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SOUTHERN COMPANY

AUG 2 8 2014

Docket Nos.: 52-025
52-026

ND-14-1108
10 CFR 50.90
10 CFR 52.63

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Supplement to Request for License Amendment and Exemption Regarding
Passive Core Cooling System (PXS) Condensate Return (LAR-13-024S2)

Ladies and Gentlemen:

By letter dated April 11, 2014, Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, requested an amendment to Combined License (COL) Numbers NPF-91 and NPF-92, for VEGP Units 3 and 4, respectively. Pursuant to the provisions of 10 CFR 52.63(b)(1), SNC also requested an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule for the plant-specific DCD Tier 1 material departures.

The proposed amendment would revise the plant-specific Tier 1 and associated Tier 2 material to describe an increase in the efficiency of the return of condensate utilized by the passive core cooling system (PXS) to the in-containment refueling water storage tank (IRWST) to support the capability for long term cooling.

This departure request is also supported by the technical documents provided with the departure request submitted for the Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030 on February 7, 2014. SNC LAR-13-024 Supplement 1 (ND-14-0624 submitted April 18, 2014) addressed a Nuclear Regulatory Commission (NRC) Staff query regarding the applicability of those technical documents submitted on the Levy Nuclear Plant, Units 1 and 2, docket.

As a result of the review of the application the NRC has issued several requests for additional information (RAIs) on the Levy Nuclear Plant, Units 1 and 2, dockets. For convenience, Enclosure 4 provides a complete listing of the pertinent NRC requests for additional information, along with the associated Levy Nuclear Plant docketed submittal correspondence addressed in Enclosure 5. (Note that Enclosures 1, 2 and 3 were provided with the original license amendment request referenced above.) Supplement 1 included no additional Enclosures.

SNC personnel have reviewed these RAIs and the associated Levy docketed submittal correspondence and determined that the Levy docketed information is also pertinent to the Vogtle LAR review. As such, SNC hereby incorporates by reference Levy Nuclear Plant previously docketed information (as specifically identified in Enclosure 5) as supplemental information to support the SNC docketed LAR-13-024.

Pursuant to 10 CFR 2.390, several of the incorporated by reference Duke Energy submittal enclosures should not be disclosed to the public. Correspondence with respect to the copyright or proprietary aspects of the items listed in Enclosure 5 or the supporting Westinghouse affidavit should reference the corresponding Westinghouse Application letter (“CAW” letter number identified in Enclosure 5) and should be addressed to the point of contact listed therein. Correspondence with respect to proprietary aspects of the information incorporated by reference should also be addressed to Brian H. Whitley at the contact information within this letter. SNC also makes a 10 CFR 2.390(b)(4) request for withholding (Enclosure 6) based on the reasons listed in the corresponding Westinghouse Application letter (“CAW” letter number identified in Enclosure 5); SNC's contractual obligation to seek proprietary treatment of the information; and the information is the type that has historically been held in confidence by SNC.

Additionally, the incorporated by reference information specifically identified in Enclosure 5 included some associated supplemental requested changes to the licensing basis documents. Corresponding associated supplemental requested changes to the Vogtle licensing basis documents are identified and requested in Enclosure 7. These should be considered along with the requested changes to the licensing basis document identified and requested in Enclosure 3 of the original LAR-13-024.

SNC's responses and supplemental requests do not impact the original LAR scope or Technical Evaluation beyond the information provided in the incorporated by reference correspondence. However, the Significant Hazards Consideration Determination and the associated Exemption Request have been determined to be impacted by the incorporated by reference information. Therefore, revisions to the Significant Hazards Consideration Determination and the associated Exemption Request are provided as Enclosures 8 and 9, respectively.

In addition, the requested approval date has been re-evaluated and this LAR is now requested to be approved by April 11, 2015.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this LAR supplement by transmitting a copy of this letter and enclosures to the designated State Official. This letter contains no regulatory commitments.

Should you have any questions, please contact Mr. Jason Redd at (205) 992-6435.

Mr. Brian H. Whitley states that: he is the Regulatory Affairs Director of Southern Nuclear Operating Company; he is authorized to execute this oath on behalf of Southern Nuclear Operating Company; and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

B. H. Whitley

B. H. Whitley

BHW/ERG/kms

Sworn to and subscribed before me this 28th *day of* August, *2014*

Notary Public: Kristin Marie Seibert

My commission expires: August 16, 2016

KRISTIN MARIE SEIBERT NOTARY PUBLIC ALABAMA STATE AT LARGE

Enclosures: 4) Docketed Requests for Additional Information Responses Applicable to License Amendment Regarding Passive Core Cooling System (PXS) Condensate Return (LAR-13-024S2)

5) Incorporated by Reference Information Applicable to License Amendment Regarding Passive Core Cooling System (PXS) Condensate Return (LAR-13-024S2)

6) Southern Nuclear Operating Company Affidavit for Proprietary Information

7) Supplement Revisions to Licensing Basis Documents

8) Revision to LAR-13-024 Enclosure 1 Section 4.3 – Significant Hazards Consideration Determination (LAR-13-024S2)

9) Revision to LAR-13-024 Enclosure 2 – Request for Exemption Regarding Passive Core Cooling System (PXS) Condensate Return

cc:

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Mr. D. A. Bost (w/o enclosures)
Mr. B. L. Ivey
Mr. M. D. Rauckhorst (w/o enclosures)
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File AR.01.02.06

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<u>Other</u>
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Mr. S. Blanton, Balch Bingham
Mr. J. R. Bouknight, South Carolina Electric & Gas Company
Mr. D. Kersey, South Carolina Electric & Gas Company
Mr. B. Kitchen, Duke Energy
Mr. S. Franzone, Florida Power & Light

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-14-1108

Enclosure 4

Docketed Requests for Additional Information Responses Applicable to License Amendment Regarding Passive Core Cooling System (PXS) Condensate Return

(LAR-13-024S2)

Levy Nuclear Plant, Units 1 and 2
Responses to NRC Requests for Additional Information
Letter No. 116 Related to SRP Sections 15.02.06 and 06.03, dated March 6, 2014,
(ADAMS Accession No. ML14077A609)
Letter No. 117 Related to SRP Section 06.03, dated April 10, 2014,
(ADAMS Accession No. ML14100A040)
and
Letter No. 118 Related to SRP Section 06.03, dated April 24, 2014
(ADAMS Accession No. ML14114A050)

NRC RAI No.	Duke Energy RAI No.	Levy Nuclear Plant Docketed Submittal	ADAMS Accession No.
15.02.06-1	L-1081	NPD-NRC-2014-017, dated June 19, 2014	ML14171A453
15.02.06-1	L-1081	NPD-NRC-2014-024, dated July 24, 2014	ML14206A951
15.02.06-2	L-1082	NPD-NRC-2014-021, dated June 27, 2014	ML14182A106
15.02.06-3	L-1085	NPD-NRC-2014-017, dated June 19, 2014	ML14171A453
06.03-1	L-1086	NPD-NRC-2014-014, dated May 5, 2014	ML14126A699
06.03-2	L-1087	NPD-NRC-2014-016, dated June 12, 2014	ML14164A444
06.03-3	L-1088	NPD-NRC-2014-016, dated June 12, 2014	ML14164A444
06.03-4	L-1089	NPD-NRC-2014-022, dated July 1, 2014	ML14183B342
06.03-5	L-1090	NPD-NRC-2014-021, dated June 27, 2014	ML14182A106
06.03-6	L-1091	NPD-NRC-2014-014, dated May 5, 2014	ML14126A699
06.03-7	L-1092	NPD-NRC-2014-012, dated April 17, 2014	ML14112A371
06.03-8	L-1093	NPD-NRC-2014-012, dated April 17, 2014	ML14112A371
06.03-9	L-1094	NPD-NRC-2014-015, dated May 19, 2014	ML14141A015
06.03-10	L-1096	NPD-NRC-2014-021, dated June 27, 2014	ML14182A106
06.03-11	L-1097	NPD-NRC-2014-021, dated June 27, 2014	ML14182A106
06.03-12	L-1099	NPD-NRC-2014-021, dated June 27, 2014	ML14182A106
General	--	NPD-NRC-2014-028, dated July 24, 2014	ML14206A953
General	--	NPD-NRC-2014-023, dated July 10, 2014	ML14196A074

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-14-1108

Enclosure 5

Incorporated by Reference Information Applicable to License Amendment Regarding Passive Core Cooling System (PXS) Condensate Return

(LAR-13-024S2)

Southern Nuclear Operating Company (SNC) personnel have reviewed the NRC requests for additional information (RAIs) and the associated Levy docketed responses and determined that the response information is also pertinent to the NRC review of the Vogtle Electric Generating Plant Units 3 and 4 license amendment request regarding Passive Core Cooling System (PXS) Condensate Return. As such, SNC incorporates by reference the (Levy Nuclear Plant previously docketed) information specifically identified below as supplemental information to support the SNC docketed LAR-13-024.

Supplemental information item 1

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-012**, dated April 17, 2014 (ADAMS Accession No. ML14112A371), Enclosures 1, 2, 3, and 4 are hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. These incorporated by reference Enclosures address RAIs 06.03-7 and 06.03-8.

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letter No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated 03/06/2014 (PROPRIETARY)
- Enclosure 2. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letter No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated 03/06/2014 (NON-PROPRIETARY VERSION)
- Enclosure 3. Westinghouse Application Letter CAW-14-3906 and Affidavit
- Enclosure 4. Proprietary Information Notice and Copyright Notice

Note that Enclosure 5 of this referenced Levy submittal provides additional Levy COL Application Revisions which have already been addressed and included as appropriate (for example, left margin annotation changes are not applicable to Vogtle UFSAR) in the SNC Vogtle LAR-13-024.

Supplemental information item 2

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-014**, dated May 5, 2014 (ADAMS Accession No. ML14126A699), Enclosures 1, 2, and 3 are hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. These incorporated by reference Enclosures address RAIs 06.03-1 and 06.03-6.

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letter No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated 03/06/2014
    - A. Responses to NRC RAIs 06.03-1 and 06.03-6 on Condensate Return Licensing Submittal (Proprietary)
    - B. Responses to NRC RAIs 06.03-1 and 06.03-6 on Condensate Return Licensing Submittal (Nonproprietary)
- Enclosure 2. Westinghouse Application Letter CAW-14-3907 and Affidavit
- Enclosure 3. Proprietary Information Notice and Copyright Notice

Supplemental information item 3

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-015**, dated May 19, 2014 (ADAMS Accession No. ML14141A015), Enclosure 1 is hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. This incorporated by reference Enclosure addresses RAI 06.03-9 (Nonproprietary).

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letter No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated 03/06/2014

Supplemental information item 4

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-016**, dated June 12, 2014 (ADAMS Accession No. ML14164A444), Enclosures 1, 2, and 3 are hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. These incorporated by reference Enclosures address RAIs 06.03-2 and 06.03-3.

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letter No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated 03/06/2014
    - A. Responses to NRC RAIs 06.03-2 and 06.03-3 on Condensate Return Licensing Submittal (Proprietary)
    - B. Responses to NRC RAIs 06.03-2 and 06.03-3 on Condensate Return Licensing Submittal (Nonproprietary)
- Enclosure 2. Westinghouse Application Letter CAW-14-3960 and Affidavit
- Enclosure 3. Proprietary Information Notice and Copyright Notice

Supplemental information item 5

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-017**, dated June 19, 2014 (ADAMS Accession No. ML14171A453), Enclosure 1 is hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. The incorporated by reference Enclosure addresses RAIs 15.02.06-1 and 15.02.06-3 (Nonproprietary).

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letter No. 116 Related to SRP Sections 15.02.06 and 06.03 for the Combined License Application, dated 03/06/2014, No. 117 Related to SRP Section 06.03, dated 04/10/2014, and No. 118 Related to SRP Section 06.03, dated 04/24/2014

Supplemental information item 6

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-021**, dated June 27, 2014 (ADAMS Accession No. ML14182A106), Enclosures 1, 2, and 3 are hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. These incorporated by reference Enclosures address RAIs 15.02.06-2, 06.03-5, 06.03-10, 06.03-11 and 06.03-12.

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letters No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated March 6, 2014, No. 117 Related to SRP Section 06.03, Dated April 10, 2014, and No. 118 Related to SRP Section 06.03, dated April 24, 2014
    - A. Responses to NRC RAIs 15.02.06-2 and 06.03-5 on Condensate Return Licensing Submittal (Proprietary)
    - B. Responses to NRC RAIs 15.02.06-2 and 06.03-5 on Condensate Return Licensing Submittal (Nonproprietary)
- Enclosure 2. Westinghouse Application Letter CAW-14-3961 and Affidavit
- Enclosure 3. Proprietary Information Notice and Copyright Notice

Enclosure 4 of the referenced Levy docketed letter also provided revisions for the proposed changes to the Licensing Basis Documents associated with LAR-13-024. These supplemental revisions are reproduced for the Vogtle docket in redline/strikeout format as **Enclosure 6** to this Vogtle LAR supplement (LAR-13-024S2).

Additionally, Enclosure 5 of the referenced Levy submittal is Levy specific and is NOT incorporated by reference for the Vogtle docket.

Supplemental information item 7

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-022** (note that cover letter identifies this as NPD-NRC-2014-021), dated July 1, 2014 (ADAMS Accession No. ML14183B342), Enclosure 1 is hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. This incorporated by reference Enclosure addresses RAI 06.03-4 (Nonproprietary).

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Response to NRC Request for Additional Information Letters No. 116 Related to SRP Sections 15.02.06 and 06.03 for the Combined License Application, Dated March 6, 2014, No. 117 Related to SRP Section 06.03, Dated April 10, 2014, and No. 118 Related to SRP Section 06.03, dated April 24, 2014

Supplemental information item 8

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-024**, dated July 24, 2014 (ADAMS Accession No. ML14206A951), Enclosures 1, 2, and 3 are hereby incorporated by reference as applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. These incorporated by reference Enclosures address RAI 15.02.06-1.

- Enclosure 1. Levy Nuclear Plant Units 1 and 2 (LNP) Supplemental Response to NRC RAI Letter No. 116 - SRP Sections 6.3 and 15.2.6
    - A. Supplemental Response to NRC RAI 15.02.06-1 on Condensate Return (Proprietary)
    - B. Supplemental Response to NRC RAI 15.02.06-1 on Condensate Return (Nonproprietary)
- Enclosure 2. Westinghouse Application Letter CAW-14-3966 and Affidavit
- Enclosure 3. Proprietary Information Notice and Copyright Notice

Supplemental information item 9

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-028**, dated July 24, 2014 (ADAMS Accession No. ML14206A953), also provides applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. Enclosures 1, 2 and 3 of this referenced Levy submittal are simply references to Enclosures 1, 2 and 3 of NPD-NRC-2014-021 which are already incorporated by reference by Supplemental Information item 6 above.

Enclosure 4 of the referenced Levy submittal also provided revisions for the proposed changes to the Licensing Basis Documents associated with LAR-13-024. These supplemental revisions are reproduced for the Vogtle docket in redline/strikeout format as **Enclosure 6** to this Vogtle LAR supplement (LAR-13-024S2).

Additionally, Enclosure 5 of the referenced Levy submittal is Levy specific and is NOT incorporated by reference for the Vogtle docket.

Supplemental information item 10

Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter **NPD-NRC-2014-023**, dated July 10, 2014 (ADAMS Accession No. ML14196A073), also provides applicable information for the Vogtle Electric Generating Plant Units 3 and 4 LAR-13-024. Enclosures 1, 2, 3, 4 and 6 of this referenced Levy submittal are simply references to Enclosures 1, 2, 3, 4 and 6 of NPD-NRC-2014-005 which have already been addressed by the original Vogtle Units 3 and 4 LAR-13-024 (April 11, 2014) and the first supplement, LAR-13-024S (April 18, 2014).

Enclosure 5 of the referenced Levy submittal also provided revisions for the proposed exemption request associated with the changes to the plant-specific Tier 1 Licensing Basis Documents as submitted along with Vogtle Units 3 and 4 LAR-13-024. The pertinent revisions are reflected for the Vogtle docket in redline/strikeout format as **Enclosure 9** to this Vogtle Units 3 & 4 LAR supplement (LAR-13-024S2).

Additionally, Enclosure 7 of the referenced Levy submittal is Levy specific and is NOT incorporated by reference for the Vogtle docket.

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-14-1108

Enclosure 6

Southern Nuclear Operating Company Affidavit for Proprietary Information

(LAR-13-024S2)

(This cover and 2 pages)

<u>Affidavit of B. H. Whitley</u>

1. My name is Brian H. Whitley. I am the Director, Regulatory Affairs, for Southern Nuclear Operating Company (SNC). I have been delegated the function of reviewing proprietary information sought to be withheld from public disclosure and am authorized to apply for its withholding on behalf of SNC.

2. I am making this affidavit on personal knowledge, in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations, and in conjunction with SNC's filings on dockets 52-025 and 52-026 requesting license amendment LAR-13-024S2. I have personal knowledge of the criteria and procedures used by SNC to designate information as a trade secret, privileged, or as confidential commercial or financial information.

3. Based on the reason(s) at 10 CFR 2.390(a)(4), this affidavit seeks to withhold from public disclosure the incorporated by reference specific material identified as Proprietary in Enclosure 5 of this Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, License Amendment Request 13-024S2 (dockets 52-025 and 52-026).

4. The following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

    a. The information sought to be withheld from public disclosure has been held in confidence by SNC and Westinghouse Electric Company.

    b. The information is of a type customarily held in confidence by SNC and Westinghouse and not customarily disclosed to the public.

    c. The release of the information might result in the loss of an existing or potential competitive advantage to SNC and/or Westinghouse.

    d. Other reasons identified in the various incorporated by reference Westinghouse Application Letters and Affidavits as identified in Enclosure 5 of this Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, License Amendment Request 13-024S2 (dockets 52-025 and 52-026).
    Those reasons are incorporated here by reference.

    e. The material to be withheld has been previously identified as proprietary (see ADAMS accession numbers ML14112A371, ML14126A699, ML14164A444, ML14182A106, and ML14206A951). The applications for withholding associated with those accession numbers are also adopted in support of this affidavit.

5. Additionally, release of the information may harm SNC because SNC has a contractual relationship with the Westinghouse Electric Company regarding proprietary information. SNC is contractually obligated to seek confidential and proprietary treatment of the information.

6. To satisfy the requirements of 10 CFR 2.390(b)(1)(i)(B) and (b)(1)(ii)(E), SNC requests that the incorporated by reference specific material identified as Proprietary in Enclosure 5 of this Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, License Amendment Request 13-024S2 (dockets 52-025 and 52-026) be withheld in their entirety.

7. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

8. To the best of my knowledge and belief, the information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method.

I declare under penalty of perjury that the foregoing is true and correct.

Brian H. Whitley ______________________ Executed on 8/28/14
Date

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-14-1108

Enclosure 7

Supplemental Revisions to Licensing Basis Documents
[Associated with the License Amendment Regarding
Passive Core Cooling System (PXS) Condensate Return]

(LAR-13-024S2)

Legend
Underlined blue – new information - example
Double strikethrough red – deleted information – ~~example~~
Green – moved information – ~~example~~, example

Southern Nuclear Operating Company (SNC) personnel have reviewed the NRC requests for additional information (RAIs) and the associated Levy docketed responses and determined that the response information is also pertinent to the NRC review of the Vogtle Electric Generating Plant Units 3 and 4 license amendment request regarding Passive Core Cooling System (PXS) Condensate Return. As such, SNC requests these supplemental revisions to the licensing basis documents be associated with the requested license amendment, LAR-13-024.

The UFSAR Subsection 5.4.14.1, Design Basis, first three paragraphs are revised as shown below.

The passive residual heat removal heat exchanger ~~exchangers~~ automatically actuates to remove~~s~~ core decay heat for an extended ~~unlimited~~ period of time as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank. The passive residual heat removal heat exchanger is designed to withstand the design environment of 2500 psia and 650°F.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in Chapter 15.~~keep the reactor coolant subcooled and prevent water relief from the pressurizer.~~

~~The passive residual heat removal heat exchanger in conjunction with the passive containment cooling system can remove heat for an indefinite time in a closed-loop (that is, no pipe break) mode of operation.~~ In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered ~~depressurized~~ to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

The UFSAR Subsection 6.3.1.1.1, Emergency Core Decay Heat Removal, the first three bullets in the first paragraph are revised as shown below. Note that this includes the proposed changes to this subsection included in the original LAR.

- The passive residual heat removal heat exchanger automatically actuates to provide reactor coolant system cooling~~ and to prevent water relief through the pressurizer safety valves~~.

- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event.~~ for an indefinite time in a closed-loop mode of operation.~~ Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to Subsection 6.3.1.1.4).

- The passive residual heat removal heat exchanger is designed to maintain acceptable ~~cool the~~ reactor coolant system conditions for at least 72~~ to 420°F in 36~~ hours following a non-LOCA event. ~~, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.~~ The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in Subsection 6.3.1.1.4.

- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions~~automatically removing core decay heat following such an event~~, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

- ~~The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation. The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420°F in 36 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.~~

The UFSAR Subsection 6.3.1.1.4, Safe Shutdown, is revised as shown below.

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~~For these events~~, the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the ~~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 operating~~.

The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level.~~as the temperature decreases and pressurizer level decreases, actuating the core makeup tanks.~~ The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

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. ~~However I~~In scenarios when ac power sources are unavailable for ~~as long as~~ approximately 22 ~~24~~ 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, stable shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

For loss of coolant accidents, ~~and other postulated events where ac power sources are lost, or~~ when the core makeup tank level~~s~~ reaches the automatic depressurization system actuation setpoint and other postulated events where ac power sources are lost but passive residual heat removal heat exchanger operation is not extended or is exhausted, the automatic depressurization system will be initiated~~s~~. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once

the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

The basis used to define the passive core cooling system functional requirements is ~~are~~ derived from Section 7.4 of the Standard Review Plan. The functional requirements are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in Subsection 5.4.7 and Section 7.4. The performance of the passive residual heat removal heat exchanger to bring the plant to 420°F in 36 hours is summarized in Subsection 19E.4.10.2.

The UFSAR is revised to include a new Subsection 6.3.1.2, Nonsafety Design Basis, as shown below. This new subsection will also be reflected in the Table of Contents.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Long-Term 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.

6.3.1.3~~2~~ Power Generation Design Basis

The UFSAR is revised to reflect the revised numbering for Subsection 6.3.1.3 in Subsection 6.3.1.1.6, Reliability Requirements, as shown below.

Subsection 6.3.1.3~~2~~ includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

The UFSAR Subsection 6.3.2.1.1, Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions, is revised, beginning at the seventh paragraph, as shown below. Note that this includes proposed changes to this subsection markups included in the original LAR.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and the passive containment cooling system, can provide core cooling for at least 72 hours. ~~an indefinite period of time.~~ After the in-containment refueling water storage tank water reaches its saturation temperature (in ~~about 2~~ several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the ~~The~~ condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located ~~at~~ near the operating deck ~~level which returns the~~ elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter and downspouts normally drain~~s~~ to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for an ~~indefinite~~ extended period of time.

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system ~~safe shutdown~~ condition. It transfers ~~removes~~ decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Offsite or onsite ac power sources are typically recovered within a day, which would allow the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If ac power is not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

The UFSAR Subsection 6.3.2.8, Manual Actions, final paragraph is revised as shown below along with a new paragraph preceding the final paragraph.

The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation. Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

The UFSAR Subsection 6.3.3, Performance Evaluation, seventh paragraph is revised as shown below along with a new paragraph following the seventh paragraph.

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 ~~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.

The UFSAR Subsection 6.3.3, Performance Evaluation, ninth paragraph is revised as shown below along with a new paragraph following the ninth paragraph.

As the core makeup tanks drain down, the automatic depressurization system valves are sequentially actuated. The depressurization sequence establishes reactor coolant pressure conditions that allow injection from the accumulators, and then from the in-containment refueling water storage tank and the containment recirculation path. Therefore, an injection source is continually available. If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in Subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in Subsection 6.3.1.1.4, the passive systems are capable of bring the plant to a safe shutdown condition and maintaining that condition.

The UFSAR is revised to include a new Subsection 6.3.3.2.1.1, Loss of AC Power to the Plant Auxiliaries, as shown below.

6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-loss of coolant accident event lasting as long as 24 hours, the automatic depressurization system will actuate if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat removal heat exchanger operation as described in Subsection 7.4.1.1, such that the automatic depressurization system does not actuate.

The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in Subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of

the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

The UFSAR Subsection 6.3.3.4.1, Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heatups, final sentence in the final paragraph is revised as shown below.

This allows it to ~~indefinitely~~ function as a heat sink.

The UFSAR Subsection 7.4.1.1, Safe Shutdown Using Safety-Related Systems, sixth paragraph is revised as shown below.

The engineered safety system actuation signal generated on low pressurizer pressure also actuates containment isolation. This prevents loss of water inventory from containment and permits extended ~~indefinite~~ operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

The UFSAR Subsection 7.4.1.1, Safe Shutdown Using Safety-Related Systems, eighth and ninth paragraphs are revised as shown below.

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 ~~indefinitely~~ provides core decay heat removal in this configuration with~~out~~ a limited ~~significant~~ increase in the containment water level.

Once the reactor coolant system and the safety systems are in this configuration, the plant is in a stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding,

conservative analysis of extended operation in this mode shows t~~T~~he passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

The UFSAR Subsection 7.4.1.1, Safe Shutdown Using Safety-Related Systems, eleventh paragraph is revised as shown below.

The Class 1E dc batteries that power the automatic depressurization system valves provide power for at least 24 hours. There is a timer that measures the time that ac power sources are unavailable. This timer provides for automatic actuation of the automatic depressurization system before the Class 1E dc batteries are discharged. The emergency response guidelines direct the operator to assess the need for automatic depressurization before the timer completes its count (approximately 22 hours). The operator assessment includes consideration for a visible refueling water storage tank level, full core makeup tanks, ~~and~~ a high and stable ~~in-containment refueling water storage tank~~ pressurizer level, and decreasing or stable reactor coolant system temperature. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

The UFSAR Subsection 9.5.1, Table 9.5.1-1, AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1, item 73, Remarks column, is revised as shown below.

73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b(1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for at least 14 days ~~indefinitely~~. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.

The UFSAR Subsection 15.0.13, Operator Actions, first sentence is revised to clarify the extended decay heat removal capability as shown below.

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a ~~the~~ safe, stable ~~shutdown~~ condition.

The UFSAR Subsection 15.2.6.1, Identification of Causes and Accident Description, fourth paragraph, seventh sentence, is revised as shown below.

The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains ~~keeps the~~ reactor coolant system conditions to satisfy the evaluation criteria~~subcooled indefinitely~~.

The UFSAR Subsection 19.59, Table 19.59-18, AP1000 PRA-Based Insights, item 1.e, Insight column (on Sheet 6 of 25), is revised to clarify the extended decay heat removal capability as shown below.

… The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for at least 72 hours~~an indefinite period of time~~. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates…	6.3.2.1.1 & 6.3.7.6

…

The UFSAR Subsection 19E.4.10.2, Shutdown Temperature Evaluation, is revised as shown below. Note that this includes proposed changes to this subsection markups included in the original LAR.

~~In SECY-94-084, Item C, Safe Shutdown (Reference 14), the NRC staff recommended the Commission's approval of 420°F or below, rather than cold shutdown condition as a safe stable condition, which the PRHR HX must be capable of achieving and maintaining following non-LOCA events, predicated on acceptable passive safety system performance and an acceptable resolution of the regulatory treatment of nonsafety systems (RTNSS) issue. The NRC requested a safety~~ 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 ~~to~~ demonstrates that the passive systems can bring the plant to a safe, stable ~~safe~~ 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.

As discussed in Subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the specified safe shutdown condition following a~~n~~ non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to this ~~a stable safe~~ condition following postulated transients. ~~The results of this~~ A non-bounding, conservative analysis ~~are~~ is represented in Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in

Table 19E.4.10-1. Though some of the assumptions in this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in Subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to ensure the results were conservative for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The efficiency of the gutter collection system was determined separate from the WGOTHIC analysis. The resulting time-dependent condensate return rate was incorporated into the LOFTRAN computer code described in Subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip. The PRHR HX ~~heat exchanger~~ is actuated on the low steam generator narrow range level coincident with low startup feed water flow rate signal. Eventually a safeguards actuation signal is actuated on ~~Low~~ low cold leg temperature and the CMTs are actuated.

Once actuated, at about ~~600~~ 2,400 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water~~flow rate~~, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about ~~3,500~~ 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about ~~31,000~~ 34,500 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F within 48,600 ~~at 82,600~~ seconds, while the core average temperature reaches 420°F within 124,400 ~~123,600~~ seconds (approximately 34.6 hours).

As discussed in Subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. At approximately 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. ~~this mode of operation can last for up to 72 hours. However, in about 22 hours after the event, if no ac power is available, or if condensate return is not available, then the operator is instructed to actuate the ADS.~~ Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

As discussed in Subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

The UFSAR Subsection 19E.9, References, is revised as shown below.

14. Not used. ~~SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.~~

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-14-1108

Enclosure 8

Revision to LAR-13-024 Enclosure 1 Section 4.3
Significant Hazards Consideration Determination

(LAR-13-024S2)

Legend
Underlined blue – new information - example
Double strikethrough red – deleted information – ~~example~~
Green – moved information – ~~example~~, example

Southern Nuclear Operating Company (SNC) personnel have reviewed the NRC requests for additional information (RAIs) and the associated Levy docketed responses and determined that the response information is also pertinent to the NRC review of the Vogtle Electric Generating Plant Units 3 and 4 license amendment request regarding Passive Core Cooling System (PXS) Condensate Return. Further, SNC has determined that the responses impact the Significant Hazards Consideration Determination provided in the original LAR. As such, a revised Significant Hazards Consideration Determination is provided below.

4.3 Significant Hazards Consideration Determination

The proposed changes would revise the Combined Licenses (COLs) in regard to ~~modifications to~~ the addition of downspouts to capture condensate at the polar crane girder (PCG) and internal stiffener locations and return it to the IRWST.

The requested amendment requires changes to Updated Final Safety Analysis Report (UFSAR) Tier 2 information, which involve changes to plant-specific Tier 1 and corresponding changes to COL Appendix C information.

An evaluation to determine whether or not a significant hazards consideration is involved with the proposed amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below:

4.3.1 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed containment condensate flow path changes provide sufficient condensate return flow to maintain In-containment Refueling Water Storage Tank (IRWST) level above the top of the Passive Residual Heat Removal Heat Exchanger (PRHR HX) tubes~~, thus~~ long enough to prevent~~ing~~ PRHR HX performance degradation from that considered in the UFSAR Chapter 15 safety analyses. The added components are seismically qualified and constructed of only those materials appropriately suited for exposure to the reactor coolant environment as described in UFSAR Section 6.1. No aluminum is permitted to be used in the construction of these components so that they do not contribute to hydrogen production in containment.

The proposed changes clarify the design basis for the PRHR HX, which removes decay heat from the Reactor Coolant System (RCS) during a non-loss of coolant accident (non-LOCA). With operator action to avoid unnecessary Automatic Depressurization System (ADS) actuation based on RCS conditions, PRHR HX operation can be extended longer than would be maintained automatically by the protection system. Though analysis shows significantly greater capacity, the extent of the capability of the PRHR HX would be changed from operating indefinitely to operating for at least 72 hours. If PRHR HX capability were

exhausted after 72 hours, the ADS would be actuated, which could result in significant containment floodup. However, probabilistic analysis shows the probability of design basis containment floodup after PRHR HX operation during a non-LOCA event is significantly lower than the probability of a small break LOCA, for which comparable containment floodup is anticipated. Therefore, the probability of significant containment floodup is not increased.

~~The proposed changes do not alter design features available during anticipated operational occurrences or accidents.~~ The proposed changes do not ~~involve~~ affect any ~~accident initiating~~ components~~/system~~ whose failure could initiate a previously evaluated ~~or~~ event, thus the probabilities of the accidents previously evaluated are not affected. The affected equipment does not adversely affect or interact with safety-related equipment or another radioactive material barrier~~, and this activity does not involve the containment of radioactive material~~. Th~~us, th~~e proposed changes clarify the post-accident performance requirements for the PRHR HX. However, the proposed changes do not prevent the engineered safety features from performing their ~~affect any~~ safety-related accident mitigating functions. The radioactive material source terms and release paths used in the safety analyses are unchanged, thus the radiological releases in the UFSAR accident analyses are not affected.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4.3.2 Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The long-term safe shutdown analysis results show that the PRHR HX continues to meet~~s~~ its acceptance criterion, i.e., to cool the Reactor Coolant System (RCS) to below 420ºF in 36 hours. The ~~affected~~ added equipment does not adversely interface with any component whose failure could initiate an accident, or any component that contains radioactive material. The modified components do not incorporate any active features relied upon to support normal operation. The downspout and gutter return components are seismically qualified to remain in place and functional during seismic and dynamic events. The containment condensate flow path changes do not create a new fault or sequence of events that could result in a radioactive material release.

The proposed change quantifies the duration that the PRHR HX is capable of maintaining adequate core cooling, and specifies that if PRHR HX cooling capability is exhausted, the ADS would be actuated. This involves the possibility of opening the ADS valves after the IRWST water level has decreased below the spargers, which promote steam condensation in the IRWST. During this condition, the loads on the IRWST, spargers and any internal structures or components in the IRWST would still be less than their limiting loads, and these

SSCs would not be adversely affected or cause a different mode of operation. Therefore, no new type of accident could be created by this condition.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident.

4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed changes do not reduce the redundancy or~~,~~ diversity ~~or performance~~ of any safety-related function. The added components are classified as safety-related, seismically qualified, and are designed to comply with applicable design codes. The proposed containment condensate flow path changes provide sufficient condensate return flow to maintain adequate IRWST water level for those events using the PRHR HX cooling function. The long-term Shutdown Temperature Evaluation results in UFSAR Chapter 19E show the PRHR HX continues to meet its acceptance criterion. The UFSAR Chapters 6 and 15 analyses results are not affected, thus margins to their regulatory acceptance criteria are unchanged. ~~The added components are classified as safety-related, seismically qualified, and are designed to comply with applicable design codes.~~ The former design basis, which stated the PRHR HX could bring the plant to 420°F within 36 hours, is changed to state the heat exchanger can establish safe, stable conditions in the reactor coolant system after a design basis event. Such safe stable conditions may not coincide with an RCS temperature of 420°F. However, the PRHR HX is able to bring the RCS to a sufficiently low temperature such that RCS conditions would be comparable to those achieved at 420°F – peak cladding temperatures, departure from nucleate boiling, and pressurizer level would be maintained within acceptable limits of the evaluation criteria. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes, thus no margin of safety is significantly reduced.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-14-1108

Enclosure 9

Revision to LAR-13-024 Enclosure 2
Request for Exemption Regarding
Passive Core Cooling System (PXS) Condensate Return

Legend
Underlined blue – new information - example
Double strikethrough red – deleted information – ~~example~~
Green – moved information – ~~example~~, example

Southern Nuclear Operating Company (SNC) personnel have reviewed the NRC requests for additional information (RAIs) and the associated Levy docketed responses and determined that the response information is also pertinent to the NRC review of the Vogtle Electric Generating Plant Units 3 and 4 license amendment request regarding Passive Core Cooling System (PXS) Condensate Return. Further, SNC has determined that the request for exemption discussion provided with the original LAR also required some minor revisions to reflect the response information. As such, a revised request for exemption is provided below.

1.0 Purpose

Southern Nuclear Operating Company (the Licensee) requests a permanent exemption from the provisions of 10 CFR 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," to allow a departure from elements of the certification information in Tier 1 of the Generic DCD. The regulation, 10 CFR 52, Appendix D, Section III.B, requires an applicant or licensee referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certification information in DCD Tier 1. The Tier 1 information for which a plant-specific departure and exemption is being requested includes changes to improve the condensate return for the Passive Core Cooling System (PXS).

This request for exemption will apply the requirements of 10 CFR 52, Appendix D, Section VIII.A.4 to allow departures from generic Tier 1 information due to the following proposed additions to the system-based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the Passive Core Cooling System as identified in Table 2.2.3-1 and Table 2.2.3-2.

The added components of the PXS are integral to providing safety-related core decay heat removal during non-LOCA events. Therefore, it is appropriate to apply inspections, test, analyses and acceptance criteria to the added PXS components to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards.

The downspout screens support the capability of the passive residual heat removal heat exchanger (PRHR HX) to maintain the reactor in a safe shutdown condition by preventing large objects from entering the downspout piping. As required by General Design Criterion 2 of Appendix A to 10 CFR Part 50, the PXS is designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions. The PXS downspout screens are safety-related; located on the Nuclear Island; and required to withstand design basis seismic and post-accident operating loads without losing the capability to perform their safety function. To provide assurance these ITAAC design commitments will be met, plant-specific Tier 1 Table 2.2.3-1 is updated to include eight new downspout screens.

The downspout piping supports the capability of the PRHR HX to maintain the reactor in a safe shutdown condition by inhibiting containment flood-up during PRHR HX operation and delaying

the need for containment recirculation following RCS depressurization. As required by General Design Criterion 4 of Appendix A to 10 CFR Part 50, the PXS containment downspout piping is safety-related and required to withstand normal and seismic design basis loads without losing functional capability. To provide assurance these ITAAC design commitments will be met, plant-specific Tier 1 Table 2.2.3-2 is updated to include the new PXS pipe lines.

2.0 Background

The Licensee is the holder of Combined License Nos. NPF-91 and NPF-92, which authorize construction and operation of two Westinghouse Electric Company AP1000 nuclear plants, named Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

The Updated Final Safety Analysis Report (UFSAR), Subsection 6.3.1.1.1, "Emergency Core Decay Heat Removal," identifies the safety-related design bases of the Passive Core Cooling System (PXS) includes the capability for the Passive Residual Heat Removal Heat Exchanger (PRHR HX) to cool the Reactor Coolant System (RCS) to the safe shutdown condition of 420°F in 36 hours. The Nuclear Regulatory Commission Staff recommended, in SECY-94-084 that reactor designs utilizing passive safety systems include a residual heat removal system capable of bringing the reactor to a safe shutdown condition of 420°F or lower following non-loss of coolant accident (LOCA) events. To support the capability of the AP1000 design to meet this design criterion, a safe shutdown temperature evaluation was performed, which assumed a specific condensate return fraction for the PXS.

Through a series of design reviews, the efficiency of the condensate return to the In-Containment Refueling Water Storage Tank (IRWST) was further evaluated. Testing results showed that the current design could have an efficiency for condensate return lower than initially assumed. These evaluations were initiated to investigate and better quantify the returned fraction of condensate to the IRWST. Supplementary testing revealed opportunities to improve the design with regard to the condensate return fraction used to evaluate long- term plant cooldown. In addition, an analysis methodology was applied to characterize both the thermodynamic and the geometric phenomena involved in prolonged non-LOCA events.

3.0 Technical Justification of Acceptability

General design criteria 34 and 35 require the PXS to be capable of removing core decay and residual heat, and provide an abundance of core cooling such that fuel design limits and the RCS design conditions are not exceeded. As the PXS provides core decay heat removal during design basis events, performance of this safety-related function is confirmed through ITAAC 2.2.3, design commitment 8.b. The changes described herein do not change the commitment to complete the performance test of the PRHR HX.

Additional detail for justification for this exemption is provided in Section 2 of the accompanying License Amendment Request in Enclosure 1, and is based on information provided in Westinghouse report APP-GW-GLR-161, Revision 1, which was addressed and provided on this docket in Supplement 1 to LAR-13-024 (ND-14-0624 submitted April 18, 2014).

4.0 Justification of Exemption

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. Because the Licensee has identified changes to the Tier 1 information related to the Tier 2 departure discussed in Enclosure 1 of the accompanying License Amendment Request, an exemption from the certified design information in Tier 1 is needed. These material departures are contained in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, and involve the addition of components to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions.

10 CFR Part 52, Appendix D, and 10 CFR §§ 50.12, 52.7, and 52.63 state that the NRC may grant exemptions from the requirements of the regulations provided six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, App. D, VIII.A.4].

The requested exemption satisfies the criteria for granting specific exemptions, as described below.

1. **This exemption is authorized by law**

   The NRC has authority under 10 CFR §§ 50.12, 52.7, and 52.63 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR §§50.12 and 52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

   Accordingly, this requested exemption is "authorized by law," as required by 10 CFR §50.12(a)(1).

2. **This exemption will not present an undue risk to the health and safety of the public**

   The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific Tier 1 DCD will continue to reflect the approved licensing basis for the Licensee, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. Because the change to the condensate return portion of the passive core

cooling system description maintains its design functions, the changed design continues to provide the protection of the health and safety of the public. Therefore, no adverse safety impact that would present any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD will also continue to provide the detail necessary to support the performance of the associated ITAAC.

Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

3. The exemption is consistent with the common defense and security

The exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would change elements of the plant-specific Tier 1 DCD by departing from the AP1000 certified (Tier 1) design information. The exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

Therefore, the requested exemption is consistent with the common defense and security.

4. Special circumstances are present

10 CFR 50.12(a)(2) list six "special circumstances" for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The rule under consideration in this request for exemption from Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, is 10 CFR 52, Appendix D, Section III.B, which requires that a licensee referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The VEGP Units 3 and 4 COLs reference the AP1000 Design Certification Rule and incorporate by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D.

The proposed changes to the condensate return portion of the passive core cooling system maintain the design margins of the Passive Core Cooling System. This change does not impact the ability of any structures, systems, or components to perform their

functions or negatively impact safety. Accordingly, this exemption from the certification information will enable the applicant to safely construct and operate the AP1 000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

5. **The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption**

   Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the Passive Core Cooling System, it is expected that other AP1000 applicants and licensees will also request this exemption. This exemption request and the associated marked-up tables demonstrate that there is a minimal change from the generic AP1000 DCD, minimizing the reduction in standardization and consequently the safety impact from the reduction.

   Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. In fact, as described in item 6 below, the exemption will result in no reduction in the level of safety.

6. **The design change will not result in a significant decrease in the level of safety.**

   The exemption revises the plant-specific DCD Tier 1 information by adding ~~altering the description of the passive core cooling system condensate return design. The~~ components ~~added~~ to the condensate return function design to enable the passive core cooling system to ~~meet~~ more effectively perform its design functions. Because these functions continue to be met, there is no reduction in the level of safety.

5.0 Risk Assessment

A risk assessment was determined to be not applicable to address the acceptability of this request.

6.0 Precedent

No precedent for this request is identified.

7.0 Environmental Consideration

A review has determined that the proposed exemption would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed

exemption does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Specific justification is provided in Section 5 of the corresponding License Amendment Request in Enclosure 1. Accordingly, the proposed exemption meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed exemption.

8.0 Conclusion

The proposed changes to Tier 1 are necessary to revise the passive core cooling system design description in the plant-specific DCD Tier 1 information. The exemption request meets the requirements of 10 CFR 52.63, *"Finality of design certifications,"* 10 CFR 52.7, *"Specific exemptions,"* 10 CFR 50.12, *"Specific exemptions,"* 10 CFR 51.22, and 10 CFR 52 Appendix D, *"Design Certification Rule for the AP1000."* Specifically, the exemption request meets the criteria of 10 CFR 50.12(a)(1) in that the request is authorized by law, presents no undue risk to public health and safety, and is consistent with the common defense and security. Furthermore, approval of this request does not result in a significant decrease in the level of safety, presents special circumstances, does not present a significant decrease in safety as a result of a reduction in standardization, and meets the eligibility requirements for categorical exclusion.