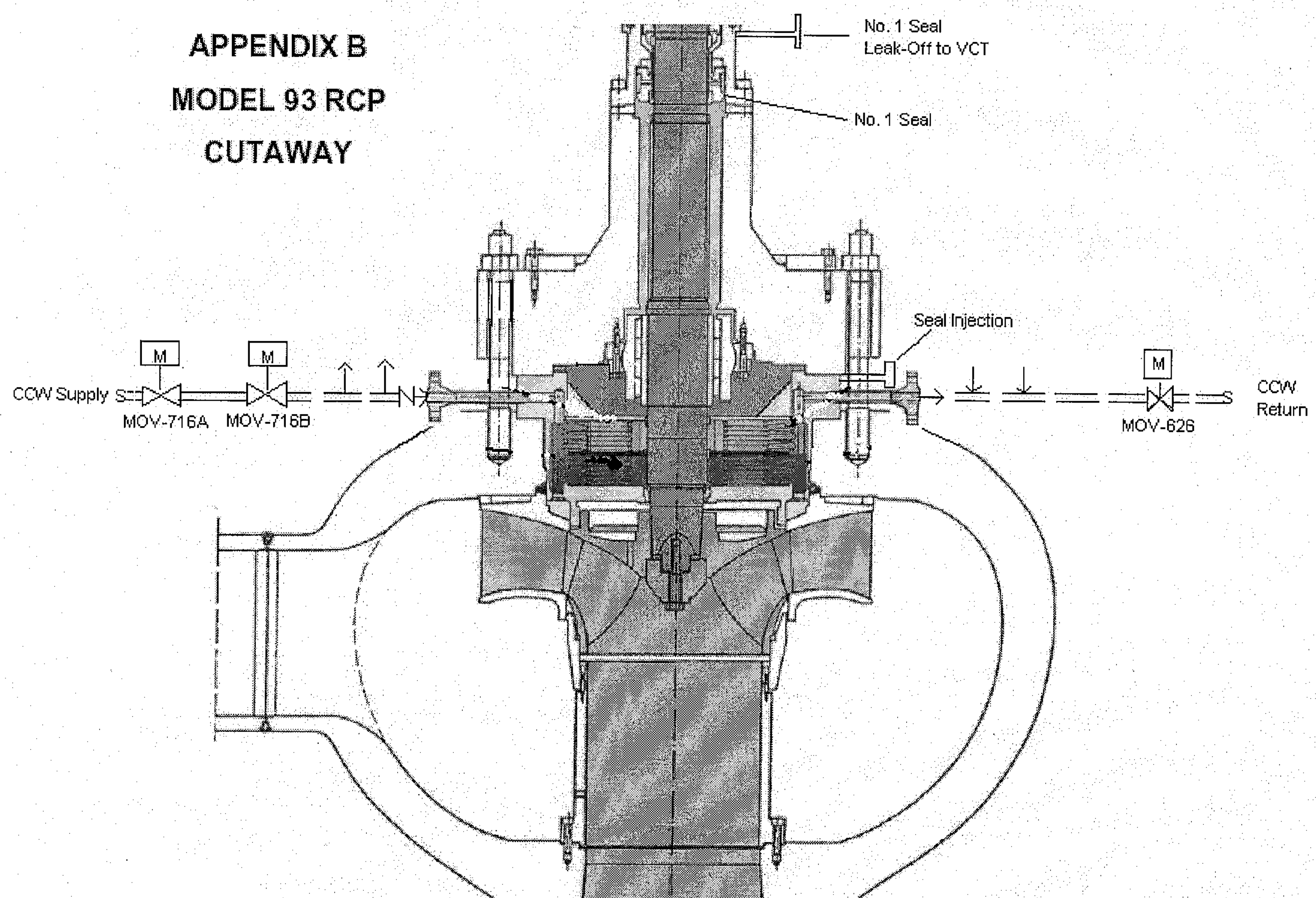
- Preliminary White Finding: NRC collective assessment of 3 Unresolved Items (URIs) as follows:
  - URI 05000251/2004007-001: Failure to prevent spurious operation of MOV-4-626, for a severe fire in 4B 4160V switchgear room
  - URI 05000251/2004007-002: Local manual operator actions to protect RCP seal package cooling not timely
  - URI 05000251/2004007-006: Local manual operator actions to protect MOV-3-716A and MOV-4-716A for control room evacuation not timely

Background

- All of these concerns deal with the actions to protect the reactor coolant pump (RCP) seals
- RCP seals are acceptably cooled by either:
  - Charging pump (CP) seal injection
  - Component cooling water (CCW) flow to the thermal barrier
- MOV-*-716A & B are the RCP thermal barrier CCW supply isolation valves
- MOV-*-626 are the RCP thermal barrier CCW return isolation valves

Reactor Coolant Pumps

APPENDIX B
MODEL 93 RCP
CUTAWAY



URI-001 Fire Zone 67

Unit 4 B Switchgear Room

- FPL concurs with Finding (URI-001)
  - FPL misclassified MOV-4-626 in the Essential Equipment List
  - FPL identified this condition
  - Entered into the corrective action program
  - Corrective action was completed in 2004
- Low safety significance
  - Delta CDF of 2.0E-8/year < 1.0E-6/year
- Additionally, Safe Shutdown Analysis is being reconstituted

Apparent Violation

- **URI-002:**
  - Failure to ensure that local manual operator actions used to verify correct alignment of MOV-3-716A and MOV-4-716A, "RCP Thermal Barrier CCE Supply Isolation Valves," and MOV-3-626, were completed in a timely manner for fires in either FZ 63 [Unit 3 B motor control center (MCC) room] or FZ 67 [4B switchgear room]
- **URI-006:**
  - Failure to ensure local manual operator actions to verify correct alignment of MOV-3-716A and MOV-4-716A were completed in a timely manner for a fire in FZ 106 [Control Room]

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Apparent Violation

- Both URIs concern timely manual actions for restoration of seal cooling and are treated together
  - Procedures allowed 20 minutes to complete manual actions to verify cooling to the RCP seal package
  - NRC IR: "However, industry analyses have determined that seal package damage could occur within 13 minutes of loss of all seal package cooling."
  - The 13 minutes is based on an estimate of time for hot RCS fluid to reach the RCP seals [WCAP-10541 (1986)]

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Background

- The estimated time to the onset of hot RCS fluid reaching the seal package depends on:
  - Volume of water between the pump casing and the seals
  - RCP #1 seal leak-off rate
- From WCAP-16396 (2005)
  - RCP with a seal purge volume of 39 gallons
  - A seal leak-off rate of 3 gpm
  - Yields 13 minutes
  - The actual time to significant heatup could vary from 8 to 40 minutes, depending on pump model and leak-off rate
- Turkey Point has Model 93 RCPs
  - RCP seal purge volume = 48.1 gallons
  - Historical average of RCP with highest seal leak-off rate is 2.44 gpm

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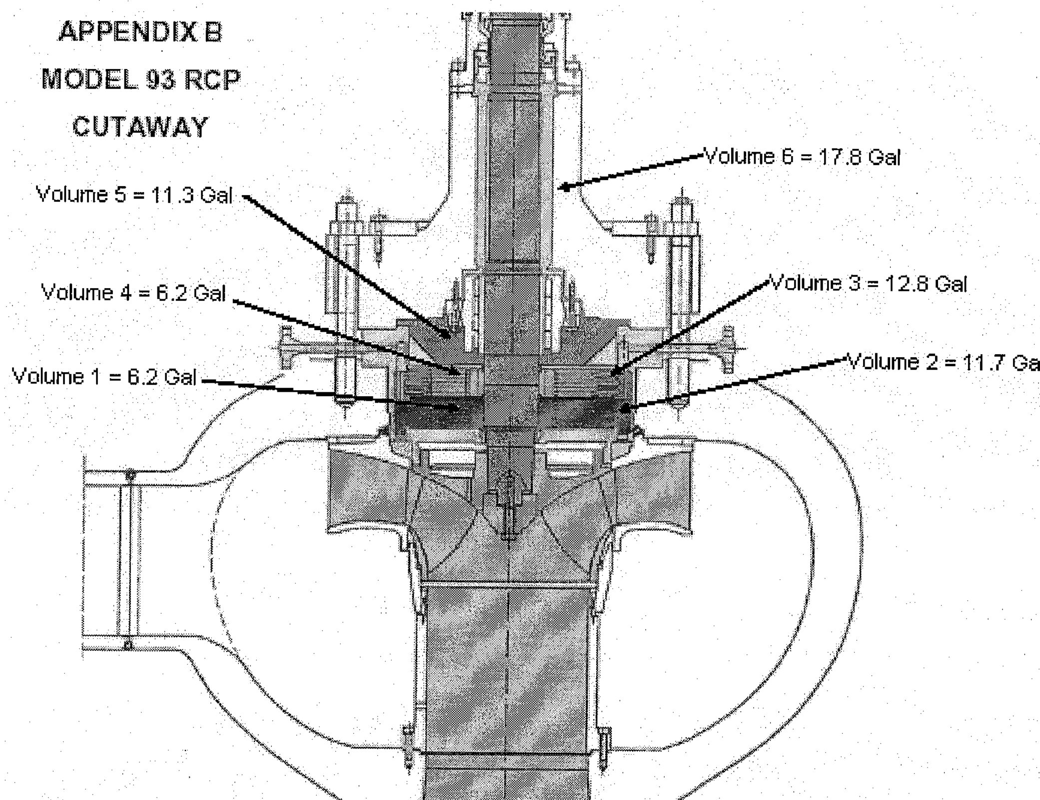
RCP Purge Volume



RCP SEAL BUFFER VOLUME = 66.0 Gallons [Volumes 1 - 6]
RCP SEAL PURGE VOLUME = 48.1 Gallons [Volumes 3 - 6]

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Turkey Point RCP #1 Seal leak-off (gpm)

Data from 10/02 – 10/05

	3A RCP	3B RCP	3C RCP	4A RCP	4B RCP	4C RCP
Average (gpm)	1.51	2.17	1.97	1.87	2.44	2.26

Results of Plant Specific Analysis

- Westinghouse TB-04-22 recommended using:
  - Plug Model to conservatively estimate time to reach the limiting temperature for restoration of cooling, or
  - Thermal Hydraulic Analysis to provide a more accurate time
- Westinghouse Plug Model
  - 48.1 gallons seal purge volume
  - 2.44 gpm RCP #1 seal leak-off rate
  - Yields approximately 20 minutes

Results of Plant Specific Analysis

- Model 93 specific Thermal Hydraulic Analysis
- Assumptions
  - Considers heat transfer to pump components
  - Considers mixing and buoyancy effects
  - Constant leak-off flow - 2.5 gpm
- Conservative acceptance criterion from TB-04-22
  - 235°F at #1 seal leak-off
  - Yields > 20 minutes

Refined Analysis

- Constant leak-off rate not consistent with test data
  - EdF Hot Thermal Shock Test (1985)
  - EdF Cold Thermal Shock Test (1987)
  - Sizewell B RCP Test (1991)
- A refined model to analyze the seal behavior was developed
  - Adds seal response as a function of temperature and rate of change of temperature

Refined Analysis Results

- The seal leak-off flow spikes, and quickly returns to normal levels as expected
- After the heat-up transient, the cooldown is slow and well behaved and seal leak-off remains at nominal level

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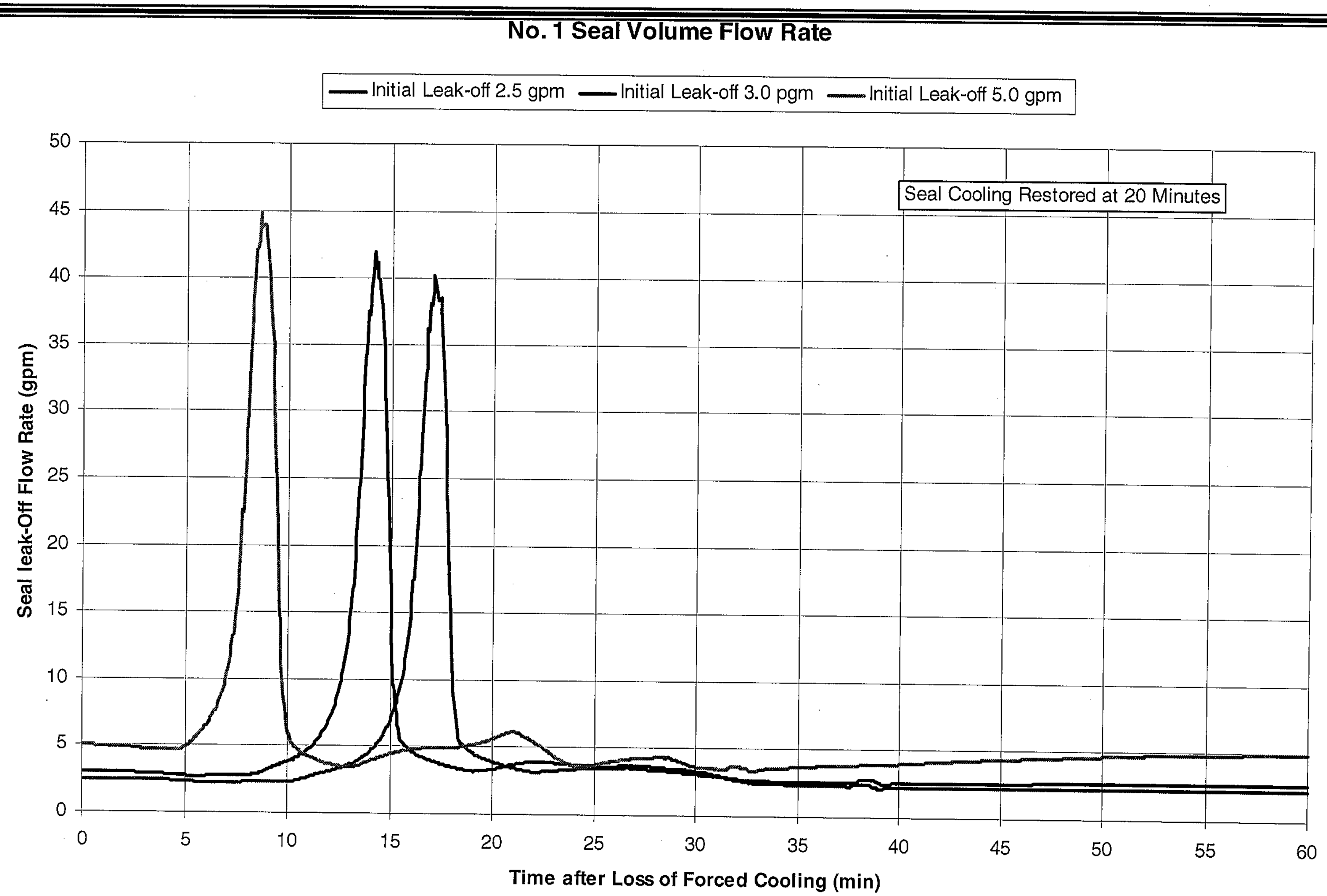
Simulated RCP Seal Leak-off Rate After Loss of Cooling



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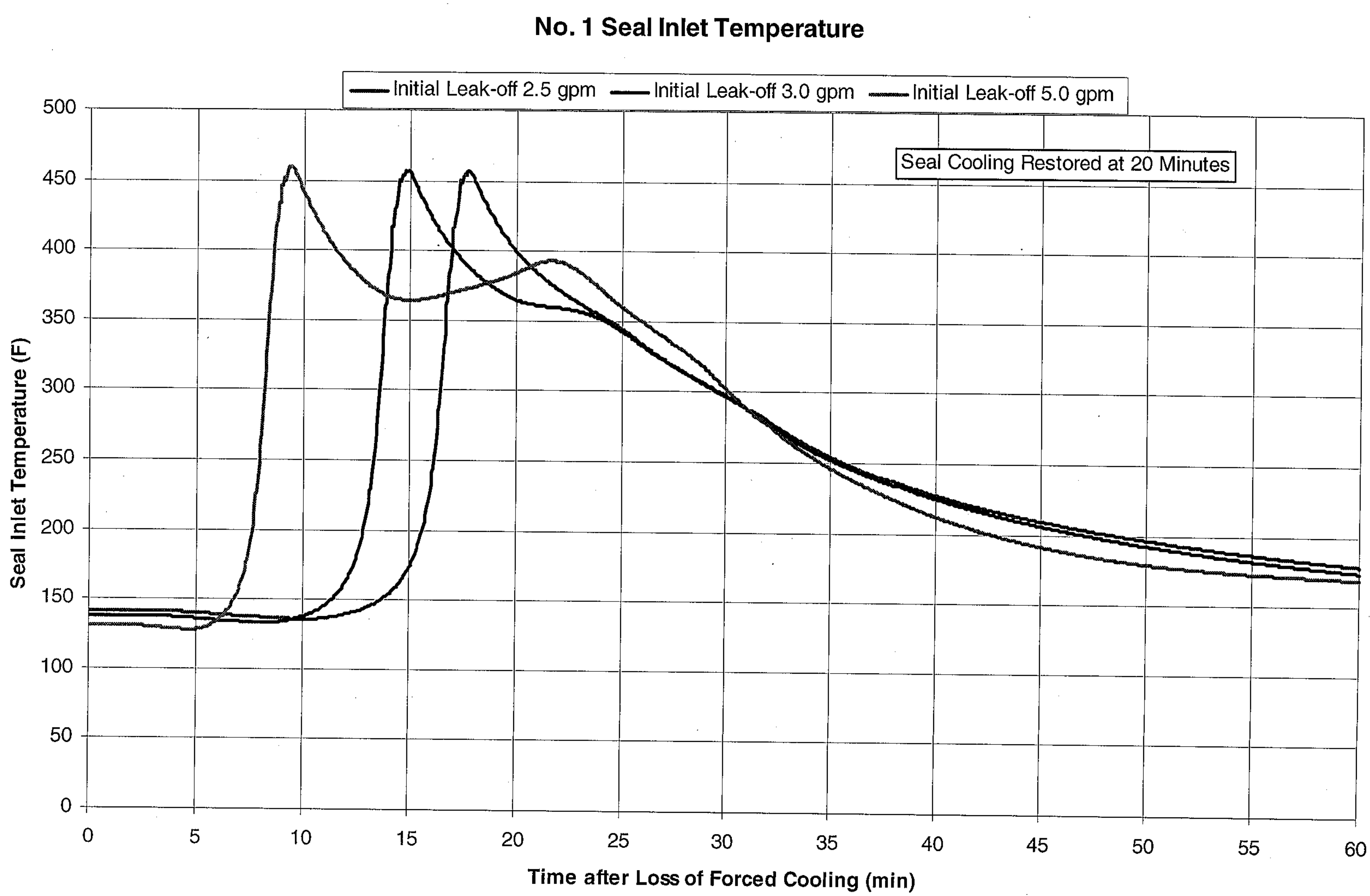
Simulated RCP Seal Temperature After Loss of Cooling



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CCW Thermal Barrier Piping

- Detailed thermal hydraulic and stress analysis performed for loss of seal cooling transients for Unit 3
  - Maximum calculated load is 2,350 psi
  - Maximum piping stress 39,773 psi versus 45,000 psi allowable for functionality
  - All support loads meet functionality criteria
- Analysis demonstrates complete piping functionality with no loss of pressure boundary or flow integrity
- By comparison between Units, our preliminary assessment is that Unit 4 analysis will also show similar results
  - Analysis to be completed in approximately 6 weeks

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Industry Experience

- There have been multiple events in the industry involving loss and recovery of seal cooling with Model 93 RCPs:
  - Ginna (1969)
    [WCAP-10541 (1986), WCAP-16396 (2005)]
    - CCW restored in 30 minutes
    - Seal injection restored in 45 Minutes
    - No abnormal leakage occurred
      - one pump seal damaged by debris
    - Damage to pump bearings and shaft were observed

Industry Experience

- Indian Point Unit 2 (1971) RCP Model 93 [WCAP-10541(1986), WCAP-16396 (2005)]
  - 2 seals without cooling for 45 minutes
  - Inspected after running them for 5 days; no seal observable damage
  - 2 seals without cooling for 60 minutes
  - No abnormal pump seal leakage occurred

Industry Experience Summary

- No indication of damage to seal as a result of high temperatures
- No indication of popping or binding of either #1 or #2 seal assemblies
- No uncontrollable leakage from the seal assemblies as a result of loss of seal cooling
- After cooling restoration, the seal leak-off returns to normal levels
- No reported damage to CCW piping
- FPL analysis also shows that seal-leak flows returned to normal levels

Conclusions

- Time for hot RCS fluid to reach the seal is approximately 20 minutes using Plug Model
- Using thermal hydraulic model and a constant leak-off rate provides > 20 minutes
- Refined Analysis demonstrates seal leak-off rates remain within acceptable values
- The manual operator action time of 20 minutes as required by Turkey Point procedures is appropriate and timely

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NRC Phase 2 SDP Summary

Unit 4 Fire Zones	SDP Phase 2 Results
61 - Unit 4 MCC	1.52E-07
67 - Unit 4 Switchgear Room	2.00E-08
98 - Unit 4 CSR Portion	6.00E-07
106- Unit 4 MCR Portion	5.80E-06
Total Delta CDF for Unit 4	**6.57E-06**

Unit 3 Fire Zones	SDP Phase 2 Results
63 - Unit 3 MCC	1.34E-07
70- Unit 3 Switchgear Room	2.00E-08
98 - Unit 3 CSR Portion	5.30E-07
106- Unit 3 MCR Portion	4.00E-06
Total Delta CDF for Unit 3	**4.68E-06**

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NRC Phase 2 SDP Equation

Unit 4 MCR FZ 106

$$\Delta CDF = DF \times \sum_{i=1}^{n} \left[F_i \times SF_i \times AF_i \times PNS_i \times CCDP_i \right]_{\text{All Scenarios}}$$

- n = number of fire scenarios evaluated for a given finding
- DF = Duration factor
- Fi = Fire frequency for the fire ignition source i
- SFi = Severity factor for scenario i
- AFi = Ignition source specific frequency adjustment factors
- PNS i = Probability of non-suppression for scenario i
- CCDPi = Conditional core damage probability for scenario i

- Probability of Hot Short is 0.3 or 0.6 depending on the control power transformer ratings
- Seal LOCA probability due to not injecting within 13 minutes 0.8 * 0.2
- **CDF Increase =**

[(1)(6E-5)(.3)(.09) + (1)(6E-5)(.6)(.09) + (.1)(4.8E-3)(.72)(.09)] x .8 x .2 x1.0 = **5.8E-06**

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Refinement to Non-Suppression for Cabinets

PNS = Probability of Non-Suppression for electrical cabinets

- Reflects a more realistic interpretation of fire events for low-voltage electrical cabinets
- Based on 12 fire events without damage to components, 1/13, or 0.08 is estimated
- 0.31 credited for Cable Spreading Room, but not other fire zones in the NRC SDP

Refinement for Control Room

PNS = Probability of Non-Suppression in control room

- 10 minute control room evacuation was assumed in the NRC phase 2 SDP based on visibility; 0.09 was used
- New EPRI 1011989(NUREG/CR-6850) provides a more realistic guidance for evaluating control room fires
- A two-zone CFAST model indicates at least 15 minutes before reaching visibility criterion; 0.01 was derived

CCDP (Conditional Core Damage Probability)

- Human Reliability Analysis (HRA) Modeling
  - Control room evacuation procedure available
  - Operators trained
  - Available time (WCAP-16141)
    - At least 95 minutes to take action
    - Once cooldown/depressurize core covered for at least 24 hours
- EPRI HRA and NRC SPAR-H methods resulted in a 0.1 or smaller probability of failure
- The above support a CCDP of 0.1 rather than the 1.0 assumed in the NRC current analysis

FPL's Phase 2 ++ SDP Results

Unit 4 Fire Zones	Revised Phase 2
61 - Unit 4 MCC	1.2E-08
67 - Unit 4 Switchgear Room	1.5E-09
98 - Unit 4 CSR Portion	1.5E-08
106- Unit 4 MCR Portion	2.4E-8
Total Delta CDF for Unit 4	**<6.0E-08**

Unit 3 Fire Zones	Revised Phase 2
63 - Unit 3 MCC	1.1E-08
70- Unit 3 Switchgear Room	1.5E-09
98 - Unit 3 CSR Portion	1.3E-08
106- Unit 3 MCR Portion	2.0E-8
Total Delta CDF for Unit 3	**<5.0E-08**

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Conclusions

- FPL concurs with URI-001 finding and that it is of low safety significance (Green)
- FPL respectfully disagrees with URI-002 and URI-006 findings
- FPL Concludes:
  - Procedure requirements are acceptable
  - There is no performance deficiency
  - Thus, no finding and no violation
- Review of the SDP analysis shows low safety significance with impact in delta CDF < 1.0E-6/year
- Additionally, we have committed to NFPA-805 requirements for Turkey Point

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NRC Apparent Violation

Open Discussion

Questions

MATERIAL PRESENTED BY NRC

AGENDA

OPEN REGULATORY CONFERENCE

TURKEY POINT NUCLEAR PLANT

NOVEMBER 16, 2005

NRC REGION II OFFICE, ATLANTA, GA.

I. OPENING REMARKS, INTRODUCTIONS AND MEETING INTENT
   Mr. H. Christensen, Deputy Director, Division of Reactor Safety (DRS)

II. NRC REGULATORY CONFERENCE POLICY
   Mr. C. Payne, Chief, Engineering Branch 2, DRS

III. STATEMENT OF THE ISSUE WITH RISK PERSPECTIVES
   Mr. C. Payne, Chief, Engineering Branch 2, DRS

IV. SUMMARY OF APPARENT VIOLATIONS
   Mr. C. Payne, Chief, Engineering Branch 2, DRS

V. LICENSEE RISK PERSPECTIVE PRESENTATION

VI. LICENSEE RESPONSE TO APPARENT VIOLATIONS

VII. BREAK / NRC CAUCUS
   Mr. H. Christensen, Deputy Director, Division of Reactor Safety (DRS)

VIII. CLOSING REMARKS
   Mr. H. Christensen, Deputy Director, Division of Reactor Safety (DRS)

Draft Apparent Violation

10 CFR 50.48 (b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of Appendix R, Section III.G. Section III.G.2 states, in part, that where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of three means of ensuring that one of the redundant trains is free of fire damage shall be provided. Section III.G.3 states that alternative shutdown capability should be provided where the protection of systems whose function is required for hot shutdown, does not satisfy the requirements of III.G.2. Section III.L of Appendix R provides requirements to be met by alternative shutdown methods. Section III.L.2.b states, in part, that "The reactor coolant makeup function shall be capable of maintaining the reactor coolant level...within the level indication in the pressurizer in PWRs."

Contrary to the above, on February 13, 2004, the inspectors identified three examples where 10 CFR 50, Appendix R requirements were not met:

a. The licensee failed to protect control circuits and cables that could cause maloperation of MOV-4-626, "RCP Thermal Barrier Component Cooling Water System Return Isolation Valve" in FZ 67. This condition existed since at least September 9, 2003, when it was first identified by the licensee.

b. The licensee failed to protect control circuits and cables that could cause maloperation of necessary RCP thermal barrier component cooling system valves MOV-3-716B and MOV-3-626 in FZ 63; and MOV-4-716B valve in FZ 67. This condition has existed since at least February 9, 2001, when the applicable procedure page was last revised.

c. The licensee failed to protect control circuits and cables that could cause maloperation of necessary RCP thermal barrier component cooling system valves in FZ 106; and did not meet the alternative shutdown capability requirements. Specifically, the licensee's procedure may not mitigate a spurious closure of valves MOV-3-716A and MOV-4-716A in a timely manner, possibly resulting in an RCP seal LOCA, and pressurizer level dropping below the indicating range. This condition has existed since at least April 24, 2002, when the applicable procedure pages were last revised.

Note: The apparent violation discussed at this Regulatory Conference is subject to further review and subject to change prior to any resulting enforcement action.