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Serial: NPD-NRC-2014-035
September 3, 2014

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

LEVY NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 52-029 AND 52-030
ROADMAP OF CHANGES IN COMBINED LICENSE APPLICATION, REVISION 7

Reference: Letter from Christopher M. Fallon (DEF) to U.S. Nuclear Regulatory Commission, dated August 28, 2014, "Levy Nuclear Plant Units 1 and 2 Submittal of COL Application, Revision 7", Serial: NPD-NRC-2014-034

Ladies and Gentlemen:

The purpose of this letter is to provide information supporting the recent Duke Energy revision of the Combined License Application (COLA) for Levy Nuclear Plant, Units 1 and 2 (see referenced letter). Attached is a "roadmap" of the changes included in the August 28, 2014 submittal along with an enclosure providing an explanation of the information contained in the roadmap.

If you have any questions, or need additional information, please contact me at (704) 382-4046.

Sincerely,

Robert Kitchen
Director – Nuclear Licensing
Nuclear Development

Enclosure/Attachment
Enclosure: Levy Nuclear Plant Units 1 and 2 Roadmap of Changes in Combined License Application Revision 7 Explanation by Column in Attachment 1
Attachment: Attachment 1 – LNP COLA Revision 7 Roadmap of Changes

cc : U.S. NRC Region II, Regional Administrator
Mr. Don Habib, U.S. NRC Project Manager
Ms. Mallecia Sutton, U. S. NRC Environmental Project Manager (w/o enclosure/attachment)

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Levy Nuclear Plant Units 1 and 2
Roadmap of Changes in Combined License Application Revision 7
Explanation by Column in Attachment 1

Column	Explanation
Change ID#	Unique identifier for tracking purposes
COLA	Identifies the change as STD (standard) or LNP specific
COLA Part	Part 1 (PT01) through Part 11 (PT11)
Chapter	FSAR or ER Chapter
Section	Section/Subsection of the Chapter or Part
Basis for Change	The source of the change
Change Summary	Short description of the change

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
Part 1						
LNP-001	LNP	1		01.01.04	Revisions to list of directors and officers for DUK and subsidiary companies	Changes include new or different officers and position titles, and replacement of retiring directors with new directors
Part 2						
LNP-002	LNP	2	1	01.04.01	NPD-NRC-2014-019	Delete the fourth paragraph added to the beginning of DCD Subsection 1.4.1 that deals with the EPC contract signed on December 31, 2008.
LNP-003	LNP	2	1	01.04.01	NPD-NRC-2014-019	Replace the second and third paragraphs added to the end of DCD Subsection 1.4.1 with the following paragraph: Not all participants have been identified at this time. Additional participants may be required. Changes to this subsection are required to identify additional participants, principal consultants, outside service organizations, or contractors for the design, construction, and operation of LNP. Changes are also required to delineate the division of responsibility among the certified plant designer, architect-engineer, constructor, and plant operator as appropriate.
LNP-004	LNP	2	1	01.01.T/Table 1.1-201	NPD-NRC-2014-019	Delete the acronym "CB&I" from Table 1.1-201
LNP-005	LNP	2	1	01.08.T/T1.8-201	Enclosure 5 of NPD-NRC-2014-028	COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add additional FSAR changes to the list of FSAR Section or Subsection references for departure LNP COL 3.2-1, to read as follows: Departure Description Summary: The condensate return portion of the Passive Core Cooling System has been upgraded to add downspouts and plug fabrication holes in the Polar Crane Girder in order to maximize the return of condensate to the In-Containment Refueling Water Storage Tank and ensure long-term operation of the Passive Residual Heat Removal Heat Exchanger to meet design requirements. The following are the departures from the DCD: Tier 1 Table 2.2.3-1 and Table 2.2.3-2, Tier 2 Table 3.2-3 (Sheet 16 of 75), Figure 3.8.2-1 (Sheet 3), Subsections 5.4.11.2 and 5.4.14.1, Chapter 6 TOC (Table of Contents, List of Figures), Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, 6.3.3.2.1.1, Figure 6.3-1 (Sheets 1 through 3), Figure 6.3-2 (Not Used), Subsection 7.4.1.1, Table 14.3-2 (Sheets 7 and 8 of 17), Subsection 15.0.13, Chapter 16 (TS Bases B3.3.3 and B3.5.4), Subsection 19E.4.10.2, Table 19E.4.10-1, Figures 19E.4.10-1 through 19E.4.10-4, and 19E.9. FSAR Section or Subsection: Table 3.2-202, Figure 3.8-201, 5.4.11.2, 5.4.14.1, 6 TOC (List of Figures), 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, 6.3.3.2.1.1, Figure 6.3-201, 7.4.1.1, 14 TOC (List of Tables), Table 14.3-202, 15.0.13, 16 (TS Bases B3.3.3 and B3.5.4), 19 TOC (List of Tables and List of Figures), 19E.4.10.2, Table 19E.4.10-201, Figures 19E.4.10-201 through 19E.4.10-204, 19E.9
LNP-006	LNP	2	1	01.08.T/T1.8-201	NPD-NRC-2013-037	Insert the following text as the fifth line item in FSAR Table 1.8-201, Summary of FSAR Departures from the DCD: LNP DEP 3.7-1 Departure to address use of site-specific horizontal seismic response spectra for the design of drilled shafts that support the seismic Category II portions of the Annex and Turbine Buildings. FSAR Section or Subsection References are: 3.7.2.8.1, 3.7.2.8.3
LNP-062	LNP	2	1	01.08.T/T1.8-201	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add FSAR changes to the list of FSAR Section or Subsection references for departure LNP COL 6.3-1, to read as follows: Departure Description Summary: The DCD states that the PRHR HX can maintain safe shutdown conditions for non-LOCA accidents "indefinitely." A quantitative duration of greater than 14 days has been adopted based on that time being long enough to minimize the need to switch to passive feed and bleed cooling except for very unlikely or extreme hazard events. The following are the departures from the DCD: Subsection 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, Subsection 7.4.1.1, Table 9.5.1-1 (Sheet 11), Subsection 15.2.6.1, Table 19.59-18 (Sheet 6), Subsection 19E.4.10.2 FSAR Section or Subsection: 5.4.14.1, 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, 7.4.1.1, Table 9.5.1-201, 15.2.6.1, Table 19.59-202, 19E.4.10.2
LNP-007	LNP	2	2	Table of Contents Chapter 2	Errata	Add "Chapter 2 – Site Characteristics" to the header of the Chapter 2 TOC
LNP-008	LNP	2	2	02.04.16	Errata	Change reference 2.4.3-212 to "Not Used"
LNP-101	LNP	2	2	Table of Contents Chapter 2 and 2.1	Errata	Formatting updates to the TOC; removed extra blank lines in Section 2.1.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-102	LNP	2	2	02.05.07	Errata	Change references 2.5.1-325, 2.5.3-227 and 2.5.4.8-202 to "Not used."
LNP-009	LNP	2	3	03.07.02.08.01	NPD-NRC-2013-037	Insert the following text as the first paragraph in FSAR Subsection 3.7.2.8.1, with a LMA of LNP DEP 3.7-1: The drilled shaft foundations supporting the buildings adjacent to the nuclear island do not conform to one of the six soil types supporting these buildings that were analyzed in the AP1000 generic analysis. In the conceptual design of the drilled shafts supporting the seismic Category II portions of the Annex Building, the vertical seismic demands are consistent with the AP1000 certified design demands. The AP1000 certified vertical seismic demands exceed the site- specific vertical seismic demands at the LNP site. The PBSRS, rather than the AP1000 CSDRS based on the envelope of the six analyzed soil profiles, is used to compute the maximum relative horiizontal displacement of the Annex Building drilled shaft foundations with respect to the Nuclear Island to evaluate the site-specific aspect of the seismic interaction of the seismic Category II portions of the Annex Building with the Nuclear Island. Thus, the drilled shafts are designed for the AP1000 certified design vertical seismic loads and the site-specific horizontal seismic loads and ensure that the maximum relative displacement of the foundation of this building and the Nuclear Island remains within the limits of the AP1000 generic design.
LNP-010	LNP	2	3	03.07.02.08.03	NPD-NRC-2013-037	Insert the following text as the first paragraph in FSAR Subsection 3.7.2.8.3, with a LMA of LNP DEP 3.7-1: The drilled shaft foundations supporting the buildings adjacent to the nuclear island do not conform to one of the six soil types supporting these buildings that were analyzed in the AP1000 generic analysis. In the conceptual design of the drilled shafts supporting the seismic Category II portion of the Turbine Building, the vertical seismic demands are consistent with the AP1000 certified design demands. The AP1000 certified vertical seismic demands exceed the site- specific vertical seismic demands at the LNP site. The PBSRS, rather than the AP1000 CSDRS based on the envelope of the six analyzed soil profiles, is used to compute the maximum relative horizontal displacement of the Turbine Building drilled shaft foundations with respect to the Nuclear Island to evaluate the site-specific aspect of the seismic interaction of the seismic Category II portion of the Turbine Building with the Nuclear Island. Thus, the drilled shafts are designed for the AP1000 certified design vertical seismic loads and the site-specific horizontal seismic loads and ensure that the maximum relative displacement of the foundation of this building and the Nuclear Island remains within the limits of the AP1000 generic design.
LNP-061	LNP	2	3	03.07.06	Errata	References 3.7.6-201, 3.7.6-202 and 3.7.6-203 were added in Revision 2 of the FSAR but were never referenced in the text. These three references are changed to read "Not Used"
LNP-011	LNP	2	5	05.04.11.02	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.11.2, with a LMA of LNP DEP 3.2-1, to read: 5.4.11.2 System Description Replace the second sentence of the second paragraph of DCD Subsection 5.4.11.2 with the following: The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure 6.3-1.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-063	LNP	2	5	05.04.14.01	Enclosure 5 of NPD-NRC-2014-021	2. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.14.1 to read: 5.4.14.1 Design Bases Replace the first sentence of the first paragraph of DCD Subsection 5.4.14.1 with the following,with a LMA of LNP DEP 6.3-1: The passive residual heat removal heat exchanger automatically actuates to remove core decay heat for an extended 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. Combine the second and third paragraphs of DCD Subsection 5.4.14.1 and revise to read as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1: 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. 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 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.
LNP-012	LNP	2	6	06.03.01.01.01	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1 1.1, with a LMA of LNP DEP 3.2-1, to read: 6 3.1.1.1 Emergency Core Decay Heat Removal Add new second and third bullets in the first paragraph of DCD Subsection 6.3.1.1.1 to read as follows: • 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. 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 reactor coolant system conditions for at least 72 hours following a non-LOCA event. 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. Replace the fourth bullet (old second bullet) in the first paragraph of DCD Subsection 6.3.1.1.1 with the following: • The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, 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.
LNP-064	LNP	2	6	06.03.01.01.01	Enclosure 5 of NPD-NRC-2014-021	Revise the first paragraph of DCD Subsection 6.3.1.1.1 by deleting entirely the fifth bullet (old third bullet). Show as "(Deleted - new fifth bullet (old third bullet))"with LMAs of LNP DEP 3.2-1 and 6.3-1.
						COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.4,with a LMA of LNP DEP 3.2-1, to read: 6.3.1.1.4 Safe Shutdown Replace the first two paragraphs of DCD Subsection 6.3.1.1.4 with the following three paragraphs, to read: 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.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-065	LNP	2	6	06.03.01.01.04	Enclosure 5 of NPD-NRC-2014-021: Superseded by Enclosure 5 of NPD-NRC-2014-028	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. 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. 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 maintain safe, stable shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time. Replace the first sentence of the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4 with the following: The basis used to define the passive core cooling system functional requirements is derived from Section 7.4 of the Standard Review Plan. Add a last sentence to the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4, to read as follows: 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.
LNP-081	LNP	2	6	06.03.01.01.06	Enclosure 5 of NPD-NRC-2014-028	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.6, with a LMA of LNP DEP 3.2-1, to read: Replace the last sentence of DCD Subsection 6.3.1.1.6 with the following: Subsection 6.3.1.3 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.
LNP-066	LNP	2	6	06.03.01.02	Enclosure 5 of NPD-NRC-2014-021: Superseded by Enclosure 5 of NPD-NRC-2014-028	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.2(new DCD Subsection 6.3.1.2), with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1, to read: 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.
LNP-067	LNP	2	6	06.03.01.03	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.3, title only, to reflect the numbering change of DCD Subsection 6.3.1.2 to 6.3.1.3, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1, to read as follows: 6.3.1.3 Power Generation Design Basis
LNP-013	LNP	2	6	06.03.02.01	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1, with a LMA of LNP DEP 3.2-1, to read: 6.3.2.1 Schematic Piping and Instrumentation Diagrams Replace the first sentence of the first paragraph of DCD Subsection 6.3.2.1 with the following: Figure 6 3-1 shows the piping and instrumentation drawings of the passive core cooling system.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
						COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1.1, with a LMA of LNP DEP 3.2-1, to read: 6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions Replace the seventh and eighth paragraphs of DCD Subsection 6.3.2.1.1 with the following, with LMAs of LNP DEP 3.2-1 and LNP DEP 6 3-1: 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. After the in-containment refueling water storage tank water reaches its saturation temperature (in 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.
LNP-014	LNP	2	6	06.03.02.01.01	Enclosure 5 of NPD-NRC-2014-021	Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck 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 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 extended period of time. Revise the first and second sentences of the ninth paragraph of DCD Subsection 6.3.2.1.1 to read as follows, with a LMA of LNP DEP 3.2-1: The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers 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. Add a new tenth paragraph to DCD Subsection 6.3.2.1.1 to read as follows, with a LMA of LNP DEP 3.2-1: 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. Off-site or on-site 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.
LNP-015	LNP	2	6	06.03.02.02.07	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7, with a LMA of LNP DEP 3.2-1, to read: 6.3.2.2.7 IRWST and Containment Recirculation Screens Replace the first paragraph of DCD Subsection 6.3.2.2.7 with the following: The passive core cooling system has two different sets of screens that are used to prevent debris from entering the reactor and blocking core cooling passages during a LOCA: IRWST screens and containment recirculation screens. The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural frames attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including: • GDC 35 of 10 CFR 50 Appendix A • Regulatory Guide 1.82 • NUREG-0897
LNP-016	LNP	2	6	06.03.02.02.07.01	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.1, with a LMA of LNP DEP 3.2-1, to read: 6.3.2.2.7.1 General Screen Design Criteria Replace the first paragraph of DCD Subsection 6.3.2.2.7.1 with the following: The IRWST screens and containment recirculation screens are designed to comply with the following criteria.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-017	LNP	2	6	06.03.02.02.07.02	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.2, with a LMA of LNP DEP 3.2-1, to read: 6.3.2.2.7.2 IRWST Screens Replace the third paragraph of DCD Subsection 6.3.2.2.7 2 with the following. During a LOCA, steam vented from the reactor coolant system condenses on the containment shell and drains down the shell to the polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell and is collected in a gutter near the operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.
LNP-068	LNP	2	6	06.03 02.08	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.8, with a LMA of LNP DEP 3.2-1, to read: 6.3.2.8 Manual Actions Add a new third paragraph of DCD Subsection 6.3.2.8 to read as follows: 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. Add a new first sentence to the fourth paragraph (old third paragraph) of DCD Subsection 6.3.2.8, to read as follows: Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation.
LNP-069	LNP	2	6	06.03.03	Enclosure 5 of NPD-NRC-2014-021: Superseded by Enclosure 5 of NPD-NRC-2014-028	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3,with a LMA of LNP DEP 3.2-1, to read: 6.3.3 Performance Evaluation Replace the seventh paragraph of DCD Subsection 6.3.3 with the following: 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. Add a new eighth paragraph to DCD Subsection 6.3.3, as follows: 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. Add the following as the last sentence to the tenth paragraph (old ninth paragraph) of DCD Subsection 6.3 3, as follows: 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. Add a new eleventh paragraph to DCD Subsection 6.3.3, as follows: 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 bringing the plant to a safe shutdown condition and maintaining that condition.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-070	LNP	2	6	06.03.03.02.01.01	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.2.1.1 (new DCD Subsection 6.3.3.2.1.1),with a LMA of LNP DEP 3.2-1, to read: 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.
LNP-071	LNP	2	6	06.03.03.04.01	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.4.1,with a LMA of LNP DEP 6.3-1, to read: 6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups Revise the last sentence of the fourth paragraph of DCD Subsection 6.3.3.4.1 to read as follows: This allows it to function as a heat sink.
LNP-018	LNP	2	6	06.03.F/Figure 6.3-201, Sheets 1-3	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Sections 6.3 will be revised to add a departure from DCD Figure 6.3-1 as Figure 6.3-201 to revise as shown in Sheets 1 through 3 of Figure 6.3-1 in the attachment to the enclosure of the referenced letter, with a LMA of LNP DEP 3.2-1. These sheets replace the figure added as Figure 6.3-201 in LNP COLA Revision 6.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-072	LNP	2	7	07.04.01.01	Enclosure 5 of NPD-NRC-2014-021: Superseded by Enclosure 5 of NPD-NRC-2014-028	COLA Part 2, FSAR Chapter 7, will be revised to add new Subsection 7.4.1.1, to read: This section of the referenced DCD is incorporated by reference with the following departures and/or supplements. 7.4.1.1 Safe Shutdown Using Safety-Related Systems Revise the second sentence of the sixth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 6.3-1: This prevents loss of water inventory from containment and permits extended operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank. Revise the last sentence of the eighth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1: The system provides core decay heat removal in this configuration with a limited increase in the containment water level. Revise the ninth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 3.2-1: 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 the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours. Revise the last three sentences of the eleventh paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 3.2-1: The operator assessment includes consideration for a visible refueling water storage tank level, full core makeup tanks, a high and stable 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.
						Add the following subsection to FSAR Chapter 8 following Subsection 8.2.1.2.1 with a LMA of LNP SUP 8.2-5: 8.2.1.2.2 Plant Response to High Voltage Open Phase Condition A monitoring system is installed on the credited GDC 17 offsite power circuit that provides continuous open phase condition monitoring of the MSU transformer HV input power supply (see Reference 201). The system detects an open phase condition (with or without a concurrent high impedance ground on the HV side of the transformer) on one or more phases under all transformer loading conditions. The open phase condition monitoring system provides an alarm to the operators in the control room should an open phase condition occur on the HV source to the MSU transformers. The system design utilizes commercially available components including state of the art digital relaying equipment and input parameters as required to provide loss of phase detection and alarm capability. Additionally, a high-voltage open phase condition with or without a ground fault can manifest itself as an unacceptable voltage on the 6.9 kV medium voltage ES-1 and ES-2 buses during normal loading conditions. The presence of unacceptable voltages on the ES-1 and ES-2 buses results in isolation of the affected medium voltage bus from the offsite power supply and enables the onsite standby diesel generators to start and restore AC power to the ES-1 and ES-2 buses and associated defense-in-depth loads. The onsite AC power system is described in DCD Section 8.3.1. Motor management relays for the medium voltage motors on ES-1 and ES-2 provide detection of unacceptably high negative sequence currents. High negative sequence current motor trips or other running load trips provide alarms in the MCR, which can assist in the detection of a high-voltage open phase condition with or without a ground fault. Electric circuit protection for the medium voltage system and equipment is described in DCD Section 8.3.1.1.1.1.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-019	LNP	2	8	08.02.01.02.02	NPD-NRC-2014-002	A high-voltage open phase condition with or without a ground fault can also manifest itself as an unacceptable voltage on the 480 VAC low-voltage buses powered from ES-1 and ES-2. The safety-related IDS battery chargers are powered from the low-voltage buses and continue to charge the IDS batteries unless the battery charger input or output monitored electrical parameters are unacceptable. If the monitored electrical parameters degrade to the point that the battery charger no longer provides sufficient DC bus voltage, the Class 1E electrical system DC bus receives power from the applicable IDS battery and the battery charger maintains isolation between the Non- Class 1E AC and Class 1E DC power systems which generates alarms in the MCR. The onsite AC power system is described in DCD Section 8.3.1 and the Class 1E DC power system is described in DCD Section 8.3.2.1.1. Operator actions and maintenance and testing activities are addressed in procedures, as described in Section 13.5. Plant operating procedures, including off-normal operating procedures associated with the monitoring system will be developed prior to fuel load. Maintenance and testing procedures, including calibration, surveillance testing, setpoint determination and troubleshooting procedures associated with the monitoring system will be developed prior to fuel load. Control Room operator and maintenance technician training associated with the operation and maintenance of the monitoring system will be conducted in accordance with the milestones for Non Licensed Plant Staff and Reactor Operator Training Programs in Table 13.4-201.
LNP-020	LNP	2	8	08.02.06	NPD-NRC-2014-002	Add the following subsection to FSAR Chapter 8: 8.2.6 References Add the following information at the end of DCD Subsection 8.2.6. 201. NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System," July 27, 2012.
LNP-073	LNP	2	9	09.05.T/T9.5.1-201	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Section 9.5 will be revised to add a departure from DCD Table 9.5.1-1, AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1, Sheet 11 of 33, as new FSAR Table 9.5.1-201, with a LMA of LNP DEP 6.3-1. This Table shall also be added to the list of tables in Chapter 9.
LNP-021	LNP	2	11	11.04.06	NPD-NRC-2013-043	Delete the two paragraphs shown below with a LMA of LNP COL 11.4-1 from COLA Part 2, FSAR Chapter 11, Subsection 11.4.6 at the end of STD COL 11.4-1: All packaged and stored radwaste will be shipped to offsite disposal/storage facilities and temporary storage of radwaste is only provided until routine offsite shipping can be performed. Accordingly, there is no expected need for permanent on-site storage facilities at LNP 1 & 2. If additional storage capacity for Class B and C waste is required, further temporary storage would be developed in accordance with NUREG-0800, Standard Review Plan 11.4, Appendix 11 4-A. To the extent that additional storage could be needed sometime in the future, the existing regulatory framework would allow Duke Energy to conduct written safety analyses under 10 CFR 50.59. If the additional storage does not satisfy 10 CFR 50.59, a license amendment would be required.
LNP-022	LNP	2	11	11.02.01.02.05.02	NPD-NRC-2013-036 Superseded by NPD-NRC-2013-039	Revise the paragraph with a LMA of LNP COL 11.2-1 to read as follows: When mobile or temporary equipment is selected to process liquid effluents, the equipment design and testing meets the applicable requirements of Regulatory Guide 1.143. When confirmed through sampling that the radioactive waste contents result in an inventory on a mobile system that is below the A2 quantity limit for radionuclides specified in Appendix A to 10 CFR Part 71, the liquid effluent may be processed with the mobile liquid waste processing system in the Radwaste Building. When pre-process sampling and controls indicate that A2 quantity limits may be exceeded by processing liquid effluent in the Radwaste Building, liquid waste is processed in the Seismic Category I auxiliary building. Procedural controls ensure that the total cumulative source term of unpackaged wastes including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building is limited consistent with RG 1.143, Revision 2, dose acceptance criteria of less than 500 millirem at the protected area boundary or an unmitigated exposure, occurring over a two hour time period, of less than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory. The unmitigated, unshielded worker dose is calculated at 10 feet from the source. Unlimited worker occupancy workstations and low dose rate waiting areas are located no closer than 10 feet from a mobile radwaste processing system or a Waste Monitor Tank.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-023	LNP	2	11	11.02.01.02.05.02	NPD-NRC-2013-039	Revise the paragraph with a LMA of LNP COL 11.2-1 to read as follows: When mobile or temporary equipment is selected to process liquid effluents, the equipment design and testing meets the applicable requirements of Regulatory Guide 1.143. When confirmed through sampling that the radioactive waste contents result in an inventory on a mobile system that is below the A2 quantity limit for radionuclides specified in Appendix A to 10 CFR Part 71, the liquid effluent may be processed with the mobile liquid waste processing system in the Radwaste Building. When pre-process sampling and controls indicate that A2 quantity limits may be exceeded by processing liquid effluent in the Radwaste Building, liquid waste is processed in the seismic Category I Auxiliary Building. Procedural controls also ensure that the total cumulative source term of unpackaged wastes including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building is limited consistent with RG 1.143, Revision 2, unmitigated radiological release criteria, so that an unmitigated release, occurring over a two hour time period, would not result in a dose of greater than 500 millirem at the protected area boundary, or an unmitigated exposure, occurring over a two hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory. The unmitigated, unshielded worker dose is calculated at 10 feet from the source. Unlimited worker occupancy workstations and low dose rate waiting areas are located no closer than 10 feet from a mobile radwaste processing system or a Waste Monitor Tank.
LNP-055	LNP	2	11	11.02.01.02.04	Errata	Add "(Reference 201)" after "NEI 08-08A" in the second paragraph of LNP FSAR Section 11.2.1.2.4 (third to last sentence).
LNP-024	LNP	2	11	11.04.06.03.01	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn
LNP-025	LNP	2	13	13.04.T/T13.4-201	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.
LNP-026	LNP	2	13	13.04.T/T13.4-201	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.
LNP-027	LNP	2	13	13.05.02.02.08	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.
LNP-028	LNP	2	13	13.05.02.02.10	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.
LNP-029	LNP	2	13	13.06	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-057	LNP	2	13	13.01.02.01	Errata	In Part 2, FSAR Subsection 13.1.2.1, change the fourth bullet of the third paragraph to read: "Programs and procedures for rules of practice as described in Section 5.2 of N18.7-1976/ANS-3.2 (Reference 203)"
LNP-082	LNP	2	13	13.05.01	NPD-NRC-2014-027	COLA Part 2, FSAR Chapter 13 will be revised to add text to Section 13.5.1, "Administrative Procedures" under the statement" The plant administrative procedures provide procedural instructions for the following:", 19th bullet as shown below. The left-hand margin annotation for this added text will be "LNP COL 13.5-1" • A process for implementing the safety/security interface requirements of 10 CFR 73.58. "A process is in effect at the time of issuance of the combined license and was developed using NRC endorsed industry guidance. This process is used to manage safety/security interface while the security procedures and emergency plan implementing procedures are being developed and implemented."
LNP-030	LNP	2	13	13.05.02.02.05	NPD-NRC-2013-036 Superseded by NPD-NRC-2013-039	Revise the text at the end of FSAR Section 13.5.2.2.5, "Radioactive Waste Management Procedures," with a LMA of LNP COL 13.5-1, to read as follows: As required by License Condition, operating procedures that include provisions to assure that A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 are not exceeded will be developed, implemented and maintained prior to initial fuel load. Procedural controls limit the radionuclide inventory to less than the A2 limit in each of the three (3) monitor tanks, and in each of up to three (3) mobile radwaste processing systems. Procedures also ensure that any additional equipment to be located in the RWB is limited to A2 quantities. Spent media transfer from a mobile radwaste processing system located in the Radwaste Building is procedurally controlled such that spent media transfer and packaging for offsite shipment must be complete prior to placing the mobile radwaste processing system back into service. The procedures ensure that the total cumulative source term of unpackaged wastes, including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building is limited consistent with RG 1.143, Revision 2, unmitigated radiological release and exposure acceptance criteria of less than 500 millirem at the protected area boundary or an unmitigated exposure, occurring over a two hour time period, of less than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory. The unmitigated, unshielded worker dose is calculated at 10 feet from the source. Unlimited worker occupancy workstations and low dose rate waiting areas are located no closer than 10 feet from a mobile radwaste processing system or a Waste Monitor Tank. The liquid radwaste system is discussed in Section 11.2.
LNP-031	LNP	2	13	13.05.02.02.05	NPD-NRC-2013-039	Revise the text at the end of FSAR Section 13.5.2.2.5, "Radioactive Waste Management Procedures," with a LMA of LNP COL 13.5-1, to read as follows: As required by License Condition, operating procedures that include provisions to assure that A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 are not exceeded will be developed, implemented and maintained prior to initial fuel load. Procedural controls limit the radionuclide inventory to less than the A2 limit in each of the three (3) monitor tanks, and in each of up to three (3) mobile radwaste processing systems. Procedures also ensure that any additional equipment to be located in the Radwaste Building is limited to A2 quantities. Spent media transfer from a mobile radwaste processing system located in the Radwaste Building is procedurally controlled such that spent media transfer and packaging for offsite shipment must be complete prior to placing the mobile radwaste processing system back into service. The procedures also ensure that the total cumulative source term of unpackaged wastes, including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building is limited consistent with RG 1.143, Revision 2, unmitigated radiological release criteria, so that an unmitigated release, occurring over a two hour time period, would not result in a dose of greater than 500 millirem at the protected area boundary, or an unmitigated exposure, occurring over a two hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory. The unmitigated, unshielded worker dose is calculated at 10 feet from the source. Unlimited worker occupancy workstations and low dose rate waiting areas are located no closer than 10 feet from a mobile radwaste processing system or a Waste Monitor Tank. The liquid radwaste system is discussed in Section 11.2.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-032	LNP	2	13	13.07	NPD-NRC-2014-019	Revise the first five bullets of LNP SUP 13.7-1 on pages 13.7-1 and 13.7-2 to read as follows: • The construction site area is defined in the Physical Security Plan and will be under the control of the primary site contractor. The 10 CFR Part 26 requirements will be implemented for the construction site area based on the descriptions provided in Table 13.4-201. • Construction Workers & First Line Supervisors (primary site contractor employees and subcontractors) are covered by the Duke-approved construction FFD Program (elements Subpart K). • Duke employees and Duke subcontractor's construction management and oversight personnel are covered by a Duke Operations FFD Program and the primary site contractors's employees and the primary site contractor's subcontractors, construction management, and oversight personnel will be covered by the Duke-approved FFD Program (elements Subpart A - H, N and 0). • Duke security personnel are covered by a Duke Operations FFD Program and the primary site contractor's security personnel are covered by the Duke-approved FFD Program (elements Subpart A - H, N and 0). This coverage is applicable from the start of construction activities to the earlier of (1) the receipt of SNM in the form of fuel assemblies, or (2) the establishment of a Protected Area, or (3) the 10 CFR 52.103(g) finding. • Duke FFD Program personnel are covered by a Duke Operations FFD Program and the primary site contractor's FFD Program personnel will be covered by the Duke-approved FFD Program (elements Subpart A - H, N and 0, and C per licensee's discretion).
LNP-033	LNP	2	14	14.03.T/T14.3-202	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Section 14.3 will be revised to add a departure from DCD Table 14.3-2, Design Basis Accident Analysis, Sheets 7 and 8 of 17, as new FSAR Table 14.3-202, Sheets 1 and 2, with a LMA of LNP DEP 3.2-1. This Table shall also be added to the list of tables in Chapter 14.
LNP-074	LNP	2	15	15.00.13	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.0.13,with a LMA of LNP DEP 3.2-1, to read: 15.0 13 Operator Actions Revise the first sentence of the first paragraph of DCD Subsection 15.0.13 to read as follows: For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable condition.
LNP-075	LNP	2	15	15.02.06.01	Enclosure 5 of NPD-NRC-2014-021	This section of the referenced DCD is incorporated by reference with the following departures and/or supplements. 15.2.6.1 Identification of Causes and Accident Description Revise the seventh sentence of the fourth paragraph of DCD Subsection 15.2.6.1 to read as follows: The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria.
LNP-076	LNP	2	19	19.59 T/T19.59-202	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Section 19.59 will be revised to add a departure from DCD Table 19.59-18, PRA Based Insights, Sheet 6 of 25, as new FSAR Table 19.59-202, Sheet 1,with a LMA of LNP DEP 6.3-1. This Table shall also be added to the list of tables in Chapter 19.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
						COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, will be revised as follows, with a LMA of LNP DEP 3.2-1: This section of the referenced DCD is incorporated by reference with the following departures and/or supplements. 19E.4.10.2 Shutdown Temperature Evaluation Revise the first and second paragraphs of DCD Subsection 19E.4.10.2 to read as follows: 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. 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 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 condition following postulated transients. A non-bounding, conservative analysis 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.
LNP-034	LNP	2	19	19E.04.10.02	Enclosure 5 of NPD-NRC-2014-021: Superseded by Enclosure 5 of NPD-NRC-2014-028	Add new paragraphs 3 and 4 to DCD Subsection 19E.4.10.2 to read as follows: 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. Revise the first sentence of the fifth paragraph (old third paragraph)of DCD Subsection 19E.4.10.2 to read as follows: Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip.
LNP-035	LNP	2	19	19E.04.10.02	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, will continue to be revised as follows, with a LMA of LNP DEP 3.2-1: Revise paragraphs 6 and 7 (old paragraphs 4 and 5) of DCD Subsection 19E 4.10.2 to read as follows: Once actuated, at about 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, 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 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 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 seconds, while the core average temperature reaches 420°F within 124,400 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. 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.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-077	LNP	2	19	19E.04.10.02	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 19, Appendix 19E, will continue to be revised as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1: Add a new eighth paragraph to DCD Subsection 19E.4.10.2 to read as follows: 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.
LNP-078	LNP	2	19	19E.09	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Chapter 19 will be revised to add a new Subsection 19E.9, with a LMA of LNP DEP 3.2-1, to read: 19E.9 References 14. Not used.
LNP-036	LNP	2	19	19E.04.T/T19E.4.10-201	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Section 19E.4.10 will be revised to add a departure from DCD Table 19E.4.10-1, Sequence of Events Following a Loss of AC Power Flow with Condensate from the Containment Shell Being Returned to the IRWST, as new FSAR Table 19E.4.10-201, with a LMA of LNP DEP 3.2-1. This Table shall also be added to the list of tables for Chapter 19.
LNP-037	LNP	2	19	19E.04.F/F19E.4.10-201 through 19E.4.10-204	Enclosure 5 of NPD-NRC-2014-021	COLA Part 2, FSAR Section 19E will be revised to add a departure from DCD Figures 19E.4 10-1 through 19E.4.10-4 as Figures 19E.4.10-201 through 19E.4.10-204, with a LMA of LNP DEP 3.2-1. These figures shall also be added to the list of figures for Chapter 19.
LNP-054	LNP	2	19	19.58.T/T19.58-201	Errata	A review of Report No. LNG-PRA-N4R-001, Rev. 4, found that the value for an EF2 Tornado event frequency in Levy COLA Table 19.58-201 (sheet 1 of 7) was incorrectly stated as 5.21E-05. Update the value for an EF2 Tornado event frequency in Levy COLA Table 19.58-201 (sheet 1 of 7) from 5.21E-05 to 5.08E-05.
LNP-059	LNP	2	19	19.58.04	Errata	Reference number 19.58.4-203 changed to "Not Used".
Part 4						
LNP-038	LNP	4		Section B - TS Bases 3.3.3 LCO 11	Enclosure 5 of NPD-NRC-2014-021	Revise LCO 11 for TS Bases B 3.3.3, last sentence of the first paragraph, to read as follows, with a LMA of LNP DEP 3.2-1: The condensate is returned to the IRWST via a gutter and downspouts.
LNP-039	LNP	4		Section B - TS Bases 3.5.4 Background	Enclosure 5 of NPD-NRC-2014-021	Revise the first two sentences of the third paragraph for TS Bases B 3.5.4, Background, to read as follows, with a LMA of LNP DEP 3.2-1: In order to preserve the IRWST water for long-term PRHR HX operation, downspouts and a gutter are provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump.
LNP-040	LNP	4		Section B - TS Bases 3.5.4 Surveillance Requirements	Enclosure 5 of NPD-NRC-2014-021	Revise SR 3.5.4.7 of TS Bases B 3.5.4, Surveillance Requirements, to read as follows, with a LMA of LNP DEP 3.2-1. This surveillance requires visual inspection of the IRWST gutters and downspout screens to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the gutters or downspout screens could become restricted.
Part 7						
LNP-041	LNP	7		A	Departure 3.7-1 to add drilled shafts departure from NPD-NRC-2013-037	Include departure for Tier 2 information revised in DCD Subsections 3.7.2.8.1 and 3.7.2.8.3 as identified in FSAR Table 1.8-201. Insert the text from COLA change item 4. as identified in the enclosure to NPD-NRC-2013-037 following the information for Departure LNP-DEP 3.2-1 and preceding the information for Departure LNP DEP 3.11-1.
LNP-042	LNP	7		A	Departure 3.7-1 to add drilled shafts departure from NPD-NRC-2013-037	Add line item as the fourth line item to the table of departures in Section A, to read as follows: LNP DEP 3.7-1 Use of site-specific horizontal seismic response spectra for the design of drilled shafts that support the seismic Category II portions of the Annex and Turbine Buildings.
LNP-043	LNP	7		A and B	NPD-NRC-2014-005 and NPD-NRC-2014-023	COLA Part 7, Departures and Exemption Requests, will be revised to incorporate the revised exemption and departure for the changes to the containment cooling portion of the Passive Core Cooling System Condensate Return as addressed in Enclosure 7 of NPD-NRC-2014-005.
LNP-079	LNP	7		A and B	Enclosure 5 of NPD-NRC-2014-021	COLA Part 7, Departures and Exemption Requests, will be revised to incorporate the revised departure 3.2-1, added departure 6.3-1, and exemption pertaining to the changes to the containment cooling portion of the Passive Core Cooling System Condensate Return as addressed in Enclosure 5 of NPD-NRC-2014-021.
Part 9						

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-058	LNP	9		09.01	Revisions to financial information	Updated financial information to reflect most current information.
Part 10						
LNP-044	LNP	10		LC#11	NPD-NRC-2014-001	Revise COLA Part 10, License Condition 11.E, from: E. DEF will distribute initial LNP public information publications, developed in coordination with CR3 and consistent with the LNP Emergency Plan, to the public within 180 days prior to fuel load at LNP. To read: E. DEF shall distribute the initial LNP public information publications, consistent with the LNP emergency plan, within 180 days prior to fuel load at LNP. DEF must coordinate the development, initial and annual redistribution, and maintenance of this information with CR3 as long as the NRC requires CR3 to distribute public information publications.
LNP-045	LNP	10		ITAAC Table 3.8-1	NPD-NRC-2014-001	Revise COLA Part 10, Table 3.8-1 Sheet 5 of 29, "Inspections, Tests, Analyses" text for Item 7.2 from: 7.2 A test of the EOF will be performed, including a test of the capabilities. To read: 7.2 An inspection of the as-built EOF will be performed, including a test of the capabilities. The EOF will meet the criteria of NUREG-0696 and 0737.
LNP-046	LNP	10		ITAAC Table 3.8-1	NPD-NRC-2014-001	Revise COLA Part 10, Table 3.8-1 Sheet 5 of 29 to add the three additional acceptance criteria below after 7.2.2: 7.2.3 The EOF is structurally built in accordance with the Uniform Building Code. 7.2.4 The EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment. 7.2.5 The EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.
LNP-047	LNP	10		LC#3,C	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.
LNP-048	LNP	10		Appendix B	NPD-NRC-2014-002	In LNP COLA Part 10, Appendix B. Inspections, Tests, Analyses and Acceptance Criteria, insert COLA change item #3 (on pages 14 of 16 and 15 of 16 of NPD-NRC-2014-002 enclosure) after Section 2.5.12, Closed Circuit TV System and before the discussion on DCD Tier 1 Section 2.6.11:
LNP-049	LNP	10		Appendix B	Superseded by NPD-NRC-2014-009 NPD-NRC-2014-002	In LNP COLA Part 10, Appendix B. Inspections, Tests, Analyses and Acceptance Criteria, insert COLA change item #4 (on page 16 of 16 of NPD-NRC-2014-002 enclosure) as a new line item 7 in Table 2.6.12-1:
LNP-050	LNP	10		Appendix B	NPD-NRC-2014-009	In LNP COLA Part 10, Appendix B. Inspections, Tests, Analyses and Acceptance Criteria, insert COLA change item #4 (on page 16 of 16 of NPD-NRC-2014-009 enclosure) as a new line item 7 in Table 2.6.12-1:
LNP-051	LNP	10		LC#12	NPD-NRC-2013-041	Revise License condition for spent fuel pool instrumentation in accordance with referenced letter
LNP-080	LNP	10		LC#12	Revisions to Fukushima Response Actions	Per NRC E-mail of 6/27/14, revise the wording of LNP License Conditions 12.A and 12.C for the revisions to the Fukushima Response Actions License Condition text.

Change ID#	COLA	COLA Part	Chapter	Section	Basis for Change	Change Summary
LNP-052	LNP	10		LC#13	NPD-NRC-2013-036	Revise the following text at the end of the Proposed License Conditions in LNP COLA Part 10: 13. RADWASTE BUILDING RADIOACTIVITY LIMITS PROPOSED LICENSE CONDITION: Prior to initial fuel load, the licensee shall develop, implement, and maintain procedural controls limiting radionuclide inventory in each of the Radwaste Building Monitor Tanks, and separately in each of up to three (3) Radwaste Building mobile radwaste processing systems to below A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 (Tables A-1 and A-3), as described in FSAR Section 13.5.2.2.5. The procedures shall also ensure that any additional equipment located in the RWB is limited to the A2 quantities and that the total cumulative radioactive inventory contained in unpackaged wastes (including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building) is limited so that an unmitigated release, occurring over a two hour time period, would not result in a dose of greater than 500 millirem at the protected area boundary or an unmitigated exposure, occurring over a two hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory.
LNP-056	LNP	10		LC#11	LNP Chapter 13 ASER (2014)	Revise the text of the Proposed License Conditions #11A, 11B, and 11C based on text from LNP Chapter 13 ASER (dated 6/30/2014): Changes to LNP Part 10 License Condition 11 are needed to align with statements in the Chapter 13 ASER. Basically, parts of LC 11B is put into LC 11A and LC 11C and then LC 11B is deleted (a new version of the ASER was provided to LNP by the NRC for review on 6/30/2014 and the text in LC 11C was revised from what was in the original ASER)
LNP-060	LNP	10		Appendix B	Passive Containment Cooling System ITAAC per LNP DEP 3.2-1	In Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, insert the following prior to the information on Physical Security ITAAC: Passive Containment Cooling System ITAAC Passive Containment Cooling system components are added to support the capability of the Passive Residual Heat Removal Heat Exchanger (PRHR HX) to enable the reactor to achieve a safe shutdown condition of 420o F within 36 hours. Component numbers for downspout screens are added to DCD Tier 1 Table 2.2.3-1 and component numbers for downspout piping are added to DCD Tier 1 Table 2.2.3-2 to provide assurance that ITAAC design commitments will be met. These tables, with the subject component numbers added, are provided in the attached Tables 2.2.3-1 and 2.2.3-2, with a LMA of LNP DEP 3.2-1. In Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, insert the three pages from DCD Tier 1 Tables 2.2.3-1 and 2.2.3-2 containing the component numbers for downspout screens and downspout piping associated with LNP DEP 3.2-1 prior to the Table 2.6.9-2 Physical Security ITAAC.
Part 11						
LNP-053	LNP	11		11G	NPD-NRC-2014-004 Superseded by NPD-NRC-2014-018	DELETED CHANGE. NPD-NRC-2014-018 was submitted to withdraw NPD-NRC-2014-004 and therefore the associated proposed COLA changes are also withdrawn.