PMLevyCOLPEm Resource

From: Kitchen, Robert [Robert.Kitchen@duke-energy.com]
Sent: Monday, July 21, 2014 12:43 PM
To: Hughes, Brian
Cc: Habib, Donald; Waters, David; Wilkins, Tillie; Jackson, Tony; Sweigart, Richard L; Sylena Smith (smithse@westinghouse.com)
Subject: Draft Levy Supplemental Information
Attachments: Markup Revised for NRC Staff Review.pdf; NRC DRAFT NPD-NRC-2014-027 RAI Ltr 119 supplement enclosure only_LT.DOCX

Brian – Attached are draft documents for the Levy COLA public call scheduled for Wed, 7/23 at 2:00.

- Supplemental info re Condensate Return Change
- Supplemental info re Safety-Security interface

Hearing Identifier: Levy_County_COL_Public
Email Number: 1248

Mail Envelope Properties (A0C0A819D3B935458A74ACB9D554AE1E4B1E22C0)

Subject: Draft Levy Supplemental Information
Sent Date: 7/21/2014 12:43:28 PM
Received Date: 7/22/2014 11:23:49 AM
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Files	**Size**	**Date & Time**
MESSAGE	224	7/22/2014 11:23:49 AM
Markup Revised for NRC Staff Review.pdf	108645	
NRC DRAFT NPD-NRC-2014-027 RAI Ltr 119 supplement enclosure only_LT.DOCX	42474	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

In a public teleconference with the NRC staff held on July 10, 2014, the staff described perceived inconsistencies in the proposed licensing basis markups in Enclosures 4 and 5 of Duke Energy letter NPD-NRC-2014-021. For example, the proposed markup of UFSAR subsection 6.3.1.1.4 states, "the passive core cooling system, in conjunction with the passive containment cooling system, has diverse capability to establish long-term safe shutdown conditions in the reactor coolant system, eventually cooling the reactor coolant system <u>to about 420°F in 36 hours</u>." However, while Section 6.3.1.1 enumerates the safety design basis of the **AP1000** plant, the analysis showing the PRHR HX can bring the plant to 420°F in 36 hours is not a design basis analysis. The staff requires clear separation of the design basis claims from non-design basis claims; and is of the opinion that the words "to about 420°F in 36 hours" retained in subsection 6.3.1.1.4 misrepresent the design basis capabilities of the **AP1000** plant. The NRC staff requested Duke Energy to provide a licensing basis markup that resolves this inconsistency, or provide justification that the existing licensing basis markup is fit for its intended purpose.

The following identify the revisions to the licensing basis markup provided in Duke Energy letter of June 27, 2014, Serial: NPD-NRC-2014-021 to address the NRC staff concern detailed above. Duke Energy will document these revisions in a supplemental response that encompasses both Enclosure 4 and Enclosure 5.

<u>DCD Subsection 6.3.1.1.4, paragraphs 1 and 3 from NPD-NRC-2014-021, Enclosure 4</u>

6.3.1.1.4 Safe Shutdown

The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system, has diverse capability to establish long-term safe shutdown conditions in the reactor coolant system, eventually cooling the reactor coolant system to about 420°F in 36 hours, with or without availability of the reactor coolant pumps.

…

In most sequences the operators would return the plant to normal system operations and terminate passive system operation in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to establish and maintain long-term safe shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

Revised paragraphs 1 and 3 of DCD subsection 6.3.1.1.4

The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions in the reactor coolant system.

…

In most sequences the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to establish and maintain long-term safe shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

Revised DCD subsection 6.3.1.1.6 (no corresponding change in this section in NPD-NRC-2014-021

Revise the last sentence of the paragraph in this subsection to reference Subsection 6.3.1.3 instead of Subsection 6.3.1.2

DCD Subsection 6.3.1.2.1 from NPD-NRC-2014-021, Enclosure 4

6.3.1.2.1 Post-Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition for 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in subsection 7.4.1.1.

Revised DCD subsection 6.3.1.2.1

6.3.1.2.1 Post-Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in subsection 7.4.1.1.

DCD Subsection 6.3.3, paragraphs 7 and 8, from NPD-NRC-2014-021, Enclosure 4

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III, and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition as described in subsection 19E.4.10.2. A non-bounding, conservative estimation of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

Revised DCD Subsection 6.3.3, paragraphs 7 and 8

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III, and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition of 420°F within 36 hours as described in subsection 19E.4.10.2. A non-bounding, conservative analysis of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

DCD Subsection 7.4.1.1, paragraph 8, from NPD-NRC-2014-021, Enclosure 4

7.4.1.1 Safe Shutdown Using Safety-Related Systems

…

A gutter located at the operating deck elevation collects condensate from the inside of the containment shell. Valves located in drain lines from the gutter to the containment waste sump close on a passive residual heat removal heat exchanger actuation signal. This action diverts the condensate to the in-containment refueling water storage tank. The system provides core decay heat removal in this configuration without an increase in the containment water level.

Revised DCD Subsection 7.4.1.1, paragraph 8

7.4.1.1 Safe Shutdown Using Safety-Related Systems

…

A gutter located at the operating deck elevation collects condensate from the inside of the containment shell. Valves located in drain lines from the gutter to the containment waste sump close on a passive residual heat removal heat exchanger actuation signal. This action diverts the condensate to the in-containment refueling water storage tank. The system provides core decay heat removal in this configuration with a limited increase in the containment water level.

Description of Change for Sheet 6 of Table 19.59-18 in NPD-NRC-2014-021, Enclosure 4

Revision to description is highlighted as follows:

The following change would be made on Sheet 6 of Table 19.59-18, “PRA-Based Insights” and in the corresponding table of the **AP1000** Probabilistic Risk Assessment (APP-GW-GL-022 rev 8):

DCD Subsection 19E.4.10.2 from NPD-NRC-2014-021, Enclosure 4, first paragraph

19E.4.10.2 Shutdown Temperature Evaluation

As discussed in Section 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to 420°F or below within 36 hours after shutdown following a non-LOCA event. The following summarizes a non- bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

Revised DCD Subsection 19E.4.10.2, first paragraph

19E.4.10.2 Shutdown Temperature Evaluation

As discussed in subsection 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

Levy Nuclear Plant Units 1 and 2 (LNP) Supplemental Response to NRC Request for Additional Information Letter No. 119 Related to SRP Section 13.6 for the Levy Nuclear Plant, Units 1 and 2 Combined License Application, Dated 05/30/14

NRC RAI #	Duke Energy RAI #	Duke Energy Response
13.06.01-1	L-1107	Supplemental response enclosed–see the following pages
13.06.01-1	L-1104	July 8, 2014; NPD-NRC-2014-020

NRC Letter No.: LNP-RAI-LTR-119

NRC Letter Date: May 30, 2014

NRC Review of Final Safety Analysis Report

Text of NRC RAI:

QUESTIONS

13.06.01-1

Background Data:

In Section 13.6, Final Safety Analysis Report (FSAR), the applicant incorporated by reference the AP1000 Design Control Document (DCD).

AP1000 Design Control Document, Section 13.6 Security states:

"The Security Plan consists of the "AP1000 Physical Security Plan," Training and Qualification Plan, and Safeguards Contingency Plan. The Security Plan will be submitted to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements for 10 CFR 52.79(a)(35) and 10 CFR 52.79(a)(36). The Security Plan will meet the requirements of 10 CFR 52.98(c). The plan is classified as Security Safeguards Information and is withheld from public disclosure pursuant to 10 CFR 73.21. Additionally, the "AP1000 Interim Compensatory Measures Report" (Reference 2), the "AP1000 Enhancement Report" (Reference 3), and the "AP1000 Safeguards Assessment Report" (Reference 4) are submitted to the Nuclear Regulatory Commission as separate licensing documents to establish the design of the AP1000 Security Systems. Each document is classified as Security Safeguards information and is withheld from public disclosure pursuant to 10 CFR 73.21."

In a letter dated June 3, 2011, Progress Energy Florida, Inc. (PEF) submitted their Security Plans revision 4 to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements for 10 CFR 52.79(a)(35) and 10 CFR 52.79(a)(36).

Section 8 of the Safeguards Contingency Plan, provides a description that the Levy County Nuclear Power Plant, Units 1 and 2, protective strategy and response scenarios are based on those as described and evaluated in the "AP1000 Safeguards Assessment Report" (Westinghouse Technical Report APP-GW-GLR-066).

The AP1000 Safeguards Assessment Report, Revision 5, (APP-GW-GLR-66) (TR-94) describes the AP1000 physical protection system and analyzes the ability of the AP1000 security design to provide protection against malevolent attempts to commit radiological sabotage using elements of the Design Basis Threat (DBT) as contained in 10 CFR 73.1(a)(1). The TR-94 report is intended to support the licensing of the portion of the AP1000 security system that is within the scope of the Design Certification (DC). TR-94, Section 3, describes how the process of the target set identification for the AP1000 was established by using the standard methodology to determine those structures, systems and components (SSC) that require protection in order to meet the performance objectives of the AP1000 physical protection system.

eRAI Question

The NRC staff requests clarification pertaining to how the applicant, once licensed, will analyze and identify changes in the site-specific conditions related to the AP1000's structures, systems, and components (SSCs) (described in certain technical reports), resulting from changes made to the Levy County Nuclear Power Plant COL between issuance of the COL and the security program implementation milestones provided in FSAR Table 13.4-201 to ensure that the security plan continues to meet 10 CFR 73.55(b)(4). Also, clarify how the applicant, once licensed, will ensure that the as-built plant continues to meet all physical protection program design and performance criteria in 10 CFR 73.55 at the time the physical protection program is implemented.

The applicant's response should:

a. Describe how all changes of SSCs and related design information are reviewed for any impact on the physical protection program.

b. Describe how the physical protection program, to include the security plans (consisting of the physical security plan, training and qualification plan, safeguards contingency plan, and cyber security plan), will be revised to address changes that affect (both beneficial and adverse) the protective strategy with the as-built configuration.

<u>Regulatory Basis:</u>

The provisions of 10 CFR 73.55(b)(4), require, in part, that, "(1) The licensee shall establish and maintain a physical protection program, to include a security organization, which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.
(2) To satisfy the general performance objective of paragraph (b)(1) of this section, the physical protection program must protect against the design basis threat of radiological sabotage as stated in § 73.1.
(3) The physical protection program must be designed to prevent significant core damage and spent fuel sabotage. Specifically, the program must:
(i) Ensure that the capabilities to detect, assess, interdict, and neutralize threats up to and including the design basis threat of radiological sabotage as stated in § 73.1, are maintained at all times.
(ii) Provide defense-in-depth through the integration of systems, technologies, programs, equipment, supporting processes, and implementing procedures as needed to ensure the effectiveness of the physical protection program.
(4) The licensee shall analyze and identify site-specific conditions, including target sets, that may affect the specific measures needed to implement the requirements of this section and shall account for these conditions in the design of the physical protection program.

The provisions of 10 CFR 50.54(p)(1) require, in part, that, "The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with appendix C of part 73 of this chapter for affecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may not make a change which would decrease the effectiveness of a physical security plan, or guard training and qualification plan, or cyber security plan prepared under § 50.34(c) or § 52.79(a), or part 73 of this chapter, or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base,

Responsibility Matrix) contained in a licensee safeguards contingency plan prepared under § 50.34(d) or § 52.79(a), or part 73 of this chapter, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for amendment to the licensee's license under § 50.90.
(2) The licensee may make changes to the plans referenced in paragraph (p)(1) of this section, without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of changes to the plans made without prior Commission approval for a period of 3 years from the date of the change, and shall submit, as specified in § 50.4 or § 52.3 of this chapter, a report containing a description of each change within 2 months after the change is made. Prior to the safeguards contingency plan being put into effect, the licensee shall have:
(i) All safeguards capabilities specified in the safeguards contingency plan available and functional;
(ii) Detailed procedures developed according to appendix C to part 73 of this chapter available at the licensee's site; and
(iii) All appropriate personnel trained to respond to safeguards incidents as outlined in the plan and specified in the detailed procedures.

Duke RAI ID #: L-1107

DEF Response to NRC RAI:

A response to these questions was provided previously in NPD-NRC-2014-020 dated July 8, 2014. This response provides additional information in response to discussions with the NRC staff on a July 16, 2014 public call. Text will be added to LNP COLA FSAR Chapter 13 to describe configuration control evaluations performed to identify impacts to various programs including Security starting at COL issuance.

A future revision of the LNP COLA will reflect the changes discussed in this response.

Associated LNP COL Application Revisions:

COLA Part 2, FSAR Chapter 13 will be revised to add text to Section 13AA.1.1.1.1.3 as shown below. The left-hand margin annotation for this added text will be "LNP COL 13.1-1"

13AA.1.1.1.1.3 Review and Approval of Plant Design Features

Design engineering review and approval is performed in accordance with the reactor technology vendor QA Program and Section 17.1. The reactor technology vendor is responsible for design control of the power block. Verification is performed by competent individuals or groups other than those who performed the original design. Design issues arising during construction are addressed and implemented with notification and communication of changes to the manager in charge of Nuclear Engineering for review. As systems are tested and approved for turnover and operation, control of design is turned over to plant staff. The manager in charge of Nuclear Engineering, along with functional managers and staff, assumes responsibility for review and approval of modifications, additions, or deletions in plant design features, as well as control of design documentation, in accordance with the Operational QA Program. For changes that impact the Current Licensing Basis, applicability determinations/departure evaluations are performed to review the proposed activities in accordance with the requirements of 10 CFR 52 Appendix D Section VIII.B.5. Beginning at COL issuance, these evaluations review proposed changes and activities to identify potential impacts (both beneficial and adverse) on the functions and performance of the elements of various programs. Examples of the programs reviewed include, but are not limited to Security, Emergency Planning, Inservice Testing and Inspection Programs, and Adverse Environmental Impacts. Design control becomes the responsibility of the manager in charge of Nuclear Engineering prior to loading fuel. During construction, startup, and operation, changes to human-system interfaces of control room design are approved using a human factors engineering evaluation addressed within Chapter 18. See Organization Charts, Figures 13.1-201 and 13AA-201 for reporting relationships.

Attachments/Enclosures:

None