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

February 19, 2015

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE NUCLEAR INNOVATION NORTH AMERICA, LLC COMBINED LICENSE APPLICATION FOR SOUTH TEXAS PROJECT NUCLEAR STATION, UNITS 3 AND 4

Dear Chairman Burns:

During the 621st meeting of the Advisory Committee on Reactor Safeguards (ACRS), February 5-7, 2015, we reviewed the NRC staff's Advanced Safety Evaluation Report (ASER) for the Nuclear Innovation North America, LLC (NINA) Combined License Application (COLA) for South Texas Project (STP), Units 3 and 4. This application conforms to the design-centered review approach. The proposed STP Units 3 and 4 will be of the certified Advanced Boiling Water Reactor (ABWR) design, with certain departures.

We wrote three letter reports relating to STP Units 3 and 4: (1) “Interim Letter: Safety Evaluation Report with Open Items Related to the South Texas Project Combined License Application Referencing the Certified Advanced Boiling Water Reactor Design,” dated August 9, 2010; (2) “Report on the Safety Aspects of the South Texas Project Nuclear Operating Company Application to Amend the Certified U.S. ABWR Design to Incorporate the Aircraft Impact Assessment Rule,” dated September 10, 2010; and (3) “Long-Term Core Cooling for the South Texas Project Advanced Boiling Water Reactor Combined License Application,” dated November 7, 2012. The third letter responded to the Commission’s Staff Requirements Memorandum dated May 8, 2008 on the subject of long-term core cooling.

Our ABWR subcommittee held 22 meetings with the applicant and staff and reviewed the COLA and associated safety evaluation reports (SERs). During these meetings, we had the benefit of discussions with representatives of the NRC staff, the applicant, supporting vendors, and the public. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 52.53 that the ACRS report on those portions of the application that concern safety.

CONCLUSIONS AND RECOMMENDATIONS

1. There is reasonable assurance that STP Units 3 and 4 can be built and operated without undue risk to the health and safety of the public. The COLA for STP Units 3 and 4 should be approved following its final revision.

2. There is reasonable assurance that the ABWR design and the STP Units 3 and 4 site satisfy the requirements resulting from the Fukushima Near-Term Task Force recommendations.

3. We identify particular issues that the staff should address with the issuance of the STP Combined License:

    a. The final plant-specific turbine missile analyses should explicitly evaluate each turbine control and protection system including the turbine speed sensors, all component failure modes, all required support systems and the measured material toughness properties for the STP Units 3 and 4 monoblock rotors.

    b. Rather than imposing a requirement for weekly testing of turbine valves until the turbine missile analysis is submitted, the staff should incorporate a risk-informed analysis to determine the appropriate test frequency.

4. The Standard Review Plan (SRP) acceptance criteria regarding Charpy V-notch energy and fracture appearance transition temperature need to be updated to address differences between turbine rotors fabricated with shrunk-on discs versus monoblock rotors.

5. Fire hazard analyses have not thoroughly evaluated the possibility of fire-induced spurious actuations that may result from heat or fire damage to digital instrumentation and control signal cabinets, when external connections to those cabinets are made via fiber optic cables. Staff consideration of this as a generic issue would be prudent.

BACKGROUND

The ABWR design was certified by the NRC on May 12, 1997, with the design certification rule codified in 10 CFR Part 52, Appendix A. On September 20, 2007, the STP Nuclear Operating Company (STPNOC) submitted a COLA to the NRC for STP Units 3 and 4, in accordance with the requirements of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." In their application, STPNOC stated that STP Units 3 and 4 would be two ABWRs located adjacent to the site of two operating Westinghouse Pressurized Water Reactors (STP Units 1 and 2) in Matagorda County, Texas. This first COLA referencing the certified ABWR design is considered the "Reference COLA". The application provided information regarding departures taken from the certified ABWR design, and provided plant-specific and supplementary information as required by the Design Control Document (DCD). In 2008, STPNOC replaced General Electric with Toshiba as the alternate vendor for the certified ABWR design. Also, on January 24, 2011, NINA became the lead applicant for licensing and construction of STP Units 3 and 4.

DISCUSSION

<u>MAJOR DEPARTURES FROM THE ABWR DCD</u>

The STP Units 3 and 4 COLA identified certain departures from the certified ABWR design. Major departures include:

- Feedwater line break (FWLB) mitigation by initiating a trip of the condensate pumps following an indication of a FWLB in the drywell;
- Updating the safety-related instrumentation and control (I&C) architecture to state-of-the-art design;
- Addition of a fourth division of Class IE power supply to the safety-related I&C system among other updates;
- Revising the classification of the Radwaste Building substructure from Seismic Category I to non-seismic Category RWIIa;
- Use of a monoblock design for the Reactor Core Isolation Cooling (RCIC) system turbine and pump;
- Elimination of redundant hydrogen recombiners;
- Allowing the alignment of a third Residual Heat Removal (RHR) system loop for the augmented fuel pool cooling and makeup modes.

In addition, subsequent to certification, certain assumptions in the DCD containment analysis were found to be non-conservative. NINA took a departure from the DCD and updated the containment analysis for STP Units 3 and 4 for: (1) modeling of flow and enthalpy into the drywell for the FWLB analysis; (2) modeling of the drywell connecting vents for the FWLB and main steam line breaks; and (3) modeling of decay heat.

PLANT DESIGN FEATURES

Turbine Missile Analysis

The main turbine orientation at STP Units 3 and 4 is considered “unfavorable” for potential turbine missile damage to safety-related systems at the adjacent unit. Under these conditions, the SRP indicates that the frequency of unacceptable damage caused by turbine missiles should be less than 10^{-7} event per year for each unit. Following the SRP guidance, the applicant selected a generic value of 10^{-2} to account for the combined conditional probability of missile strikes on safety-significant equipment and damage from those strikes. We questioned the use of this value without any detail in the COLA regarding the design of shielding or the configuration of potential missile targets. We also noted that Toshiba Technical Reports UTLR-0008-P, “Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines” and UTLR-0009-P, “Probabilistic Evaluation of Turbine Valve Test Frequency” did not contain complete analyses of the protection and control systems for the STP turbines, or data that apply to the plant-specific design and components.

Our further review noted that certain common mode failures may affect the turbine normal speed controls and the emergency overspeed trip functions, generating a false input of zero turbine revolutions per minute. These failure modes may be particularly important during conditions when the primary overspeed trip system is offline. Under such situations, a turbine overspeed may need to be mitigated by operator action or by the power-load imbalance trip function.

Hence, we recommend that the final STP plant-specific turbine missile analyses should explicitly evaluate each turbine control and protection system including the turbine speed sensors, all component failure modes (including common cause failures), and all required support systems (e.g., AC or DC power supplies). The analyses should also include an evaluation and justification for the amount of time that the primary overspeed trip system may be out of service during turbine operation.

Although the staff considered the reported turbine missile analyses to be adequate for the combined operating license (COL) licensing review, the SER requires that a turbine system maintenance program be submitted within three years following receipt of the COL. This program would include an integrated analysis of turbine missile damage based on the as-built plant to show that the turbine meets the minimum requirements as given in Final Safety Analysis Report (FSAR) Table 3.5-1, "Requirement for the Probability of Missile Generation". Following the SRP guidance, the staff is also imposing a license condition to require weekly turbine valve testing and volumetric inspection of all low-pressure turbine rotors during alternate refueling outages until staff acceptance of the turbine maintenance program.

We question the safety merit of imposing a weekly turbine valve-testing interval against an increased risk of plant transients during that testing. The staff should perform or evaluate an integrated, risk-informed analysis to determine the appropriate turbine valve testing frequency, pending acceptance of the plant-specific turbine testing and maintenance program.

Monoblock Turbine Rotor

The use of a monoblock main turbine rotor introduced a material properties departure that the staff accepted regarding values of fracture appearance transition temperature (FATT) and Charpy V-notch energy at the minimum operating temperature, which are different from the SRP acceptance criteria. The material toughness measures specified in the STP COLA are indicative of lower toughness than the values specified in SRP Section 10.2.3. The applicant noted that these lower toughness acceptance values are based on material test data taken from deep-seated specimens (specimens taken from near the center of the forging) for monoblock rotors versus at the surface of a shrunk-on disc forging.

The staff found the values for FATT and Charpy V-notch energy included in the COLA to be acceptable for integral rotor forgings, since actual measured values for the STP Units 3 and 4 rotors will be used in the applicant's turbine missile probability analysis. This approach is consistent with industry practice (EPRI report ER-5619-SR, "Center Fracture Toughness of Monoblock Rotors"). The lower toughness values are expected to yield acceptable results because of lower stresses in the monoblock rotor design.

Our interim letter dated August 9, 2010, noted that the technical bases for acceptance of these departures were not well documented by the staff. The staff should consider an SRP revision to address the changing technology related to the acceptable values of FATT and Charpy V-notch energy for monoblock rotors. This affects not only new designs, but also replacement turbines for existing plants. The staff acknowledged this situation. Therefore, a definite plan to revise the SRP guidance should be developed.

Fire Damage to I&C Cabinets with only Fiber Optic Cables

The applicant has committed to update the STP Units 3 and 4 fire hazards analysis based on the as-built plant configuration. According to the analyses in the COLA, the effects from spurious actuations are not evaluated in locations that contain only fiber optic cables. Based on evidence from fire tests, we concur with the conclusion that fiber optic cables are not susceptible to fire damage that results in spurious actuations.

We asked whether the final fire hazards analyses will evaluate the effects from spurious actuations that may be caused by heat from a fire inside or nearby cabinets that contain digital signal processing circuitry, if the external connections to those cabinets are made via fiber optic cables. The applicant asserted that the likelihood of such an event is extremely small. The staff agreed with this conclusion and stated further that the design is acceptable because the redundancy resulting from separate divisions in separate fire areas would mitigate the effects from any spurious actuations. The staff conclusion is similar for other advanced nuclear power plant certified designs. However, the number and types of possible fire-induced spurious actuations will depend on the final circuit designs. The likelihood of those actuations will also depend on the as-built configuration and separation between different components and cabinets in each fire location.

Fire hazard analyses have not thoroughly evaluated the possibility of fire-induced spurious actuations that may result from heat or fire damage to digital instrumentation and control signal cabinets, when external connections to those cabinets are made via fiber optic cables. Staff consideration of this as a generic issue would be prudent.

Open Phase Events: Bulletin 2012-01

On July 27, 2012, the NRC issued Bulletin 2012-01, “Design Vulnerability in Electric Power System” to all holders of operating licenses and combined licenses for nuclear power reactors. This Bulletin was prompted by an event at Byron Station Unit 2 that involved the loss of one of the three phases of the offsite power circuit (single-phase open circuit condition). The Bulletin discusses the possibility that an open phase condition, with or without accompanying ground faults, located on the high-voltage side of a transformer connecting a General Design Criterion 17 offsite power circuit to the plant electrical system could result in a degraded condition in the onsite power system.

The applicant has stated that all three phases of the main power transformer and reserve auxiliary transformers will be monitored for an open phase and ground faults in any combination of one or more phases. This will initiate alarms in the Main Control Room when an open phase or ground fault is detected, and if required, operators will complete manual actions to address the alarms. Site-specific Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) will verify that detection and alarm components are installed as designed.

In addition, negative sequence voltage relays will be installed on the three Class 1E 4.16 kV busses. In the event of an unbalanced condition, the relays will open the power supply circuit breakers to the Class 1E 4.16 kV bus, protecting equipment on that bus and creating an under-voltage condition. The under-voltage signal will start the diesel generator before any of the Class 1E loads experience degraded conditions exceeding those for which the equipment is qualified. The relay setpoints and time delay values for relay actuation will be in accordance with the applicant's setpoint control program, and will be covered by the plant Technical Specifications. The staff reviewed and accepted the STP Units 3 and 4 design, Technical Specification changes, and the ITAAC proposed by the applicant. We concur.

SITE CHARACTERISTICS

Site characteristics include potential hazards in proximity of the plant, meteorology, hydrology, geology, seismology, and geotechnical parameters. An applicant must show that the actual site characteristics are bounded by the site parameters for the certified design, unless departures are justified by additional analyses.

Meteorology & Hydrology

STP Units 3 and 4 will use the main cooling reservoir (MCR) for non-safety-related normal plant cooling. The MCR is shared among STP Units 1, 2, 3, and 4. Makeup water to the MCR is supplied from the Colorado River and pumped into the MCR intermittently throughout the year. Each of STP Units 3 and 4 has a safety-related Ultimate Heat Sink (UHS), a Seismic Category I structure with an enclosed concrete flood-protected basin and a counter-flow mechanically induced draft cooling tower. The basin contains a 30-day supply of cooling water. The primary sources of makeup water to the UHS are site wells with the MCR as the backup source.

The applicant followed current regulatory guidance to determine the Probable Maximum Flood, the Probable Maximum Precipitation, and the Probable Maximum Water Surge, and as a consequence developed flood design considerations for the site. Both the design-basis flood (DBF) elevation and the design-basis precipitation exceed the ABWR DCD values. The site-specific DBF of 40 feet results from a breach of the MCR. To mitigate the potential effects of the DBF, the Category I structures are designed to be watertight up to 40 feet and with closed access doors (with controlled leakage) up to 51 feet, providing 11 feet of margin beyond the DBF. We reviewed the MCR breach flooding analysis and concur with the staff conclusions.

The applicant also considered the potential for flooding due to storm surge from the Gulf of Mexico and used high-resolution bathymetry to develop a model to calculate the probable maximum storm surge level, which was less than the flood level generated from the MCR breach. The staff reviewed the analysis and found it to be acceptable. We concur with the staff.

FUKUSHIMA REQUIREMENTS

In 2011, the NRC Near-Term Task Force (NTTF) issued a series of recommendations for improving nuclear power plant safety in the U.S. following the Fukushima earthquake and tsunami.

Seismic Reevaluations (2.1)

NTTF Recommendation 2.1 stated that plants should reevaluate the seismic hazards at their sites against current NRC requirements and guidance. The NRC issued a letter dated March 12, 2012, requesting that all operating nuclear power plants in the U.S. re-evaluate seismic hazards using the most recent information and methodologies available. The letter stated that nuclear power plant sites in the Central and Eastern U.S. should use the seismic source model in NUREG–2115, “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities,” to characterize their seismic hazards. Following the issuance of this letter to the operating nuclear power plants, the staff also requested all COL and Early Site Permit applicants to address this issue.

Consistent with NTTF Recommendation 2.1, the applicant evaluated the potential impact of the Central and Eastern United States Seismic Source Characterization (CEUS-SSC) model in NUREG-2115 on the characterization of the seismic hazard at the STP site. The applicant used the 1-, 10-, and 100-Hertz hard rock hazard curves for the nearby Houston Test Site because they concluded that both sites share similar geologic and tectonic settings, and similar activity rates. The applicant used the hard rock CEUS-SSC ground motion response spectra (GMRS) and then applied STP site-specific amplification factors to compare with the STP site-specific GMRS. The applicant concluded that the GMRS developed using the CEUS-SSC is very close to, and not significantly above, the site-specific GMRS in the STP COLA.

The staff performed a confirmatory probabilistic seismic hazard analysis for the STP site and the Houston Test Site, and compared the confirmatory 1-, 10-, and 100-Hz hazard curve results with the Houston Test Site results contained in the NUREG-2115 report. The staff performed a confirmatory site response calculation and used the resulting amplification functions along with CEUS-SSC hard rock hazard curves for the Houston Test Site to develop probabilistic hazard curves and a GMRS. The staff compared its CEUS-SSC GMRS to the STP Units 3 and 4 COLA FSAR GMRS for the entire 0.5- to 100-Hz frequency range and concluded that no revisions to the STP Units 3 and 4 COLA GMRS were necessary.

We agree with the staff assessment that the applicant’s design for safety against design-basis seismic events is adequate.

Mitigation Strategies for Beyond Design Basis Events (4.2)

STP Units 3 and 4 each have an installed combustion turbine generator (CTG) as an alternate AC power source for mitigation of a station blackout event. In addition, the STP Units 3 and 4 design can withstand a sustained loss of all AC power, including the loss of both CTGs, for 72 hours while maintaining core cooling. The Class 1E batteries have a full-load capacity of 8 hours, which can be extended beyond 36 hours if load-shedding procedures are performed.

For an extended loss of AC power, STP Units 3 and 4 plan to use the turbine-driven RCIC system to maintain core cooling for at least 36 hours using installed plant equipment. Additionally, the Alternating Current-Independent Water Addition (ACIWA) system is a seismically qualified system with an external permanent diesel-driven pump and a permanent piping connection to the UHS water supply. It is manually aligned to provide water to the RHR system for core and containment cooling without reliance on AC power and is to be used beyond 36 hours. Also, seismically qualified external connections on opposite sides of the Reactor Building can be used to provide makeup water and sprays for the spent fuel pool (SFP). Thus, STP Units 3 and 4 have the installed equipment to implement an extended coping time in excess of 72 hours without reliance on AC power for core and spent fuel cooling and for reactor coolant system and primary containment integrity.

In the STP Units 3 and 4 FLEX (Diverse and Flexible Coping Strategies) Integrated Plan, the applicant uses a two-phase strategy to provide and maintain core, containment, and spent fuel cooling for mitigating beyond-design-basis external events. Phase 1 is 36 hours in duration. Offsite supplies can be delivered to the site from the National Strategic Alliance for FLEX Emergency Response Center and become operational within 36 hours after the start of the event. There is no need for temporary portable Phase 2 equipment to provide core, containment, or spent fuel cooling, and thus, there will be a direct transition into Phase 3 at the end of Phase 1. The FLEX Integrated Plan does not take credit for the CTGs that are part of the STP design, even though it is believed that one, if not both CTGs, would survive and would likely be the first option for responding to such an event.

The FLEX Integrated Plan credits the installed RCIC, ACIWA, and Containment Overpressure Protection (COPS) systems to provide core, containment, and spent fuel cooling during Phase 1 and Phase 3. Advanced design features of the RCIC system, with local manual operation if needed, are credited to maintain RCIC operation for 36 hours. To prevent containment structural failure, the COPS rupture disk is designed to actuate at 90 psig containment pressure and vent the containment at approximately 20 hours into the event. The staff audit, using MELCOR analyses, concurred that there would be adequate core cooling during the first 36 hours after the onset of the event, and that COPS could function to relieve containment pressure and provide cooling, if needed. The audit supported the analysis that at approximately 20 hours, the COPS rupture disk would activate at 90 psig to relieve suppression pool pressure and allow cooling. After 36 hours, the reactor vessel is depressurized below 90 psig, thus allowing adequate ACIWA injection flow.

The staff reviewed the applicant's DC power load shedding calculations to extend the battery capacity beyond 36 hours to support the Phase 1 coping ability. The staff reviewed the battery duty cycle considerations and accepted the load shedding approach. The staff added several license conditions for adequate post-COL implementation of the overall integrated mitigation measures. Given these conditions, we concur that the STP Units 3 and 4 mitigation strategies plan is acceptable.

Spent Fuel Pool Instrumentation (7.1)

The staff evaluated the STP Units 3 and 4 proposed spent fuel pool level instrumentation with respect to NRC Order EA-12-051. SFP level instrumentation enhancements are consistent with guidance provided in NEI 12-02, Revision 1, “Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” and JLD-ISG-2012-03, “Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation.”

The SFP level instrumentation channels will be designed consistent with JLD-ISG-2012-03, and will be reliable at temperature, humidity, and radiation levels consistent with normal operation, and for accident conditions corresponding to a pool drain down to near the top of the spent fuel racks. The level instrumentation reliability will be established through inclusion in the Reliability Assurance Program. The ITAAC require verification that the SFP level instrumentation is installed properly and meets all design features as discussed in FSAR Appendix 1E, Section 1E.2.6. The applicant will develop operating procedures, testing, and calibration requirements for the installed instruments. We concur with the staff that these instruments are designed in accordance with the guidance in JLD-ISG-2012-03 and meet the Commission Order EA-12-051.

Enhanced Emergency Plan Staffing and Communication (9.3)

The Fukushima accident highlighted the need to better determine the levels of plant and offsite staffing needed to respond to a multi-unit event. Additionally, there is a need to ensure that communication equipment has adequate power to allow coordination of the response to an event during an extended loss of AC power. The applicant proposed four ITAAC related to enhanced communication and staffing. However, the staff determined that these items should be addressed as a license condition. The proposed license condition ensures that communications and staffing will be adequate for emergency planning operations. We concur with this approach.

SUMMARY

There is reasonable assurance that STP Units 3 and 4 can be built and operated without undue risk to the health and safety of the public. The COLA for STP Units 3 and 4 should be approved following its final revision.

Sincerely,

/RA/

John W. Stetkar
Chairman

REFERENCES

1. ACRS Letter, Subject: "Interim Letter: Safety Evaluation Report with Open Items Related to the South Texas project Combined License Application Referencing the Certified Advanced Boiling Water Reactor Design, dated August 9, 2010 (ML102000423)

2. EDO Letter, Subject: "Response to Advisory Committee on Reactor Safeguard Interim Letter: Safety Evaluation Report with Open Items Related to the South Texas project Combined License Application Referencing the Certified Advanced Boiling Water Reactor Design," dated September 10, 2010 (ML102440570)

3. ACRS Letter, Subject: "Long-Term Core Cooling for the South Texas Project Advanced Boiling Water Reactor Combined License Application," dated November 7, 2012 (ML12312A215)

4. ACRS Letter, Subject: "Report on the Safety Aspects of the South Texas Project Nuclear Operating Company Application to Amend the Certified U.S. ABWR Design to Incorporate the Aircraft Impact Assessment Rule," dated September 20, 2010 (ML102630190)

5. Regulatory Guide 1.143, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," dated November 2001 (ML013100305)

6. STP Units 3 and 4 COLA, Final Safety Analysis Report, Rev. 10 and 11 (ML13310B522 and ML14307B516 respectively).

7. Toshiba Technical Report UTLR-0008-P, Revision 1, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," September 2010 (ML102930100)

8. Toshiba Technical Report UTLR-0009-P, Revision 1, "Probabilistic Evaluation of Turbine Valve Test Frequency," September 2010 (ML102930101)

9. EPRI report ER-5619-SR , "Center Fracture Toughness of Monoblock Rotors," January 1988

10. NRC letter, Subject: "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ML12053A340)

11. NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," February 17, 2012 (ML12048A776)

12. STP 3&4 ABWR FLEX Integrated Plan, Revision 2, June 19, 2014 (ML14175A141)

13. NRC Order EA-12-051, “Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,” dated March 12, 2012 (ML12054A679)

14. NEI 12-02, Revision 1, “Industry Guidance for Compliance with NRC Order EA-12-051, “To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation”,” August 2012 (ML122400399)

15. JLD-ISG-2012-03, Revision 0, “Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation,” August 29, 2012 (ML12221A339)

List of NRC Safety Evaluation Reports Reviewed:

Chapter	Chapter Title	Transmittal Memo (Accessions Numbers)	ASER (Accession Numbers)
1	Introduction and Interfaces	ML110750239	ML110750306
2	Site Characteristics	ML12347A244	ML12299A111, ML12298A415
	Section 2.4 Non-Concurrence resolution	ML12347A244	ML12348A249
	Section 2.5 (Geology, Seismology, and Geotechnical Engineering)	ML14041A049	ML13360A133
3	Design of Structures, Components, Equipment and Systems	ML13295A330	ML13081A232
	Sections 3.7 and 3.8, Seismic Design and seismic Category 1 Structures	Ml13151A234	ML13129A138
4	Reactor	ML110340401	ML110340385
5	Reactor Coolant system and Connected Systems	ML110350192	ML110350197
6	Engineered safety Features	ML110310205	ML110310255
7	Instrumentation and Control Systems	ML103020012	ML103020088
8	Electric Power		ML102630174
	Section 8.2, Offsite Power System, Bulletin 2012-01 Response	ML14254A340	ML14219A688, ML14219A683
9	Auxiliary Systems	ML14267A502	ML14240A146
10	Steam and Power Conversion System	ML110620627	ML110620635
11	Radioactive Waste Management [1]	ML110340304	ML110340320
12	Radiation Protection [1]	ML111330693	ML111330697
13	Conduct of Operations [2]	ML110380331	ML110380349
14	Initial Test Program	ML110610503	ML110610549
15	Safety Analyses	ML103000455	ML103020103
16	Technical Specifications	ML110190249	ML110190259
17	Quality Assurance	ML110280198	ML110280134
18	Human Factors Engineering		ML102380198
19	Probabilistic Risk Assessment and Severe Accidents and Loss of Large Areas of the Plant due to Explosions or Fires	ML110800559	ML110800569
19, Att. A	Loss of Large Areas of the Plant Due to Explosions or Fires	ML111330333	ML111330352
22	Requirements Resulting from Fukushima Near-term Task Force Recommendations	ML14064A236 ML14219A701	ML14059A016

Notes:

1. NRO provided revisions to ASER for Chapters 11 (ML14016A390) and 12 (ML14013A213) (transmittal letter - ML14037A057), which after consideration at the P&P Session of the 616th ACRS meeting on July 11, 2014, the Committee decided no further review was necessary.
2. NRO provided revisions to ASER for Chapters 13 (ML13311A962-13.3 and 13.4S) (transmittal letter - ML13316B329), which after consideration at the P&P Session of the 611th ACRS meeting on February 7, 2014, the Committee decided no further review was necessary.

List of NRC Safety Evaluation Reports Reviewed:

Chapter	Chapter Title	Transmittal Memo (Accessions Numbers)	ASER (Accession Numbers)
1	Introduction and Interfaces	ML110750239	ML110750306
2	Site Characteristics	ML12347A244	ML12299A111, ML12298A415
	Section 2.4 Non-Concurrence resolution	ML12347A244	ML12348A249
	Section 2.5 (Geology, Seismology, and Geotechnical Engineering)	ML14041A049	ML13360A133
3	Design of Structures, Components, Equipment and Systems	ML13295A330	ML13081A232
	Sections 3.7 and 3.8, Seismic Design and seismic Category 1 Structures	Ml13151A234	ML13129A138
4	Reactor	ML110340401	ML110340385
5	Reactor Coolant system and Connected Systems	ML110350192	ML110350197
6	Engineered safety Features	ML110310205	ML110310255
7	Instrumentation and Control Systems	ML103020012	ML103020088
8	Electric Power		ML102630174
	Section 8.2, Offsite Power System, Bulletin 2012-01 Response	ML14254A340	ML14219A688, ML14219A683
9	Auxiliary Systems	ML14267A502	ML14240A146
10	Steam and Power Conversion System	ML110620627	ML110620635
11	Radioactive Waste Management [1]	ML110340304	ML110340320
12	Radiation Protection [1]	ML111330693	ML111330697
13	Conduct of Operations [2]	ML110380331	ML110380349
14	Initial Test Program	ML110610503	ML110610549
15	Safety Analyses	ML103000455	ML103020103
16	Technical Specifications	ML110190249	ML110190259
17	Quality Assurance	ML110280198	ML110280134
18	Human Factors Engineering		ML102380198
19	Probabilistic Risk Assessment and Severe Accidents and Loss of Large Areas of the Plant due to Explosions or Fires	ML110800559	ML110800569
19, Att. A	Loss of Large Areas of the Plant Due to Explosions or Fires	ML111330333	ML111330352
22	Requirements Resulting from Fukushima Near-term Task Force Recommendations	ML14064A236 ML14219A701	ML14059A016

Notes:

1. NRO provided revisions to ASER for Chapters 11 (ML14016A390) and 12 (ML14013A213) (transmittal letter - ML14037A057), which after consideration at the P&P Session of the 616th ACRS meeting on July 11, 2014, the Committee decided no further review was necessary.
2. NRO provided revisions to ASER for Chapters 13 (ML13311A962-13.3 and 13.4S) (transmittal letter - ML13316B329), which after consideration at the P&P Session of the 611th ACRS meeting on February 7, 2014, the Committee decided no further review was necessary.

Accession No: ML15039A006 **Publicly Available Y** **Sensitive N**
Viewing Rights: ☒ NRC Users or ☐ ACRS Only or ☐ See Restricted distribution

OFFICE	ACRS	SUNSI Review	ACRS	ACRS	ACRS
NAME	MBanerjee	MBanerjee	MBanks	EMHackett	EMH for JWS
DATE	02/18/15	02/18/15	02/19/15	02/19/15	02/19/15

OFFICIAL RECORD COPY