APPENDIX F

REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 7, 2011

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: REPORT ON THE SAFETY ASPECTS OF THE PROGRESS ENERGY FLORIDA, INC. COMBINED LICENSE APPLICATION FOR LEVY NUCLEAR PLANT, UNITS 1 AND 2

Dear Chairman Jaczko:

During the 589th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 1-3, 2011, we reviewed the NRC staff's Advanced Safety Evaluation (ASE) for the pending Progress Energy Florida, Inc. (PEF) Combined License Application (COLA) for the Levy Nuclear Plant (LNP), Units 1 and 2. This application incorporates the Westinghouse Electric Company (WEC) AP1000 certified design, and it conforms to the design-centered review approach (DCRA).[1] The DCRA is Commission policy which allows the staff to perform one technical review and reach a decision for each COLA standard issue outside the scope of the design certification and to use this review and decision to support decisions on multiple COLAs.

The first COLA that receives a complete NRC staff review is designated as the reference COLA (RCOLA). Any subsequent application referencing the same design is designated as a subsequent COLA (SCOLA). We reviewed Southern Nuclear Operating Company's Vogtle Electric Generating Plant (VEGP), Units 3 and 4, RCOLA and issued a letter report on January 24, 2011. We reviewed South Carolina Electric and Gas Company's V.C. Summer Nuclear Station, Units 2 and 3, SCOLA and issued a letter report on February 17, 2011.

The LNP COLA is the second AP1000 SCOLA. Our AP1000 Subcommittee held a meeting on October 18-19, 2011, to review the SCOLA and the staff's ASE. During the meeting, we met with representatives of the NRC staff, PEF and its vendors, and with the public. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 52.87 that the ACRS report on those portions of the application which concern safety.

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1 The DCRA is described in Regulatory Issue Summary (RIS) 2006-06, "New Reactor Standardization Needed to Support the Design-Centered Licensing Review Approach," as endorsed by the Commission's Staff Requirements Memorandum in response to SECY-06-0187, "Semiannual Update of the Status of New Reactor Licensing Activities and Future Planning for New Reactors," dated November 16, 2006.

The VCSNS COLA is an AP1000 SCOLA. Our AP1000 Subcommittee held two meetings (July 21-22, 2010, and January 10-11, 2011) to review various chapters of the SCOLA and the staff's ASER. During these meetings, we met with representatives of the NRC staff, SCE&G and its vendors, and with the public. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 52.87 that the ACRS report on those portions of the application which concern safety.

CONCLUSIONS AND RECOMMENDATIONS

1. There is reasonable assurance that VCSNS, Units 2 and 3, can be built and operated without undue risk to the health and safety of the public. The SCOLA for VCSNS, Units 2 and 3, should be approved following its final revision.

2. Recommendations 2 through 5 in our January 24, 2011, letter concerning the VEGP, Units 3 and 4, RCOLA are also applicable to the VCSNS, Units 2 and 3, SCOLA.

3. The staff should limit the use of the current version of the HABIT code to neutral density gas dispersion modeling.

BACKGROUND

By letter dated March 27, 2008, SCE&G submitted a combined license application to the NRC for VCSNS, Units 2 and 3, in accordance with the requirements of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." In the application, SCE&G stated that VCSNS, Units 2 and 3, would be two Westinghouse AP1000 advanced pressurized water reactor units and would be located at the existing VCSNS site.

As an AP1000 SCOLA, SCE&G has organized and annotated its application to identify: a) sections that incorporate by reference the AP1000 DCD; b) sections that are standard for COL applicants in the AP1000 RCOLA; and c) sections that are site-specific and thus only apply to VCSNS, Units 2 and 3.

DISCUSSION

Our review of the VCSNS, Units 2 and 3, SCOLA was conducted in parallel with our review of both the AP1000 Design Certification Amendment application and the VEGP, Units 3 and 4, RCOLA. As a consequence, the RCOLA and SCOLA on which the staff's ASER is based reference Revision 17 of the DCD, whereas the current version is Revision 18, and there may be a further revision prior to certification rulemaking. Similarly, the SCOLA utilizes standard content in the RCOLA which may be revised prior to approval. Since the remaining licensing steps do not provide for further ACRS review of the DCD, RCOLA, or VCSNS Units 2 and 3 SCOLA revisions that incorporate changes in design and commitments made by applicants during our reviews, the staff should review with us any changes and commitments which deviate significantly from those presented during our review.

Since the VCSNS, Units 2 and 3, SCOLA relies on the standard information found in the RCOLA, the recommendations described in our January 24, 2011, letter concerning the VEGP, Units 3 and 4, RCOLA in the following areas are also applicable to our VCSNS, Units 2 and 3, SCOLA assessment: containment interior debris limitation, in-service inspection/ in-service testing program requirements for squib valves, power uncertainty measurement, and incorporation of DCD or COLA changes. Likewise, the discussion of site-specific probabilistic risk assessment in our January 24, 2011, letter is applicable.

The V. C. Summer Nuclear Station Site

VCSNS is located approximately 30 miles northwest of Columbia, in Jenkinsville, South Carolina. The site location is adjacent to, and elevated about 150 ft. above, the Parr Reservoir which is created by a dam on the Broad River. It is also adjacent to the Monticello Reservoir. A nearby pumped storage facility connects the two reservoirs. VCSNS Unit 1 began commercial operation in 1984. The site location relative to water courses and topography effectively precludes flooding as a hazard to the site. The expanded three-unit nuclear station, in addition to the pumped storage facility, will be served by twelve 230 kV transmission lines.

Offsite Hazards

The review of offsite hazards for VCSNS, Units 2 and 3, included toxic gas that might be released by a transportation accident on the Norfolk Southern rail line located approximately one mile from the plant. SCE&G used a public domain United States Environmental Protection Agency developed computer code, ALOHA, which treats appropriately the modeling of the dispersion of both heavy and neutral-density gases.

Analysis results using ALOHA showed that vapor cloud explosions do not pose a threat to safety-related structures, systems, and components at VCSNS, Units 2 and 3. The analysis was performed using conservative assumptions such as dispersion over flat terrain, whereas the plant is located well above possible release locations on the rail line. Shock pressures were well below 1 psi, which is considered the minimum pressure wave amplitude to cause damage. The analysis also showed that toxic vapor clouds would not lead to control room concentrations that would pose a threat to operators.

For its confirmatory calculations of toxic gas effects, the staff used the HABIT code. However, HABIT only models neutral density gas dispersion and does not consider heavy gas effects. The calculated concentrations are lower than those in the ALOHA analyses, which is to be expected in view of several postulated releases consisting of heavy gases, which disperse more slowly.

In our letter report dated September 16, 1999, we recommended that “the staff should document evidence of the validity and the capability of computer codes endorsed in regulatory guides such as the HABIT code.” During our full committee meeting on February 10, 2011, the staff stated that it is pursuing validation of some aspects of the HABIT code. We recommend that use of the current version of HABIT be limited to neutral density gas dispersion modeling.

Seismic Source Model

SCE&G used source models provided by the Electric Power Research Institute. These were updated in light of more recent data and evolving knowledge, particularly for the Charleston and New Madrid Seismic Source Zones. No modifications to the Eastern Tennessee Seismic Source Zone were required. The VCSNS, Units 2 and 3, site-specific safe shutdown earthquake (SSE) was developed in accordance with Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," and information that was used in the VEGP, Units 3 and 4, Early Site Permit review and approval. Following our initial subcommittee meeting in July 2010, the seismic source information was updated.

Seismic Design Parameters

The peak ground acceleration (PGA) values for horizontal and vertical ground motions are 0.23g and 0.22g, respectively. The input seismic design ground motion response spectra (GMRS) for the SSE in the free field at plant grade exceeds the standard AP1000 certified seismic design response spectra (CSDRS) at frequencies of about 15 to 80 Hz (horizontal) and 20 to 80 Hz (vertical). However, the VCSNS site meets the AP1000 DCD criteria for a hard rock site, and the site-specific GMRS is bounded by the AP1000 hard rock high frequency spectrum. The staff concluded that the technical bases described in the AP1000 DCD were applicable to VCSNS, Units 2 and 3, for justifying that high-frequency exceedances of the AP1000 CSDRS are considered to be non-damaging.

Monitoring for Leakage from the Radioactive Waste Discharge Line

Liquid radioactive waste is diluted to below allowable offsite discharge limits by onsite blending with cooling tower blowdown. It then flows offsite through approximately one mile of high density polyethylene (HDPE) pipe downgrade to an outfall at the Parr Reservoir. Piping connections at the onsite blending location will be accessible for inspection, but the downstream portion of the line will be buried along a rail spur and will not be readily accessible for inspection.

Although this material has excellent properties and is acceptable for its intended service, operating experience in nuclear power plants is limited. Localized lack of fusion can occur during the joining of HDPE piping segments in the field. Such defects, if not detected by initial inspection and hydrostatic testing and repaired, can propagate through the pipe wall by slow crack growth. Since many joints will be formed in the field with no provision to inspect them using volumetric (UT) methods, undetected defects may grow and cause leaks during the 60-year service life of the pipe.

Monitoring wells will be relied upon as the only method for detecting groundwater contamination. SCE&G's groundwater monitoring program should be designed to provide for early detection of any leaks that develop in the HDPE waste water discharge line. The monitoring wells should detect contamination close to the pipe along its entire run, before it becomes widespread, and well before compliance with 10 CFR 20.1406 is challenged.

Deviation from RCOLA Standard Approach

As compared to the VEGP RCOLA, the VCSNS, Units 2 and 3, SCOLA included only one additional departure or exemption of note from the DCD. There is a slight increase in the maximum, safety, non-coincident wet bulb temperature of 1.2°F above the AP1000 DCD value of 86.1°F. The effects of this increase were evaluated by the staff and determined to be acceptable.

In summary, we agree with the staff's conclusions as documented in the staff's ASER regarding the safety issues associated with the SCE&G COLA for VCSNS, Units 2 and 3. We conclude that there is reasonable assurance that VCSNS, Units 2 and 3, can be built and operated without undue risk to the health and safety of the public. The SCE&G COLA for VCSNS, Units 2 and 3, should be approved following its final revision.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

REFERENCES

1. Southern Carolina Electric and Gas Company (SCE&G) Letter, "Combined License Application for V.C. Summer Nuclear Station Units 2 and 3," dated March 27, 2008 (ML081300460)

2. SCE&G Letter, "Combined License Application for V.C. Summer Nuclear Station Units 2 and 3," Revision 2, dated January 28, 2010 (ML100350739) (Rev. 2 was used as the basis for the staff's ASER)

3. During the course of ACRS review, the staff provided the following ASER chapters:

Chapter	Chapter Title	Transmittal Memo to ACRS (Accession Numbers)	ASER (Accession Numbers)
1	Introduction and Interfaces	ML101550427	ML101370358
2	Site Characteristics (without Hydrology)	ML101550273	ML101390008
	Section 2.4 (Hydrology)	ML102450029	ML102140255
3	Design of Structures, Components, Equipment, and Systems	ML101550236	ML103070512
4	Reactor	ML101450515	ML100621218
5	Reactor Coolant System and Connected Systems	ML101550558	ML100670055
6	Engineered Safety Features	ML102080334	ML102980694
7	Instrumentation and Controls	ML101540411	ML101370712
8	Electric Power	ML101540620	ML102370262
9	Auxiliary Systems	ML101540643	ML102670044
10	Steam and Power Conversion Systems	ML101450456	ML101020031
11	Radioactive Waste Management	ML101550661	ML100700102
12	Radiation Protection	ML101550687	ML101820007
13	Conduct of Operations (without Emergency Planning)	ML103200058	ML100840174
	Section 13.3 (Emergency Planning)	ML101550691	ML102020681
14	Initial Test Programs	ML101550695	ML102660181
15	Accident Analysis	ML101550697	ML103070532
16	Technical Specifications	ML101550699	ML101890864
17	Quality Assurance	ML101550701	ML101890606
18	Human Factors Engineering	ML101550703	ML101250016
19	Probabilistic Risk Assessment	ML103010338	ML102950269
19 Appendix 19.A	Loss of Large Areas of the Plant due to Explosions or Fires (LOLA)	ML101590342	Public Version ML103350636 Non-Public Version ML103370008
Appendix A	License Conditions, ITAAC, and FSAR Commitments	ML101550427	ML103360056

4. ACRS Letter, “Report on the Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document,” dated December 13, 2010 (ML103410351)

5. ACRS Letter, “Long-Term Core Cooling for the Westinghouse AP1000 Pressurized Water Reactor,” dated December 20, 2010 (ML103410348)

6. ACRS Letter, “Report on the Safety Aspects of the Southern Nuclear Operating Company Combined License Application for Vogtle Electric Generating Plant, Units 3 and 4,” January 24, 2011 (ML110170006)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April 18, 2016

The Honorable Stephen G. Burns
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: EXEMPTIONS TO THE AP1000 CERTIFIED DESIGN INCLUDED IN THE LEVY NUCLEAR PLANT UNITS 1 AND 2 COMBINED LICENSE APPLICATION

Dear Chairman Burns,

During the 633rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), April 7-9, 2016, we reviewed five exemption requests for the Westinghouse Electric Company (WEC) AP1000 certified design which Duke Energy Florida, LLC (Duke Energy) has included in the combined license application (COLA) for the Levy Nuclear Plant (Levy) Units 1 and 2. We also reviewed the NRC staff's related Advanced Safety Evaluation Report (ASER), Chapter 21. The exemptions include changes that are grouped into six departures from the AP1000 Design Control Document (DCD), Revision 19. Our AP1000 Subcommittee held a meeting on April 5, 2016, to review the departures and the staff's ASER. The Subcommittee also met with Duke Energy, WEC, and the staff on April 9 and September 17, 2014, to review the development of the changes that are needed to achieve the intended design functions for passive residual heat removal (PRHR). These changes are included in the exemption concerning condensate return and PRHR.

During the meeting, we had the benefit of discussions with representatives of the staff, Duke Energy, and WEC, and we had input from members of the public. We also had the benefit of the referenced documents. This report fulfills the requirement of 10 CFR 52.87 that the ACRS report on those portions of the application which concern safety.

CONCLUSIONS AND RECOMMENDATION

1. Five exemptions to the AP1000 certified design have been included in the Levy combined license application. The five exemptions are needed to enable the certified design to perform intended functions and should be approved.

2. The causes for the exemptions have been identified and addressed for the AP1000 certification.

3. Generic lessons learned, relative to the reactor design process leading to certification, should be identified and further evaluated.

BACKGROUND

By letter dated July 28, 2008, Progress Energy Florida, Inc., now Duke Energy, submitted a COLA for Levy Units 1 and 2 to the NRC. On December 7, 2011, we issued a letter report to the Commission recommending approval following implementation of the stated recommendations. Subsequently, changes needed to achieve the intended design functions for PRHR were identified. Development of these changes was undertaken by WEC, with oversight from Duke Energy, and these changes are now required to be included in the COLA, pursuant to Interim Staff Guidance DC/COL-ISG-011. These departures are common to all COLAs referencing the AP1000 design, and similar changes will be necessary for AP1000 combined license holders.

Ongoing detailed design of the AP1000 units, and investigation into the extent of the condition that created the need for the PRHR-related changes, identified other needed changes requiring approval of exemptions in four additional areas. Duke Energy noted the areas requiring departures from the certified AP1000 design during our review of its William States Lee III Nuclear Station (Lee) Units 1 and 2 COLA in 2015. These were listed as follows in our letter, dated December 14, 2015, concerning the Lee COLA:

- Condensate return and PRHR
- Main control room operator dose
- Main control room heat load
- Plant monitoring system flux doubling to comply with IEEE 603
- Hydrogen vent in containment

DISCUSSION

The five exemptions and associated departures from the AP1000 certified design are needed to implement intended functions of the certified design. Each is distinct and separate from the others. The changes will be made for the common purpose of correcting errors and omissions in the certified design, which have been identified during licensing and detailed design development subsequent to certification. Therefore, we also reviewed elements that are common to the departures; in particular, the implementation of the quality assurance program requirements in 10 CFR Part 50, Appendix B during design. Finally, we also reviewed the staff's assessment of the effect of the departures on the previously completed probabilistic risk assessment.

Condensate Return and Passive Residual Heat Removal

The AP1000 design provides for closed-loop cooldown and passive heat removal under accident conditions not involving loss of coolant. Reactor coolant circulates naturally through a PRHR heat exchanger located within the in-containment refueling water storage tank (IRWST). The PRHR heat exchanger converts IRWST water to steam, and the subsequent condensation of this steam on the containment vessel interior surface passively transfers residual heat by conduction through the containment wall to the outside air. This closed-loop cooling requires that sufficient condensed water be returned to the IRWST to ensure the inventory needed to maintain the cooldown status and to continue the PRHR process for as long as necessary.

Features in the containment that are required to direct condensate back to the IRWST are described in AP1000 DCD, Revision 19. The rate of condensation varies with time, and the return of condensate to the IRWST is subject to some loss. A constant loss rate of 10 percent was assumed in the DCD analysis. Based on this assumption, DCD, Revision 19 states that (a) acceptance criteria associated with the Chapter 15 design basis safety analyses remain satisfied indefinitely, and (b) cooldown to 420°F can be achieved in 36 hours and maintained indefinitely, based on Chapter 19 assumptions and acceptance criteria.

Duke Energy has proposed for its Levy COLA an exemption seeking approval of two departures that concern cases (a) and (b) above. These departures involve physical changes in containment to increase condensate return. Downspouts, collection points, and connecting piping have been added to the polar crane girder and the internal stiffener, and many attachment plates on the containment inner surface have been eliminated. Additional testing was performed to estimate better the condensate collection on surfaces and losses at discontinuities such as attachment plates and to provide an improved basis for the estimation of condensate losses.

Based on testing and the additional features provided to return sufficient condensate back to the IRWST, a loss rate of 18 percent of the water that condenses on the containment vessel inner surface has now been assumed for cases (a) and (b) above. Water that condenses on other surfaces within containment is assumed to be entirely lost to the IRWST.

Analyses by WEC and the staff of PRHR performance were extensive. WEC used WGOTHIC and LOFTRAN with some confirmatory analyses using RELAP. Adiabatic and heat-loss models of the reactor coolant system, and the potential loss of subcooling in the reactor coolant system on heat transfer in the PRHR heat exchanger, were examined. The staff's confirmatory calculations used MELCOR and RELAP, and their results agreed well with the WEC calculations. The analyses included both the most limiting Chapter 15 non-loss-of-coolant-accident transient that credits the PRHR heat exchanger, which is the loss of normal feedwater coincident with the loss of AC power to the plant auxiliaries, and the safe shutdown analysis in Chapter 19. Based on these analyses, the duration for case (a) was extended to 72 hours, and the duration for case (b) was revised from an indefinite period to at least 14 days. Also, criteria for activation of the backup automatic depressurization system in order to establish open loop PRHR were updated.

Main Control Room Operator Dose

WEC identified several discrepancies in the certified design analyses supporting the determination of main control room (MCR) operator dose following a design basis accident (DBA). Specifically, (1) the analyses did not account for the direct dose from the MCR emergency ventilation system filter, (2) the normal ventilation system radiation monitor setpoints were not based upon all DBA release scenarios, and (3) the methodology used to estimate MCR dose contribution from direct radiation and skyshine was not up-to-date.

This exemption includes changes which add shielding for the ventilation filter, reduce the allowable secondary coolant iodine activity, update the radiation dose analyses, and revise the normal ventilation system radiation monitor logic and setpoints. The result of the changes provides a revised MCR dose for the DBA, which slightly increases the margin to the 5 rem limit.

Main Control Room Heat Load

Duke Energy identified that heat sources in the MCR had increased with detailed design development and now exceed those assumed in the certified design. Also, the design had not considered an event in which the MCR could be isolated and dependent on the emergency ventilation system, while offsite power remained available and powering certain MCR equipment. This event results in significantly higher heat loads than are considered in the certified design.

The exemption includes changes that add automatic, two-stage de-energization of select non-safety MCR heat loads. This load shed retains power for plant controls and parameter indications at the operators' normal work stations. Also, changes were made to establish limits, with surveillance requirements, for the initial MCR conditions and to ensure operation of the electrical load shedding functions.

With these changes, analysis projects that operators may remain in the MCR indefinitely, consistent with NUREG-0700 limits, following its isolation and resulting dependence on the emergency ventilation system.

Plant Monitoring System Compliance with IEEE 603

The source range neutron flux logic is a control system feature of the plant monitoring system that isolates dilute water sources to the reactor coolant system, in order to protect against inadvertent criticality due to boron dilution during shutdown conditions. Under some plant conditions, it is necessary to manually block or bypass the operation of this feature.

Operating bypasses are addressed in IEEE Standard 603-1991, and this standard is applicable to COLAs referencing the AP1000 certified design. WEC identified that, due to an omission, the certified design did not meet the requirements of the standard because this protection function could be blocked and would not be reset automatically when plant conditions require it. The exemption includes a change that will revise the plant monitoring system logic to comply with the standard and with regulatory requirements.

Hydrogen Vent Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC)

WEC identified that changes in structural details internal to the containment have occurred which are inconsistent with the certified design ITAAC for one of the compartments, relative to the venting of any hydrogen accumulation in the compartment following a severe accident. The

departure change to the ITAAC recognizes the possibility of a standing hydrogen flame that is closer to the containment boundary than allowed by the current ITAAC. Although the possible standing flame is closer to the containment boundary, results from analyses indicate that the higher temperatures would not compromise the structural integrity of the containment wall or of the equipment hatch cover and seals, and therefore, is acceptable.

NRC Staff Review

On March 7, 2016, the ASER for the five exemptions included in the Levy COLA was transmitted to the ACRS for review. It documents the staff's very thorough and technically complete review of the changes as they were developed over the past three years. The staff has identified that each of the exemptions is necessary in order to perform the intended functions, and therefore, meet the underlying purposes of the AP1000 certification rule.

The concluding statement in ASER Section 21.0 is "The staff finds that the cumulative risk impact of these design changes and departures is negligible." The changes are necessary to perform the intended functions that were the basis for the DCD risk calculation. However, the risk has not been calculated for the condition without the changes. While it is clear that there has been no increase in risk, it should not be concluded that the actual reduction in risk achieved by these changes is negligible.

Design Certification Quality Assurance Program

Detailed development of a certified design, involving the increasing engagement of combined license holders and applicants, should be expected to identify needed design and analysis changes. However, there are lessons to be learned from the Levy COLA experience.

Following initial discussions with our Subcommittee in 2014, WEC, Duke Energy, and the staff performed thorough evaluations, including the quality assurance program implementation. The results were reflected in the April 2016 Committee presentations. We conclude that the causes of the errors and omissions that made these exemptions necessary were addressed and programmatic changes applicable to the AP1000 certification were made where necessary.

We recommend that staff evaluate on a generic basis whether there are any lessons learned, relative to ongoing and future oversight of the quality assurance program implementation during development of designs seeking certification under 10 CFR Part 52. Prospective combined license applicants may not be in a position to provide such oversight during this phase, and they may find it difficult to do so following certification when customer oversight can be more effective. We would appreciate the opportunity to meet with the staff on this generic matter at an appropriate time.

Conclusion

The five exemptions, which include six departures from the AP1000 certified design that will be included in the Levy Units 1 and 2 COLA, effectively address errors and omissions in the current certification and should be approved. As indicated in our letter on the Lee Units 1 and 2 COLA, dated December 14, 2015, other combined license applicants referencing the AP1000 certified design will also include the exemptions in accordance with the design centered review approach described in that letter. Current combined license holders will submit license amendments to incorporate these, or similar, changes.

Sincerely,

/RA/

Dennis C. Bley
Chairman

REFERENCES

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, “Supplemental Response to NRC RAI Letter 124 - SRP Section 6.3 to Address Containment Condensate Return Cooling Design,” January 14, 2016 (ML16020A105).

2. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, “Revised Partial Response to Request for Additional Information Letter No. 121 Related to SRP Section 6.2.5, Combustible Gas Control in Containment,” January 6, 2016 (ML16008A082).

3. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, “Departure from AP1000 DCD Revision 19 to Address Compliance with IEEE 603-1991,” September 1, 2015 (ML15247A153).

4. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, “Revised Response to Request for Additional Information Letter No. 121 Related to SRP Section 6.2.5 and 6.4 for the Levy Nuclear Plant, Units 1 and 2 Combined License Application,” July 1, 2015 (ML15189A255).

5. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, “Response to Request for Additional Information Letter No. 122 Related to SRP Section 6.4, Control Room Habitability,” March 26, 2015 (ML15089A193).

6. U.S. Nuclear Regulatory Commission, “Levy, Units 1 and 2 – Chapter 21, ‘Design Changes Proposed in Accordance with ISG-11’,” March 7, 2016 (ML16026A016).

7. Progress Energy, “Application for Combined License for Levy Nuclear Power Plant Units 1 and 2,” July 28, 2008 (ML082260277).

8. Advisory Committee on Reactor Safeguards, “Report on the Safety Aspects of the Progress Energy Florida, INC. Combined License Application for Levy Nuclear Plant, Units 1 and 2,” December 7, 2011 (ML11339A126).

9. U.S. Nuclear Regulatory Commission, Interim Staff Guidance DC/COL-ISG-011, “Finalizing Licensing Basis Information,” November 2, 2009 (ML092890623).

10. Advisory Committee on Reactor Safeguards, “Report on the Safety Aspects of the Duke Energy Carolinas, LLC, Combined License Application for William States Lee III Nuclear Station, Units 1 and 2,” December 14, 2015 (ML15348A196).

11. Westinghouse Electric Company, “Westinghouse AP1000 Design Control Document Revision 19,” June 13, 2011 (ML11171A500).

12. IEEE Standard 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Stations,” June 27, 1991.