United States Nuclear Regulatory Commission Official Hearing Exhibit
In the Matter of: NUCLEAR INNOVATION NORTH AMERICA LLC
(South Texas Project Units 3 and 4)
Commission Mandatory Hearing
Docket #: 05200012 & 05200013
Exhibit #: STP-001-MA-CM01
Identified: 11/19/2015
Admitted: 11/19/2015
Withdrawn:
Rejected:
Stricken:
Other:

Exhibit STP-001

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	Docket Nos. 52-012 COL
	52-013 COL
NUCLEAR INNOVATION NORTH AMERICA LLC	
(South Texas Project Units 3 and 4)	November 12, 2015

APPLICANTS' RESPONSES TO COMMISSION'S PRE-HEARING QUESTIONS

Nuclear Innovative North America LLC (NINA) provides the following responses to the questions in the Commission's October 16, 2015 Order (Transmitting Pre-Hearing Questions) regarding the mandatory hearing for South Texas Project (STP) Units 3 and 4. NINA's responses are limited to those questions directed to it.

Responses to Commission Questions

Question 2: In order for the NRC to certify a design, the design must be "essentially complete." 10 C.F.R. § 52.41. When promulgating Part 52, the Commission stated that the phrase "essentially complete nuclear power plant . . . is defined as a design which includes all structures, systems, and components which can affect safe operation of the plant except for site-specific elements . . . ." Final Rule: Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors, 54 Fed. Reg. 15,372, 15,382 (Apr. 18, 1989). NINA is proposing to take over 300 departures from the certified ABWR [Advanced Boiling Water Reactor] design. Given the large number of departures, is this application still referencing an essentially complete certified design, or should the design being referenced in the combined license (COL) application be considered a unique design?

NINA Response: The provisions in 10 C.F.R. § 52.41 apply to applicants for a design certification; they do not apply to applicants for a COL. As provided in the Statement of

Considerations for Part 52, 54 Fed. Reg. 15372, 15382 (April 18, 1989), the term "essentially complete" means:

- "a design which includes all structures, systems, and components which can affect safe operation of the plant except for site-specific elements such as the service water intake structure and the ultimate heat sink;" and

- "a design that has been finalized to the point that procurement specifications and construction and installation specifications can be completed and made available for audit if it is determined that they are required for Commission review in accordance with the requirements of § 52.47(a)."

Both the ABWR design certification application and the STP COL application (COLA) satisfy this definition, and therefore both are "essentially complete."

Furthermore, neither the number nor the nature of departures for STP Units 3 and 4 affect the underlying applicability of the ABWR design certification to STP. STP Units 3 and 4 have the same structures, systems, and components (SSCs) as specified in the ABWR Design Control Document (DCD), except for the elimination of the new fuel storage racks and hydrogen recombiners (both of which do not have any significant impact on safety). Furthermore, the STP SSCs perform the same functions as specified in the DCD. In general, the departures pertain to the details in Tier 2 of the DCD, as indicated by the fact that only 17 of the departures pertain to Tier 1 of the DCD, and the net effect on the standardization of the design is minimal. Therefore, it is appropriate for the COLA for STP Units 3 and 4 to reference the ABWR design certification.

Question 3: Condition 2.d.(12)(d) in the draft COL addresses the transportation physical security plan. Paragraph 2 of that condition requires NINA to update FSAR § 13.6.4, in its first FSAR [Final Safety Analysis Report] update after issuance of the license.

If the required information can be provided with the first FSAR update, why can it not be provided now?

Further, the license condition states that the FSAR update should include requirements to meet 10 C.F.R. § 74.15, but does not provide specifics on how the licensee must meet those requirements.

Explain how this license condition meets the Commission's requirement that conditions be "precisely drawn so that the verification of compliance becomes a largely ministerial act." *Private Fuel Storage, LLC* (Independent Spent Fuel Storage Installation), CLI-00-12, 52 NRC 23, 34 (2000).

<u>NINA Response</u>: NINA has provided the NRC with the requested information in a letter dated July 7, 2015, available at ADAMS accession number ML15194A054. That letter includes the following information:

> As stated in the reference, Nuclear Innovation North America (NINA) will have written agreements in place with the fuel manufacturer to transport the fuel to the STP 3&4 site under the manufacturer's Transportation Physical Security Plan. The fuel manufacturer must be licensed under Part 70, and the carrier will have a general license pursuant to 10CFR70.20b. The written agreement will require the manufacturer to be licensed, to have a Transportation Physical Security Plan, and to arrange for in-transit physical protection. These arrangements will satisfy the requirements of 10CFR73.67(g)(2)(iii) for NINA.
>
> NINA will have in place receipt inspection procedures to receive the fuel. The fuel will not change ownership until the successful completion of the receipt inspection. As part of these procedures, NINA will include requirements to inspect the integrity of the containers and the tamper seals upon receipt of the shipment. Procedural steps will also require NINA to notify the shipper of receipt of the material as required in 10CFR74.15.

Thus, the docket already includes sufficient information for the NRC to issue the COLs for STP Units 3 and 4. Updating of the FSAR will simply be a ministerial action to incorporate the information quoted above. The FSAR has not been updated at this time because the NRC has the requested information, and the annual update to the FSAR is not yet due per 10 C.F.R. § 50.71(e)(3)(iii).

<u>Question 4</u>: License condition 2.D.(14)(i) regarding Cyber Security states that "8 months before fuel is allowed onsite (within the protected area) NINA shall develop a written protective strategy . . ." to meet 10 C.F.R. § 73.54.

Please describe whether the Staff has found that all of the cyber security requirements necessary for licensing are met at this time.

Further, please address how this condition meets the Commission's requirement that conditions be "precisely drawn so that the verification of compliance becomes a largely ministerial act." *Private Fuel Storage, LLC* (Independent Spent Fuel Storage Installation), CLI-00-12, 52 NRC 23, 34 (2000).

NINA Response: NINA's COLA included a Cyber Security Plan (CSP) in the Safeguards portion of Part 8 of the COLA. As stated in the Staff's Final Safety Evaluation Report (FSER) Section 13.8.6:

> …the information in the applicant's CSP adequately addresses the relevant requirements and guidance of 10 CFR 73.54 and RG 5.71, respectively. The staff also determined that the CSP includes all features considered essential to a cyber security program. In particular, the staff determined that the CSP complies with the applicable Commission regulations including 10 CFR 73.1; 10 CFR 73.54; 10 CFR 73.55(a)(1); 10 CFR 73.55(b)(8); 10 CFR 73.55(m); and 10 CFR Part 73, Appendix G. Therefore, the staff determined the information in the STP CSP to be acceptable.

As stated in the FSER, the Staff has found that all of the cyber security requirements necessary for licensing are met at this time.

Since NINA has included a Cyber Security Plan as part of the station Security Plans, the license condition can be satisfied by establishing that the plan exists and is implemented. This condition is similar to the license conditions applicable to other operational program descriptions, and is ministerial in nature.

Question 5: NINA would be the licensee responsible for design and construction of STP Units 3 and 4. STPNOC [STP Nuclear Operating Company] will be the operator and license holder for STP Units 3 and 4 upon issuance of the 10 C.F.R. § 52.103(g) finding or authorization for interim operation pursuant to 10 C.F.R. § 52.103(c).

Describe how the transition to operation will take place. Will there be duplicate programs running in parallel, such as the corrective action program?

NINA Response: During the construction phase prior to the 52.103(g) finding, STPNOC, acting under NINA's direction, will develop and implement, as appropriate, the programs, procedures, staffing, training, etc. necessary to operate the plant. As systems are completed and turned over from the engineering, procurement and construction (EPC) contractor, they will be controlled, operated and maintained by STPNOC. By the time the requirements of 52.103(g) are met, STPNOC will have all necessary processes, programs, and staff in place and will have accepted control of all necessary systems and components.

In the case of the corrective action program, there will be a construction corrective action program maintained by the EPC contractor that will be designed and used for construction activities, and STPNOC will manage operational issues in its existing program. Applicable data from the construction corrective action program will be turned over to STPNOC.

Question 6: The Staff's findings in § 1.4S of the Safety Evaluation Report (SER) regarding the alternate vendor qualifications requirements in 10 C.F.R. § 52.73 appear to rely in large part on findings from a 2009 inspection report.

Please describe in more detail what information, if any, is included in licensing basis documents to assure the applicant's continued ability to meet the requirements in 10 C.F.R. § 52.73 and Part 52, Appendix A.III.

NINA Response: Part 52, Appendix A.III requires COL applicants that reference the ABWR design certification to incorporate the ABWR DCD by reference in the COLA, and does not pertain to alternate vendors.

The requirements in 10 C.F.R. § 52.73 apply to applications for a COL. After issuance of a COL, 10 C.F.R. § 52.73 does not impose a continuing obligation to demonstrate the alternate vendor's qualifications to supply the standard design, and there is no regulation that requires that such information be included in licensing basis documents (e.g., the FSAR).

In that regard, Toshiba's qualifications to supply the ABWR design are largely based on the fact that it previously has been a prime contractor and supplier of the ABWR, and it possesses the design basis documents and proprietary documents referenced in the DCD (or has been able to recreate the documentation). Those facts will not change after issuance of the COLs for STP Units 3 and 4 (any more than they would change if General Electric had been selected as the vendor for STP Units 3 and 4).

<u>Question 7</u>: The FSAR provides that both Toshiba Power Systems Company and Shaw Group Incorporated, as part of the Consortium, in conjunction with subcontractors, are responsible for the Engineering, Procurement, and Construction (EPC) of STP Units 3 & 4.

a. For NINA: Both Toshiba and Shaw are identified as responsible for the EPC. Please clarify.

<u>NINA Response</u>: The EPC Contract is between STPNOC, as agent for NINA Texas 3 LLC, and NINA Texas 4 LLC, and an unincorporated consortium formed by Toshiba America Nuclear Energy Corporation (TANE), and Stone and Webster Inc. (a subsidiary of Shaw). Thus, it is the consortium that is responsible for implementing the EPC contract.

<u>Question 9</u>: Section 1.5S of the SER discusses the proposed exemption from the financial qualifications requirements in 10 C.F.R. Part 50. As directed by the Commission in the Staff Requirements Memorandum for SECY-13-0124, the Staff is anticipating the outcome of the forthcoming financial qualifications rulemaking as part of the basis for granting the exemption. In license condition 2.D.(14)(K)(1) in the draft combined license, paragraph (iii) states that "this documentation will include operative closing documents, and **may** include documented proof of parent and affiliate assurances... ."

What is the purpose of the "may" clause? Is it meant to convey permission from the NRC on the types of information allowed, or is it meant as an example of the type of information that may be submitted?

<u>NINA Response</u>: The "may" clause is meant to provide an example of the type of information that may be submitted. In that regard, Appendix C.II.A.2 to 10 C.F.R. Part

50 has long identified parental or affiliate assurances as an example of a source of construction funds.

Question 10: In promulgating 10 C.F.R. § 50.44, the Commission stated that paragraph (c) of the final rule sets forth combustible gas control requirements for all future water-cooled nuclear power reactor designs and "these requirements reflect the Commission's expectation that future designs will achieve a higher standard of severe accident performance." 50 Fed. Reg. 32,138 (Aug. 8, 1985). The Staff proposes elimination of the hydrogen recombiners requirements in STD DEP T1 2.14-1, finding special circumstances are present as described in 10 C.F.R. § 50.12(a)(2)(ii), which states that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

Given this departure proposed from the Standard ABWR Design, how does the South Texas Project Units 3 and 4 combined license application meet the Commission's statement quoted above when issuing 10 C.F.R. § 50.44 that future designs will achieve a higher standard of severe accident performance?

NINA Response: As the quoted sentence from the Policy Statement indicates, the expectation for higher standards of performance pertains to severe accident safety performance in a holistic sense, not to the performance of any particular system. As provided in the Policy Statement:

> The inherent flexibility of this Policy Statement (that permits risk-risk tradeoffs in systems and sub-systems design) encourages thereby innovative ways of achieving an improved overall systems reliability at a reasonable cost.

Even with the departure from the combustible gas control requirements in the ABWR DCD, STP Units 3 and 4 still provides a substantially higher standard of severe accident performance than existing operating plants, as provided by SSCs such as the AC-independent water addition (ACIWA) system, combustion turbine generator, lower drywell flooder (LDF), Containment Overpressure Protection System (COPS), and basaltic concrete used to limit production of non-condensable gases from a core-on-the-floor event. See, e.g., NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling-Water Reactor Design," pg. 19-5.

In that regard, the hydrogen recombiners were not identified as a significant contributor to severe accident safety performance for the ABWR. Instead, as discussed on page 19-37 of the ABWR FSER, containment inerting in the ABWR provides the primary protection against hydrogen generation during a severe accident. As stated in the FSER:

> Hydrogen combustion is not an important containment challenge in the ABWR since the atmosphere is made inert during normal operation.

In summary, STP Units 3 and 4 provide a higher standard of severe accident performance and therefore satisfy the Commission's Policy Statement.

Question 11: Tier 1 departure 2.14-1 and the associated exemption request proposes to eliminate the flammability control system from the ABWR certified design. The SER states that the departure and exemption are justified because of changes to 10 C.F.R. § 50.44 that occurred after certification of the ABWR. However, § 50.44(c) states that it is applicable to design certifications issued after October 16, 2003. The ABWR was certified in 1997.

Please provide further explanation for why changes to § 50.44 justify the elimination of the flammability control system. Further, if the applicant is now relying on the current version of § 50.44, please discuss whether the combined license application must also meet the requirements of § 50.44(c)(1) and (5).

NINA Response: Changes to 10 C.F.R. § 50.44 in 2003 were the result of studies that improved the understanding of combustible gas behavior during a design basis Loss of Coolant Accident (LOCA) and during a beyond design basis (i.e., severe) accident. These studies confirmed that the hydrogen release postulated from a design basis LOCA was not large enough to lead to early containment failure and that the risk from hydrogen combustion occurred during severe accidents. Thus, elimination of hydrogen recombiners, which had previously been required by 10 C.F.R. § 50.44 to mitigate hydrogen produced during a design basis LOCA, would not result in a significant decrease in the level of safety.

The requirement for hydrogen recombiners in the version of 10 C.F.R. § 50.44 in effect in 1997 was applicable to the certified ABWR design while the current version of 10 C.F.R. § 50.44 is applicable to the STP COLA because the COLA was filed after 2003. STP had to either apply the finality provided by the certified design and conform to the 1997 version of 10 C.F.R. § 50.44 or apply for a departure from the certified design and apply the current rule.

As a result, a departure from Tier 1 of the ABWR DCD is included in the COLA that conforms to the current version of 10 C.F.R. § 50.44. As part of the justification for the departure, STP evaluated the departure against the five criteria in 10 C.F.R. § 50.44(c) and demonstrated that each of the criteria was satisfied or not applicable to the ABWR. See COLA Part 7, Section 2.1, STD DEP T1 2.14-1.

Question 12: Departure STD DEP T1.2.14-1 in the combined license application removes the Flammability Control System, which was part of the original ABWR design. The stated basis for removal of the system is that it is no longer required by the revision of 10 C.F.R. § 50.44 that occurred after the ABWR design was approved. Section 50.44 was revised because inerted containments provide protection from hydrogen combustion. The Fukushima event showed, however, that hydrogen combustion events can occur outside of the inerted primary containment and cause significant damage to the secondary containment building. Was the possible benefit of the Flammability Control System in the context of severe accident mitigation and recovery considered with respect to removal of the System for the STP combined license application?

NINA Response: The NINA review of the Fukushima event confirms that the Flammability Control System (FCS) removed from the primary containment in the ABWR design would not prevent hydrogen combustion in the secondary containment.

The FCS, which is part of the ABWR certified design, was eliminated from the STP Units 3 and 4 design consistent with 10 C.F.R. § 50.44, as amended in 2003, using guidance provided in Regulatory Guide (RG) 1.7, Control of Combustible Gas Concentrations in Containment, Revision 3. Technical justification for the revision to 10

C.F.R. § 50.44 in 2003 is described in the Statement of Considerations (SOC) for 10 C.F.R. § 50.44 in 68 Fed. Reg. 54123 (Sept. 16, 2003). As stated in the SOC, the purpose of the FCS as provided in the pre-2003 version of the rule was to address hydrogen generated during a LOCA. The FCS, as provided in that rule, was not intended to address severe accidents (which instead are addressed by inerting the containment).

Because the AC-powered FCS recombiners in the certified ABWR design are designed to recombine the long-term, relatively small amounts of oxygen and hydrogen due to radiolysis in a design basis LOCA, they would provide no benefit in a Fukushima-like severe accident.

Question 14: The acceptance criterion in RG 1.91 states that safety-related concrete structures are considered safe if the air overpressure from an explosion is below 1 pound per square inch (psi). The calculated hydrogen explosion air overpressure shown at the nearest safety-related systems, structures, and components (SSCs) in FSAR Table 2.2S-10, Revision 12 is 0.987 psi. Because this value is so close to 1 psi, did the Staff or NINA perform any additional analysis to demonstrate the safety of safety-related SSCs? If so, what were the results?

NINA Response: STP Units 3 and 4 FSAR Section 2.2S.3.1.1, Explosions, summarizes the approach used and the conservatisms assumed in the calculation of overpressure at safety-related concrete structures resulting from the explosion of hydrogen stored on or near the STP site. The calculations and assumptions regarding allowable distances are in accordance with RG 1.91, Evaluations of Explosions Postulated to Occur at Nearby Facilities and On Transportation Routes Near Nuclear Power Plants, Revision 1. Additional conservatisms assumed in the calculations are summarized in FSAR Section 2.2S.3.1.1 and 2.2S.3.1.2.

Given the conservative assumptions, the conservative acceptance criterion of 1 psi, and the fact that the acceptance criterion is satisfied, further analyses are not necessary and were not performed by NINA.

Question 15: a. For the Staff and NINA: Does the annual tornado strike frequency (or, recurrence interval) of 1.75E-04 provided in SER Section 2.3S.1.4.3.2 correspond to the strike frequency for one of the units, both of the units combined, or individual structures within a unit? If it is for an individual structure, what is the annual tornado strike frequency for each unit?
b. For the Staff and NINA: It appears NINA used the characteristic dimension of 200 feet (ft) from NUREG/CR-4461, Rev. 2 when calculating annual tornado strike frequency. But some structures for STP Units 3 and 4 appear to have dimensions greater than 200 ft (e.g., Figure 1, RAI [Request for Additional Information] Response 03.05.01.06-1). Why is it reasonable to use the 200 ft characteristic dimension provided in NUREG/CR-4461?

NINA Response: NINA provided tornado occurrence data in the COLA (RAI 02.03.01-2 and FSAR Section 2.3S.1.3.2). The expected strike frequency using the 200 ft dimension comes from an analysis performed by the NRC Staff based on the occurrence data and is not in the FSAR.

STP Units 3 and 4 tornado design basis is in accordance with the characteristics in RG 1.76 for the STP site. The FSAR uses NUREG/CR-4461, Rev. 2, to determine the tornado intensity region applicable to the STP site, not the strike frequency.

Question 16: 10 C.F.R. § 100.23(c) requires NINA to investigate the geological and seismological characteristics of the STP site and its environs to support estimates of the Safe Shutdown Earthquake Ground Motion, among other things. Hydrocarbon extraction in the Eagle Ford Shale in South Texas may now be more extensive than it was at the time the FSAR and SER were developed.

For NINA: Does the recent extraction of hydrocarbons in the Eagle Ford Shale change NINA's response to RAI 0.2.05.01-14 (August 27, 2008) related to the potential for future subsidence due to human activities and effects from these activities?

NINA Response: The nearest approach of the Eagle Ford extraction area is approximately 70 miles from the STP site. Given that distance, extraction of hydrocarbons from that area will have no impact on the potential for subsidence at the

STP site. In addition, given the nature of the Eagle Ford extraction regime it is very unlikely that distance would change much in the future. Therefore, the conclusions NINA reached in the subject RAI response remain valid.

<u>Question 18</u>: 10 C.F.R. § 100.23(d)(4) requires consideration of liquefaction potential of the STP site. Section 2.5S.4.8 of the FSAR presents NINA's evaluation of liquefaction potential at the STP site. In response to RAI 02.05.04-28 and as documented in Section 2.5S4.4.8 of the SER, NINA evaluated seismic-induced settlements for sandy soils using a procedure developed by Ishihara and Yoshimine (1992). NINA applied this method to soils with a factor of safety against liquefaction less than 1.4. According to Ishihara and Yoshimine (1992), seismic-induced volumetric strains are expected even when the factor of safety exceeds 1.4.

a. Why was volumetric strain evaluated for only soil layers with a factor of safety less than 1.4?

b. Would seismic-induced volumetric strains at the site be anticipated to propagate to the foundation elevation and affect the performance of safety related structures, systems, and components when considering soil layers having a factor of safety against liquefaction greater than 1.4?

c. The method developed by Ishihara and Yoshimine (1992) for evaluating seismic-induced settlement is for free-field conditions (no structure being supported by the soil). What effect (if any) will structures have on estimated seismic compression?

<u>NINA Response</u>:

a. Small seismic-induced volumetric strains above a factor of safety (FOS) of 1.4 have been recognized for some time. Ishihara and Yoshimine (1992) states that "…it was considered desirable to set a critical factor of safety at which the post-liquefaction volume change becomes essentially zero." Figure 10 of the cited reference for Ishihara and Yoshimine (1992) shows that seismic-induced volumetric strains for soils with calculated liquefaction factor of safety above 1.4 are very small, on the order of ¼ of 1%. Volumetric strains are essentially zero at a factor of safety of 2.

FSAR Section 2.5S.4.11, Design Criteria, notes that Regulatory Guide 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant

Sites," indicates that FOS < 1.10 is generally considered a trigger value, FOS≈1.10 to 1.40 is considered intermediate, and FOS≥1.40 is considered high.

FSAR Section 2.5S.4.11 also states:

> As used in Subsection 2.5S.4.8, an FOS of 1.10 was considered a threshold value to evaluate the potential effects of liquefaction of site soils. On this same issue, the Committee on Earthquake Engineering of the National Research Council (Reference 2.5S.4-53) stated that "There is no general agreement on the appropriate margin (factor) of safety, primarily because the degree of conservatism thought desirable at this point depends upon the extent of the conservatism already introduced in assigning the design earthquake. If the design earthquake ground motion is regarded as reasonable, a safety factor of 1.33 to 1.35 […] is suggested as adequate. However, when the design ground motion is excessively conservative, engineers are content with a safety factor only slightly in excess of unity." This position, and the FOS < 1.10 trigger value from Reference 2.5S.4-52, is consistent with the value selected for the analyses of STP 3 & 4 site soils, also considering the conservatism employed in ignoring over consolidation, the geologic age of the deposits, and other factors noted above.

Consequently, we believe that the issue of sandy soils with calculated factors of safety against liquefaction between 1.4 and 2.0 in the settlement calculations has been adequately addressed.

The NRC Staff performed a similar assessment in the FSER for STP Units 3 and 4 and reached similar conclusions.

b. As stated in the response to part a, volumetric strains associated with high factors of safety against liquefaction are very small and are typically not considered to be a hazard. The more significant volumetric strains associated with factors of safety less than 1.4 have been evaluated and found acceptable. Consequently, the much smaller volumetric strains are not expected to affect the performance of SSCs at the foundation elevation.

c. The zones that could experience seismic-induced settlement are located at significant depth (typically 170 to 320 ft below the foundation elevation of safety-related structures), thus the seismic-induced settlement can be estimated from the results of one-dimensional analysis of the free-field condition (without considering the structure being supported by the soil).

<u>Question 19</u>: 10 C.F.R. § 100.23(d)(4) requires consideration of the liquefaction potential of the STP site. In FSAR Section 2.5S.4.8.2.2, NINA describes the use of the "Chinese Method" to evaluate liquefaction potential of clayey soils. Peer-reviewed literature (e.g., Boulanger and Idriss 2004, Bray and Sancio 2004) states that the "Chinese Method" should no longer be used to evaluate liquefaction potential of clayey soils. In light of this peer-reviewed research, explain your conclusion that there is no liquefaction potential for the clayey strata at the STP site.

<u>NINA Response</u>: NINA used the methodology cited in the current approved version of RG 1.198, the "Chinese Method", and determined that clayey soils at the STP site are not vulnerable to liquefaction. Recent analytical approaches focus on the use of the Plasticity Index to assess the potential for liquefaction in clayey soils. NINA reviewed the data presented in the FSAR regarding the Plasticity Index of clayey soil at the STP site and confirmed that our original conclusion remains valid.

<u>Question 20</u>: Confirm that the actual thicknesses provided for the ultimate heat sink basin, the cooling tower enclosures, and the reactor service water pump house walls and slabs acting as missile barriers exceed the thicknesses calculated by both the National Defense Research Council formula, specified in NUREG–0800 [Standard Review Plan (SRP)] Section 3.5.3, and the TM 5–855–1 formula, used by NINA, and are protective against penetration and perforation as well as scabbing. The SER text only discusses scabbing.

<u>NINA Response</u>: As explained in Section 3.3.2 of the FSER, the penetration, perforation, and scabbing thicknesses of the Ultimate Heat Sink (UHS), the Cooling Tower Enclosures, and the Reactor Service Water (RSW) Pump House walls and slabs acting as missile barriers were determined using the U.S. Department of Army TM 5-855-1 formula in "Fundamentals of Protective Design for Conventional Weapons", November 1986.

Since this method is not referenced in SRP Section 3.5.3, the Staff issued RAI 03.03.02-3, requesting the applicant to discuss the validity of this method in light of the corresponding SRP acceptance criteria and to justify any deviations. In response to this RAI it was determined that in comparison to the National Defense Research Council (NDRC) formula specified in SRP Section 3.5.3, the TM 5-855-1 formula predicts a higher penetration but a lower required thickness to prevent scabbing. However, actual thicknesses provided for the UHS basin, cooling tower enclosures, and RSW Pump House walls and slabs acting as tornado or hurricane missile barriers exceed those required to prevent penetration and perforation, as well as scabbing, using either the NDRC formula or the TM 5-855-1 formula.

Question 21: Regulatory Guide (RG) 1.206, § C.I.2.2.2.7 provides that the applicant should consider hazards from aircraft from nearby airports and aviation routes and § C.I.2.2.2.8 provides that the applicant should provide projections of the growth of these activities. Has a projection of aircraft traffic over the life of the facility been considered in the aircraft hazard evaluation? Discuss the results of that assessment.

NINA Response: NINA considered the Federal Aviation Administration (FAA) projections of total operations at nearby facilities as they related to possible growth activities in the aircraft hazard evaluation. An explanation concerning the use of the FAA projection data, including the availability of the projections from the FAA along with how the projections were used in the aircraft hazard evaluation was presented in the response to RAI 03.05.01.06-1. As detailed in the response, the total number of operations at nearby facilities, Palacios Municipal Airport and Scholes International Airport at Galveston, is based upon available FAA projections for the year 2025. The FAA did not project further than the year 2025 at the time of the COLA.

The selection of the projected 2025 FAA data is justified, with respect to accounting for possible growth at the evaluated facilities, based on the data in the Terminal Forecast Detail Report, Forecast Issued December 2008. The FAA projections during the forecast period (2008-2025) included in this report for the Palacios Municipal Airport and the Scholes International Airport at Galveston revealed:

- The projected number of operations for Palacios Municipal Airport remains steady during the reflected projection forecast period (2008-2025) after a recorded decline in the number of operations from the year 2004 to 2005. Accordingly, any extrapolation of FAA projections may yield a lower number of operations than used in the aircraft accident evaluation.
- The total number of operations for Scholes International Airport at Galveston decreased by over 40,000 from the year 2005 to 2006. The FAA projected that the number of operations would continue to decrease during the first year (2008) of the reflected projection forecast period (2008-2025) and then increase slightly over the remaining forecast period, recovering only about 10,000 of the approximate 40,000 decrease in the total number of operations.

An additional evaluation further justified the use of the FAA 2025 projected data. The evaluation revealed that the number of commercial operations in the vicinity of the plant would need to exceed 155,480 (which would be more than double the projected FAA 2025 figures at both airports) to exceed the order of magnitude of 10^{-07} threshold in SRP Section 3.5.1.6.

In addition, the current FAA projections, contained in the most recent Terminal Area Forecast Detail Report, Forecast Issued January 2015 was reviewed to further substantiate the assessment of the potential growth at the nearby facilities. The FAA January 2015 projections demonstrate:

- The projected number of operations for Palacios Municipal Airport continue to remain steady at the former 2025 projected number of operations, 2960, in the Forecast Report Issued December 2008, through the year 2040. Accordingly, any interpolation of data out farther than 2040 would remain steady at the 2025 projected number of operations used in the evaluation.
- The actual total number of operations for Scholes International Airport at Galveston in 2009 was less than half the number of operations initially projected for the year 2009. The FAA continues to project a slight increase in the number of operations during the extended forecast period in the Forecast Report Issued January 2015; however, the new forecast shows the new projected total number of operations for the year 2025 is 36,237 versus the initial projection of 70,342 in the Forecast Issued December 2008. Accordingly, the total number of operations used in the aircraft accident evaluation continues to remains conservative.

Question 22: In performing the original site-specific soil structure interaction (SSI) and structure-soil-structure interaction (SSSI) analyses of embedded structures, NINA used the System for Analysis of Soil Structure Interaction (SASSI) Subtraction Method (SM) of analysis. In its letter to the Department of Energy dated April 8, 2011, the Defense Nuclear Facilities Safety Board identified a technical issue in SASSI that when the SM is used to analyze embedded structures, the results may be nonconservative. In RAI 03.07.01-29 the Staff requested that NINA demonstrate the acceptability of the SM and the results or to provide a plan and schedule to ensure that the structures, systems, and components (SSCs) are designed to meet General Design Criterion (GDC) 2 requirements. Describe the process followed to address the use of SM for site-specific SSI and SSSI analyses.

NINA Response: In order to address the issue identified by the Defense Nuclear Facilities Safety Board (DNFSB) regarding the use of the Subtraction Method (SM) in

SASSI to analyze embedded structures, the plan/process shown in Table 03.07.01-29 S1.2 provided in Revision 1 of Supplement 1 response to RAI 03.07.01-29 (ML113360516) was used. In accordance with this plan, extensive evaluations were performed and, where required, in-structure response spectra and/or structural designs based on SM were modified to ensure STP Units 3 and 4 designs are conservative. These evaluations took into account the recommendations for reviewing past SASSI SM analyses, and advice on avoiding SM errors in future analyses that DOE provided in a letter from Daniel B. Poneman to Peter S. Winokur dated July 29, 2011, responding to the DNFSB. These evaluations are described in detail in FSAR Section 3H.10. Staff review of these evaluations is described in Section 3.7.2.4.20 of the FSER.

Question 23: SER Section 3.7.2.4.19 discusses the Staff's assessment of a 10 C.F.R. Part 21 evaluation performed by Fluor Enterprises, Inc. By letter dated August 30, 2010, Fluor notified the NRC about an exceedance of the ABWR DCD seismic design input requirements for the main steam line seismic analysis of the turbine building for STP Units 3 and 4. Why was it not necessary for NINA to take a departure from the standard design in light of the exceedance addressed by Fluor?

NINA Response: No departure is required because the Turbine Building design is required to comply with the ABWR DCD seismic design input requirements, and site-specific inspections, tests, analyses and acceptance criteria (ITAAC) will confirm that the final design is consistent with the ABWR DCD requirements. As was noted in our response to RAI 03.07.02-30 the design of the STP Units 3 and 4 Turbine Building was in progress at the time that Fluor identified the issue. The analysis performed by Fluor was based on design work in progress and neither the Turbine Building design used by Fluor nor the seismic analysis performed by them was accepted by NINA.

Question 24: SER Section 3.8.4.4.1 discusses the Staff's evaluation of lateral seismic earth pressures on below-grade external walls. The site-specific pressures for the Reactor Building and the Control Building exceed the corresponding pressures considered in the standard design.

a. What are the implications of such exceedance and the applicability of the standard design to the STP Units 3 and 4 site in this regard?

b. Why was it not necessary for NINA to take a departure from the standard design for the site-specific lateral seismic earth pressures?

NINA Response:

a. As explained in FSAR Section 3H.2.6, the soil pressure profile from the site-specific SSSI analysis for the Control Building is bounded by the certified design soil pressures from DCD, with one exception. The soil pressure from the site-specific SSSI analysis slightly exceeds the certified design soil pressure at a depth of about 26 to 30 feet below the ground surface. At all other elevations the DCD soil pressures are higher than the site-specific soil pressure. The total load on the wall due to the certified design soil pressure on the wall panel will be significantly higher than the total load on the wall due to the soil pressure from the site-specific SSSI analysis. Therefore, the design based on the certified design soil pressures is adequate.

   Similarly, as explained in FSAR Section 3H.1.6, the SSSI seismic soil pressures slightly exceed the DCD soil pressures for the Reactor Building west wall. However, the induced out-of-plane shear and moment in each wall panel due to the DCD soil pressures are greater than the out-of-plane shear and moment due to site-specific SSSI soil pressures. Therefore, the site-specific SSSI seismic soil pressures are acceptable.

b. DCD Tier 1 Table 5.0, ABWR Site Parameters, does not include any parameters specific to lateral seismic earth pressures. STP DEP T1 5.0-1 included a departure in the soil minimum shear wave velocity. The effect of site soil shear wave velocity on the lateral seismic earth pressure was evaluated, as described in FSAR Sections

3H.1.6 and 3H.2.6 for the Reactor and Control Buildings, respectively. This evaluation showed that the standard design is acceptable for site-specific conditions, e.g., site-specific safe shutdown earthquake (SSE) and site-specific soil shear wave velocity. Since DCD Tier 1 Table 5.0 did not include any specific lateral seismic earth pressures and since the differences in lateral seismic earth pressures are a byproduct of STP DEP T1 5.0-1, a departure from standard design for lateral seismic earth pressures was not necessary.

Question 25: SER Section 3.8.4.4.5 discusses the Staff's evaluation related to site-specific departure STP DEP 3.5-2, "Hurricane Generated Missile Protection." This departure addresses the impact of new data and new guidance in RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants" on the STP plant design. Based on this new data and guidance, some site-specific hurricane parameters exceed the tornado-based parameters used as the bounding wind design parameters in the ABWR standard plants. Discuss the implications of this exceedance for site-specific seismic Category I structures.

NINA Response: Some wind and missile parameters for hurricanes, as described in RG 1.221, exceed those due to tornado. Based on this, a site-specific departure, STP DEP 3.5-2, was taken. Evaluation of site-specific Category I structures for the hurricane parameters described in RG 1.221 is described in FSAR Section 3H.11. Based on these evaluations, it was concluded that the site-specific Category I structures, as well as the ABWR standard plant structures, are adequate to withstand the wind and missile parameters of hurricane applicable to the STP site.

Question 27: What edition(s) of ASTM E-185, and the ASTM standards referenced therein, will be used to demonstrate compliance with Appendix H requirements?

NINA Response: DCD Tier 2, Subsection 5.3.1.5.2 documents that ASTM E185-82 will be used to demonstrate compliance with Appendix H requirements.

<u>Question 30</u>: The application describes a single reactor vessel surveillance program for two units based on a four-capsule withdrawal schedule. Will NINA remove surveillance capsules for each unit or from only a single unit? If NINA intends to remove capsules from a single unit, did NINA receive Staff approval for an integrated reactor vessel surveillance program?

<u>NINA Response</u>: NINA will remove surveillance capsules from both STP Units 3 and 4 reactor pressure vessels (RPVs) on the prescribed frequencies as discussed in the attachment to STPNOC's letter to NRC dated July 23, 2009, available in ADAMS at accession number ML092080079.

<u>Question 31</u>: How has NINA demonstrated compliance with the upper-shelf energy requirements for ferritic reactor pressure vessel beltline components and welds in: (a) the procured, pre-service, unirradiated condition to ensure the materials have a Charpy upper-shelf energy of at least 75 ft-lb, and (b) the irradiated condition to ensure that the materials will have a Charpy upper-shelf energy of at least 50 ft-lb throughout the licensed operating periods for the reactors?

<u>NINA Response</u>: The procured, pre-service, unirradiated condition of the base material and weld material of the beltline (reactor core region) of the RPVs for STP Units 3 and 4 are specified by purchase specification to satisfy the upper-shelf energy (USE) of 75 ft-lb, and compliance with this specification is being confirmed by Charpy V-notch tests. Tests have been completed for the STP Unit 3 RPV and have been initiated for the STP Unit 4 RPV. NINA will verify compliance as part of RPV receipt, turnover and ITAAC closure. The RPVs are not yet complete and have not been accepted by NINA.

The predicted worst-case irradiation effects on the beltline vessel materials are addressed in the calculation and analysis provided in the response to ABWR DCD Question 251.5 (DCD Chapter 20). With an initial value of 75 ft-lb, the end-of-life USE exceeds 50 ft-lb (68 Joule). FSAR Section 5.3.1.6.1 describes compliance with reactor vessel material surveillance program requirements. Based on the regulatory requirements of 10 C.F.R. Part 50 Appendix H, the STP Units 3 and 4 "Reactor Pressure Vessel

Material Surveillance Program" was prepared to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment. Specimens which are to be exposed to irradiation will be installed in removable specimen capsules at the inside reactor vessel wall opposite the active core. The surveillance program will test specimens which have the same heat history as the material used in the reactor pressure vessel beltline region fabrication with a specimen withdrawal schedule of 6 Effective Full Power Years (EFPY), 20 EFPY, end-of-life (EOL) (not to exceed the peak EOL fluence), and the schedule further determined based on the test results of the first two specimen capsules per ASTM E185.

Question 32: The Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, NUREG-1503, (July 1994), Section 5.2.1.1 states: "[A]ll ASME Code, Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda given in the [ABWR Standard Safety Analysis Report]. The [combined license] applicant should ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda in effect at the time of [the combined license] application, as endorsed in 10 CFR 50.55a. The . . . applicant should identify in its application the portions of the later code editions and addenda for NRC staff review and approval. The portions of the later Code editions and addenda must be identified to the NRC staff for review and approval with the COL application. This was DFSER COL Action Item 14.1.3.3.2.1-1."

For NINA: How and where does the application address COL Action Item 14.1.3.3.2.1-1? In addition, does the combined license application have any design departures that require ASME Section III relief requests under 10 C.F.R. § 50.55a that require Staff approval before issuance of a combined license?

NINA Response: The COLA contains Departure 1.8-1. This Departure is used to update the requirements to those current Regulatory Guides, Codes, and Industry Standards. This departure made changes to FSAR Tables 1.8-20 and 1.8-21 to update the

ASME Code edition and addenda in effect at the time of the COLA. This departure addresses COL Action Item 14.1.3.3.2.1-1.

The COLA does not identify any design departures that require ASME Section III relief requests under 10 C.F.R. § 50.55a that require Staff approval before issuance of a combined license.

<u>Question 33</u>: The Containment Overpressure Protection System (COPS) opens to vent the containment when the wetwell pressure is 0.72 megapascals, which is significantly below the estimated failure pressure of the drywell head. The Fukushima event showed that hydrogen can leak into the secondary containment and lead to explosions.

a. Is the COPS opening pressure low enough to prevent significant hydrogen leakage into the secondary containment?
b. Does the protection provided by the ABWR COPS meet the requirements for hardened vents in Mark I or Mark II containments required by Order EA-13-109?

<u>NINA Response</u>:

a. Yes. ABWR DCD Section 19.2.4.3, Containment Analysis and Key Results, states:

> Leakage through fixed (mechanical and electrical) penetrations is negligible compared to leakage through large operable penetrations such as the drywell head, equipment hatches, and personnel airlocks. The leakage potential for those operable penetrations was evaluated. Very small [less than 0.0127 cm (0.005 in.)] separation displacements of the sealing surfaces at 0.722 MPa were calculated for the pressure-unseating drywell head closure and equipment hatches. No significant leakage is therefore anticipated before the capability pressure is reached.

Nevertheless, all containment penetrations and hatches leak to some extent. At elevated pressure under severe accident conditions, molecular hydrogen will leak into secondary containment. The consequence of this leakage depends on a number of factors, such as the amount of fuel damage and hydrogen generation, containment pressure, the condition of the numerous penetration seals and drywell head gasket, and the ability to ventilate secondary containment.

Page 22-30 of the Staff's FSER for STP Units 3 and 4 notes that the operators will establish a natural circulation path to ventilate the Reactor Core Isolation Cooling (RCIC) pump room by blocking open a RCIC door and removing an overhead hatch for additional cooling. Other aspects of this ventilation plan are discussed in FSER Section 22.2.4.2.6, Ventilation, beginning on page 22-50 of the FSER for STP Units 3 and 4. This action is part of an overall strategy for ventilating secondary containment which involves establishing a ventilation flow path in the Reactor Building. A license condition on the ventilation plan is discussed in Section 22.3.5f on page 22-69 of the FSER for STP Units 3 and 4. Although the STP Units 3 and 4 FLEX strategy prevents core damage and subsequent hydrogen generation, this ventilation strategy would provide a path to remove any hydrogen, if core damage does occur.

b. Yes, the protection provided by the ABWR COPS meets the requirements for hardened vents in Mark I or Mark II containments required by Order EA-13-109. Furthermore, the ABWR FSER, NUREG-1503, p. 6-33 states:

> Generic Letter (GL) 89-16, "Installation of Hardened Wetwell Vent," addressed the need for modifications of BWR containment designs to reduce their vulnerability to severe accident challenges. The staff finds that the ABWR design has included the containment overpressure protection system which addresses this GL.

<u>Question 34</u>: In FSAR Section 6C.1, NINA committed to following the guidance in RG 1.82, Rev. 3, "Water Sources for Long-Term Recirculation Following a Loss-of-coolant Accident." Since the STP application was submitted, new guidance has become available regarding the long-term cooling regulated under 10 C.F.R. § 50.46(b)(5), which is described in RG 1.82, Rev. 4. Did the Staff consider the newly-identified issues in Revision 4 of RG 1.82 when reviewing the application (i.e., vortexing, flashing, deaeration, and chemical effects)? If so, provide details on how NINA addressed the safety considerations available in the updated guidance. If not, explain why NINA's approach meets 10 C.F.R § 50.46(b)(5).

NINA Response: The ABWR Emergency Core Cooling Systems (ECCS) strainers were certified to RG 1.82 Revision 1. STP Units 3 and 4 voluntarily upgraded the ECCS suction strainers to a much larger, cassette-type strainer design and committed to RG 1.82 Revision 3, which was the RG in place at the time of the initial application and during NRC review. RG 1.82 Revision 3 added requirements for consideration of chemical effects on strainer head loss and its impact on ECCS pump available Net Positive Suction Head (NPSH), and added a requirement to evaluate the impact of debris passing through the strainers on downstream components, including fuel.

The STP Units 3 and 4 containment design eliminates fiber, aluminum and zinc (except for qualified containment coatings). Additionally, STP Units 3 and 4 uses strainer sizes consistent with an existing Japanese ABWR (the Reference Japanese ABWR (RJABWR)), which is sized based on fiber quantities two orders of magnitude larger than assumed (based on latent fiber) at STP Units 3 and 4. STP Units 3 and 4 have also committed to a rigorous suppression pool cleanliness program to ensure that latent debris is minimized. Despite these design and programmatic enhancements, the STP Units 3 and 4 strainer head loss was conservatively evaluated assuming one cubic foot of latent fiber, debris (e.g., dirt/dust), 4.5 ft^2 of latent aluminum (assumed to completely corrode), postulated zinc (from assumed destroyed coating), and silicon from assumed exposed concrete. These evaluations did not take credit for containment overpressure. Due to the large surface area of the ECCS strainers, the head loss due to fiber, debris (e.g. dirt/dust), and chemical effects is very small based on analyses and confirmatory testing for the RJABWR.

As part of the evaluation of the effect of debris passing through the strainers on downstream components, STP Units 3 and 4 committed to a future downstream test on the fuel that will be used in the initial cycle of operation. This test will be performed 18 months prior to initial operation. The test will conservatively assume that all debris small enough in size will pass through the strainers and reach the fuel.

As for the issues mentioned in the question, flashing and deaeration are not an issue for STP Units 3 and 4 due to the low pressure drop across the strainers. Vortexing is also not an issue because the suppression pool contains a significant elevation of water above the strainers.

Question 35: In FSAR Section 6C.3.1.9.3, NINA applies an NRC-approved topical report, WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," (available at ADAMS accession number ML081150383) when evaluating chemical effects on long-term recirculation cooling following a loss of coolant accident. However, WCAP-16530-NP-A is specific to pressurized water reactor designs. Boiling water reactors have post-loss-of-coolant accident containment conditions that may result in different chemical interactions than those analyzed in WCAP-16530-NP-A. Discuss how the application of pressurized water reactor [PWR] guidance to a boiling water reactor design is adequate to meet the long-term core cooling requirements of 10 C.F.R. § 50.46(b)(5).

NINA Response: The STP Units 3 and 4 primary containment design prohibits fiber, aluminum, and zinc (other than in qualified coatings). For operational margin, conservative quantities of latent fiber, aluminum, destroyed qualified coatings (which include inorganic zinc) and silicon from assumed exposed concrete were evaluated. At the time of these evaluations, the Boiling Water Reactor Owners Group (BWROG) chemical effects testing had not been started, so methods used for PWR chemical effects in WCAP-16530 were specifically adapted to the STP Units 3 and 4 ABWR suppression pool water chemistry. The uses of WCAP-16530 were discrete and were performed in a conservative manner using the water chemistry specific to the STP Units 3 and 4 suppression pool. For example, all latent aluminum was assumed to form precipitates. Additionally, the debris quantification analyses used the bounding suppression pool pH value which resulted in corrosion of all zinc assumed from destroyed qualified coatings. Finally, due to the very small quantity of fiber compared to the large strainer surface area, there is a negligible buildup of a fiber bed to trap particulate and chemical debris in its interstices, and therefore the impact on head loss due to chemical effects is minimal.

Question 36: FSAR Section 6C.5, which provides the Strainer Sizing Analysis Summary, states: "Debris on the screen creates a pressure drop as predicted by NUREG/CR-6224 and NUREG/CR-6808 which is referenced by Regulatory Guide (RG) 1.82. Pressure drop caused by the mixed particulates and fiber bed is calculated by the equation shown on NUREG/CR-6224 Appendix B."

RG 1.82, Rev. 4 states: "In future evaluations, BWR [boiling water reactor] strainer designs should consider subsequent guidance developed during the resolution of GSI-191 and GL 2004-02 including chemical and downstream effects and strainer head loss and vortexing." The NUREG/CR-6224 correlation was primarily developed for application to flat screens and plates, but the Advisory Committee on Reactor Safeguards (ACRS) has questioned its application to flat screens and plates (see G. Wallis review of the NUREG/CR-6224 Head Loss Correlation, 8/20/2004 (available at ML042400166)). Comparisons between the NUREG/CR-6224 correlation predictions and test data for flat screens and plates showed that the correlation did not predict or bound measured data in many situations (see NUREG-1862; NUREG/CR-6917). Additionally, the correlations are not applicable to the complex geometries present in a strainer and may yield non-conservative results that may result in undersized strainers. Consequently, RG 1.82, Rev. 4 further states: "Licensees should validate the adequacy of ECCS [Emergency Core Cooling System] strainer designs through testing applicable to plant-specific conditions. Analytical or empirical head loss correlations should not be used to validate plant-specific debris bed head losses." In view of these concerns, provide details of the plant-specific strainer tests that have been performed to validate the adequacy of the STP strainer design.

<u>NINA Response</u>: The STP Units 3 and 4 ECCS suction strainers use the same size strainers that are installed in the RJABWR. The size of the RJABWR strainers is based on fiber quantities two orders of magnitude greater than those assumed (based on latent fiber) at STP Units 3 and 4. The RJABWR strainers were tested by the strainer vendor to confirm the analytical correlations used in the sizing of these strainers. This testing was

described in Toshiba proprietary report [FSAR Appendix 6C, Reference 6C-11], which was provided to the NRC Staff for review by letter dated February 15, 2010.

The ACRS also reviewed the adequacy of the STP Units 3 and 4 long term cooling in five meetings from 2010 to 2012. During these meetings, the ACRS reviewed ECCS suction strainer NPSH margin, chemical effects, and downstream effects testing. This review culminated in a November 7, 2012 letter to the Commission concluding that ACRS concurred with the Staff's assessment that long-term core cooling for design basis conditions was adequately met pending successful resolution of the downstream effects testing. This testing will be required as part of a license condition, as discussed in Proposed License Condition 06.02-1 in FSER Appendix A.

Question 37: The Staff issued RAI 07.09-5 requesting that NINA provide sufficient information addressing a safety and hazards analysis, a sneak circuit analysis, and a timing analysis for the digital instrumentation and control (I&C) systems. In response to this RAI (STPNOC Letter U7-C-STP-NRC- 090157), NINA stated that the verification of the I&C system design and analysis will be accomplished during the ITAAC phase. In SER section 7.9S.4, the Staff found the related ITAAC acceptable, such that when the ITAAC is performed and the acceptance criteria are met the facility would have been constructed and will operate in conformance with the combined license and the NRC regulations.

The response to RAI 07.09-5 references the response to RAI 14.03.05-04 that states: "The safety-related I&C systems are deterministic. The response times for the system elements, including architecture, communications (including timing and loading) and processing elements will be analyzed in accordance with BTP 7-21 to verify that the systems' performance characteristics are consistent with the safety requirements established in the design basis for these systems . . . . Regarding the request for additional ITAAC, STPNOC's position is that the existing ITAAC is appropriate as discussed in the response to RAI 14.03.05-8." The response to RAI 14.03.05-8 states, in turn: "The ITAAC that can be considered I&C related Design Acceptance Criteria (DAC) are provided in STP 3&4 COLA Part 2 Tier 1, Section 3.4 Table 3.4 Items 7-15. This is supported by the ABWR DCD Subsection 14.3.3.4 and NUREG-1503 Section 14.3.3.4. As noted therein, the DAC provide the process and acceptance criteria by which the details of the I&C systems' design are developed, designed and evaluated."

The acceptance criterion listed in the ABWR DCD FSAR Tier 1, Section 3.4 Table 3.4 Item 8.e states that the Software Management Plan shall define "[t]he Design Definition phase design activities, which shall address the development of the following implementing equipment design and configuration requirements . . . [d]ata communications protocol, including timing analysis and test methods . . . ."

a. Is this the only ITAAC that provides the timing analysis of the entire safety I&C system from sensor output to the final actuation device?
b. Is there an ITAAC that verifies that the as-built safety I&C system can complete the required safety functions (e.g., reactor trip and engineered safety features [ESF] actuation system functions) in the required time assumed in the safety analysis?

NINA Response:

a. ITAAC 3.4.8.e is the principal ITAAC that provides for timing analysis of the I&C system. While no other ITAAC is specifically dedicated to validating response time of the safety I&C system, there are several ITAAC that cover analysis and validation of performance requirements which include response time.

- ITAAC 3.4.2 requires that a report exist that concludes the Safety System Logic and Control (SSLC) design basis performance requirements are met (which would include timing requirements).
- ITAAC 3.4.8.b(7) requires performing equipment integration and validation testing that demonstrates that safety-related functions identified in the design input requirements are met.
- ITAAC 3.4.8.h(1) Integration Test Reports listed in the acceptance criteria would contain results from integration testing which would include response time testing.
- ITAAC 3.4.10.e states that "validation shall be performed ... of the developed software as installed on the target hardware that demonstrates compliance of the software with the software requirements specifications and compliance of the device(s) under test with the system design specifications."

Also, each ESF logic and control system (ELCS) controlled system (Residual Heat Removal (RHR), RCIC, High Pressure Core Flooder (HPCF), etc.) has ITAAC verifying initiation within a specific time period. For example, RHR ITAAC 2.4.1.3.c(2) verifies the RPV injection valve receives a signal to open provided a low pressure permissive signal is present, and the valve opens within 36 seconds after receiving the low reactor pressure permissive signal. The response time allocation to ELCS is inherent in meeting the overall timing requirement, and is similar for the ITAAC for other ELCS controlled systems. These ITAAC verify that the system as a whole meets the required response time to support the safety analysis.

b. ITAAC 3.4.8.e provides a timing analysis that covers all SSLC systems. In addition, several other ITAAC cover various performance verifications that verify that timing requirements are met, as discussed in the response to Part a.

Question 38: Branch Technical Position (BTP) 7-21 states, "Design basis documents should identify design practices that the applicant/licensee will use to avoid timing problems. Risky design practices such as non-deterministic data communications, non-deterministic computation, use of interrupts, multitasking, dynamic scheduling, and event-driven design should be avoided." The STP combined license application, Tier 2 Section 7DS.2.3.2.2 states that "The timing analysis is performed as required by the NRC in the Plant Specific Action Items described in the Safety Evaluation report for the Common Q Topical Report, WCAP-16097-P-A. This topical report provides additional information on the deterministic performance of safety systems based on use of the Common Q platform." Per the guidance of BTP 7-21, use of multitasking designs should be avoided in order to meet performance and timing requirements. However, the NRC Staff has approved platforms (e.g., Common Q) that employ multitasking provided certain limitations are enforced (e.g., Central Processing Unit loading limitations). In the case of the Common Q platform, WCAP-16097-P-A states that as long as the Central Processing Unit load is less than 70 percent, then the application program will operate deterministically. Since the Engineered Safety Features Logic and Control System will be developed using the Common Q platform, is there an ITAAC to verify that the as-built system will meet the 70 percent Central Processing Unit load restriction?

NINA Response: There are several ITAAC that verify the Common Q Central Processing Unit (CPU) (Processor Module) requirements are met. Use of Common Q for the ELCS imposes requirements, including application restrictions, identified in the SER to WCAP-16097-P-A. While there is not a specific ITAAC to verify that 70% loading of the Common Q CPU is met, the following ITAAC cover verification of requirements for ELCS which would include validating that loading of as-built ELCS AC160 processors remains below 70%.

- ITAAC 3.4.8.b(7) requires performing equipment integration and validation testing that demonstrates that safety-related functions identified in the design input requirements are met.
- ITAAC 3.4.10.e states that "validation shall be performed ... of the developed software as installed on the target hardware that demonstrates compliance of the software with the software requirements specifications and compliance of the device(s) under test with the system design specifications."
- ITAAC 2.7.5.2 validates that the ECFs use deterministic communication protocols. For Common Q to maintain determinism in communications protocols, the processor load must remain <70% per WCAP-16097-P-A.

Question 39: How does the application address the issue of spurious actuations induced by means other than heat or fire, such as instrumentation and control (I&C) system failures (e.g., control system or safety I&C system common cause failures)? This issue is of special concern when these active failures may lead to transients that are not analyzed in the safety analysis.

NINA Response: The topic of common mode failures is discussed extensively in NUREG-1503, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, in Section 7.2.6, Defense-in-Depth Analysis, on pages 7-32 through 7-41. The NRC Staff raised numerous issues, performed independent studies, employed Lawrence Livermore National Laboratory to perform diversity studies, and worked with the ABWR I&C system designers to develop a set of system features to provide a highly reliable design which provides assurance that essential safety functions can be performed considering common mode I&C system failures.

The ABWR FSER concludes:

> Appendix C [of the SSAR] addresses the issues identified above and specifies the diversity and set of equipment that will not be subject to potential common-mode failures. Therefore, DFSER Open Items 7.2.6-1, 7.2.6-2, and 7.2.6-3 are resolved and the design meets the staff's proposed applicable regulation for digital instrumentation and control systems.

These system features are described in detail throughout Chapter 7 of the DCD and the FSAR for STP Units 3 and 4, especially in Appendix 7C, Defense Against Common-Mode Failure in Safety-Related, Software-Based I&C Systems.

The STP FSAR Chapter 7DS states that the platforms that implement the SSLC system have been designed in large part based on four essential design principles: (1) redundancy, (2) independence, (3) the need for defined determinism in data processing and communication, and (4) implementation of a diversity and defense-in-depth (D3) philosophy, as well as the attribute of simplicity. The four principles and one attribute are embodied in the underlying basis of IEEE-603. The safety-related digital instrumentation and control platforms as described in Tier 1 Section 3.4; Tier 2 Subsections 7.1, 7.1S, 7.2, 7.3, 7.6, and 7.9S; and elsewhere in the FSAR satisfy IEEE-603 and thus the four principles plus one attribute. Conformance to IEEE-603 is explicitly described in each of the above Tier 2 sections.

The fundamental design philosophy of the I&C systems and the Essential Communication Functions was developed to ensure safety system actuation, avoid spurious or inadvertent system actuation, and be tolerant of single failures and common mode failures. Specific features of the design, including the prevention of spurious actions, have been discussed in detail throughout the DCD and the COLA for STP Units 3 and 4 for both the safety-related systems (Reactor Trip and Isolation System (RTIS) and the ELCS) and the non-safety-related process control systems such as Feedwater Control, Turbine Control and the Plant Information & Control System (PICS).

The primary safety-related I&C protection systems and the primary process control systems are designed to a high level of reliability and make up a significant portion of the plant's overall defense-in-depth strategy. An integral part of the systems reliability is the prevention of spurious control commands due to I&C component failures or abnormal inputs into the systems. The RTIS and SSLC systems use a high level of redundancy and signal validation of control signals to eliminate potential spurious actuations. Tier 1 Section 3.4 and Tier 2 Section 7DS of the FSAR address the RTS and SSLC design architecture. Both of the protection systems provide 2-out-of-2 final actuation verification of important signals as depicted in Figures 3.4b and Figures 7DS-1 through 7DS-4. This design provides a high level of assurance that a spurious signal will not result in an actuation. The design of primary non-safety-related control systems implement similar levels of redundancy and final control actuation validation to ensure a high level of systems and overall plant reliability.

The same principles which minimize failure due to common mode failure also minimize spurious actions due to common mode failure.

In addition, the prevention of spurious actuations that may impact the safety analysis is further accomplished through the use of independent equipment actuations related to a process function. For example, a Component Interface Module (CIM) will receive input control commands from two Safety Logic Function (SLF) Remote Digital

Logic Controllers (RDLCs) for components where it is desired to reduce the probability of spurious actuation. For this case, the CIM performs 2-out-of-2 coincidence logic for the commands received from two SLF RDLCs. Furthermore, separate and independent actuation of associated process equipment must occur for some critical functions. For example, a pump and its discharge valve must both receive independent actuation signals to start the pump and open the valve to allow a process to run.

Question 42: How has the Staff evaluated that, in the event of a fire in the digital instrumentation and control panels located in the Control Room Complex, operators can identify the loss of control due to fire-induced spurious actuations and initiate transfer to the remote shutdown panel prior to the plant experiencing unrecoverable conditions?

NINA Response: FSER page 9-177 states:

> In its response to RAI 09.05.01-1, dated August 12, 2009 (ML092260197), the applicant states that Departure STD DEP T1 3.4-1 will not change the design basis behind the ABWR's spurious actuation assumption. Because of this the applicant stated that there will be no change to the fire protection design criteria and no change in the potential for spurious actuations. The staff found the applicant's response concerning Departure STD DEP T1 3.4-1 acceptable. The ABWR DCD and the STP FSAR present an acceptable design basis for the fiber optic cables. The staff agreed that the use of fiber optics for control and instrumentation has an extremely low potential for spurious actuations, either single or multiple.
>
> The STP Units 3 and 4 FSAR Section 9.5.1.1.7 states:
>
> As stated above, the systems are separated by fire areas on a divisional basis. The ESF Logic and Control System (ELCS) utilizes redundant fiber optic links to communicate ESF system level actuation status to the Remote Digital Logic Controllers (RDLCs), which control the remote input/output functions and the actuation of the electromechanical components. The RDLC utilizes diagnostics to verify the validity of each redundant message. The redundant messages received by the RDLC must match for component actuation to occur. The probability of spurious messages occurring on each of the redundant links that both pass the communication diagnostics and that also match between the two redundant links is essentially zero.

Spurious actuations due to a fire in the control room complex are extremely unlikely. Loss of control can be identified via numerous diagnostic and alarm functions.

Within the control room complex, the divisional equipment and related cables are installed in one of the two separate electrical equipment rooms on either side of the actual main control room. Within each room, the two divisional sets of equipment and cabling are physically separated from each other and non-safety equipment and cabling. Each set of divisional equipment is installed in qualified cabinets that minimize the potential spread of any electrical or fire related event to the other set of divisional equipment or the non-safety-related equipment. Some of these cabinets have installed temperature alarms. Should a fire occur in a division, multiple system alarms will be actuated in the main control room to indicate the loss of divisional communication or the initiation of signals that do not exist in the other divisions (based on the use of coincident logic). Should a fire become large enough to impact more than one division, the room fire detection system will alarm in the main control room as well as the high potential that the control room operator will smell and see smoke in the electrical equipment room. The multiple system health alarms and coincident logic alarms will provide significant evidence that a fire exists within the control room complex.

Based on this information and the habitability of the main control room, the plant operators will make the decision to transfer control of the plant to the remote shutdown system in accordance with plant procedures.

<u>Question 43</u>: How are fire-induced shorts to ground on ungrounded systems evaluated to ensure that safe shutdown functions are not impeded by spurious operation(s) caused by ground fault equivalent hot shorts?

NINA Response: Spurious operation of equipment due to single or multiple ground faults or hot shorts will be part of the as-built Fire Hazards Analysis as stated in FSAR Section 9.5.1.1.7:

> The evaluation of single and multiple spurious operations that could adversely impact post-fire safe shutdown will be performed in a manner that is consistent with the methodology of NEI 00-01, Revision 2 as modified by the guidance of RG 1.189 Revision 2 as it applies to Single and Multiple Spurious Operation Analysis.

Question 44: Please clarify how NINA's combined license application, Revision 12 (Attachment 3 to the NINA letter dated April 21, 2015 (ML15124A267)) addresses the ACRS's concern expressed in its February 19, 2015, letter to Chairman Burns (ML15039A006) with respect to the turbine control and protection system. The ACRS stated:

> 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.

Is the Staff's safety determination based upon this updated commitment from NINA, and if so, what obligation is placed on the licensee to fulfill the commitment as stated?

NINA Response: FSER Section 3.5.1.3.4 states that "With this license condition [requiring certain inspection and testing as part of the turbine maintenance program] … the staff finds that the overall probability of turbine missile damage to SSCs important to safety is acceptably low." This is appropriate because the turbine missile analysis will be performed in the future during detailed design of the turbine.

NINA's obligation to meet this updated commitment is reflected in FSAR Section 3.5.4.5 and is subject to the NRC rules in 10 C.F.R. § 50.71(e), which requires that the FSAR be updated if it is changed and that the NRC be notified of the change.

Question 45: 10 C.F.R. § 20.1601(d) requires that the licensee establish controls in a way that does not prevent individuals from leaving a high or very high radiation area. Therefore, a door to a high or very high radiation area must be designed in a way that would allow an individual inadvertently locked inside such an area, to leave the area.

How are the 10 C.F.R. §§ 20.1601 and 20.1602 requirements met for radiation protection doors in containment?

NINA Response: Detailed plant design has not been completed at this date; however, any doors that isolate high or very high radiation areas inside or outside containment will be designed such that they will not prevent personnel from exiting the area as required by 10 C.F.R. § 20.1601(d).

In addition, NEI 07-03A ,"Generic Template Guidance for Radiation Protection Program Description" is incorporated by reference in the STP Units 3 and 4 FSAR and stipulates that access to high and very high radiation areas will be controlled in accordance with RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants." Regarding the possibility of an individual being inadvertently locked inside a high or very high radiation area, Section 1.5 (Physical Barriers) of RG 8.38 states, "If doors are self-locking, personnel must be able to open them from the inside without a key."

Question 46: The FSAR provides that a Plant Operations Review Committee (PORC) will be established to advise the plant General Manager on all matters related to nuclear safety at STP Units 3 & 4. When does the PORC begin—upon issuance of the combined license, at fuel load, or at some other time?

NINA Response: As described in the NINA Quality Assurance Plan Description and in FSAR Section 13.1.1.1.3, the PORC is an operational requirement. It is required to be in place at fuel load but may be established earlier at the Plant Manager's discretion.

Question 47: The application (FSAR, page 13.1-10) provides: "Toshiba will have overall responsibility for design and configuration control." Will the licensee have access to all design documents (including proprietary documents) for the life of the plant?

NINA Response: Yes, NINA/STPNOC will have access to all design documents (including proprietary documents) for the life of the plant.

Question 48: The FSAR incorporates by reference NEI 06-13, "Template for an Industry Training Program Description." The SER references NEI 06-13A, Revision 1 as the acceptable template. Explain the discrepancy.

NINA Response: When the NRC endorses an NEI guidance document it is re-issued with an "A" suffix. The NRC conclusion in the FSER recognizes NEI 06-13:

> The staff's review confirmed that the applicant has addressed the relevant information relating to training by incorporating NEI 06–13 by reference. The staff's review also confirmed that the applicant has adequately addressed the guidance in NUREG–0800, Sections 13.2.1 and 13.2.2, and no outstanding information is expected to be addressed in the COL FSAR related to this section. The information is therefore acceptable.

Revision 1 to NEI 06-13A was issued after the STP Units 3 and 4 COLA was submitted and added a cold license training plan appendix. No other changes to the document were made.

Question 49: In the draft license, the Staff proposes the following license condition:
"No later than 8 months before fuel is allowed onsite (protected area), NINA shall develop a written protective strategy that describes in detail the physical protection measures, security systems, and deployment of the armed response team relative to site-specific conditions, including but not limited to, the final facility layout, and the location of target set equipment and elements in accordance with 10 CFR Part 73, Appendix C.II.B.3.c.(v)."
Final SER § 13.6 states: "However, the staff has not proposed any license condition implementation requirements for the STP COL application since the implementation milestones for these security programs are specified by 10 CFR 73.55(a)(4). Because the implementation milestones for these security programs are controlled by 10 CFR 73.55(a)(4) rather than by license condition, the applicant will need to update Table 13.4S-1 to reflect this."

a. Why is this license condition necessary? Were there similar conditions in previously-issued combined licenses?

b. Why is there a license condition for **development** of the protective strategy in the draft license? The Staff's Final SER addresses **implementation** license conditions; is this the same thing?

NINA Response:

a. In accordance with the guidance given in SECY-05-197, NINA proposed a license condition for the implementation milestone only.

b. The Staff developed this license condition.

<u>Question 50</u>: In the draft license, the Staff proposes the following license condition: "No later than 8 months before fuel is allowed onsite (within the protected area), NINA shall develop a written protective strategy that describes in detail the cyber protection measures, systems, and deployment of the cyber security program relative to site-specific conditions to include, but not be limited to, the final facility design and the location of target set equipment and elements in accordance with 10 CFR 73.54."

a. Why is this license condition necessary? Were there similar conditions in previously-issued combined licenses?

b. Why is there a license condition for **development** of the cyber security program but not the **implementation** as indicated in FSAR 13.4S-1?

<u>NINA Response</u>:

a. In accordance with the guidance given in SECY-05-197, NINA proposed a license condition for the implementation milestone only.

b. The Staff developed this license condition.

<u>Question 51</u>: The Staff issued RAI 17.04-5 requesting that NINA address in FSAR Subsection 17.4S.1.1.2 the interface responsibilities of the expert panel related to risk-significant structures, systems and components (SSCs) within the scope of the design reliability assessment program (D-RAP) that are not modeled in the applicant's probabilistic risk assessment (PRA). In response to the Staff's request, NINA added a commitment to the FSAR under which the licensee will identify and periodically review any proposed changes resulting in an increase in the deterministically established risk of an SSC not modeled in the PRA with the expert panel at a frequency determined by the panel.

a. Identify any plant-specific SSCs that are not modeled in the PRA and have been identified as risk-significant using a deterministic basis and explain why they are considered risk-significant.

b. If there are none that meet these criteria, then explain in general how SSCs are determined to be risk-significant when they are not modeled in the PRA.

<u>NINA Response</u>:

a. NINA has added SSCs to the list of risk-significant SSCs for use in the D-RAP, using the criteria in FSAR Section 17.4S.1.4. See, e.g., ML15294A429. This process is ongoing and will be completed as part of completion of detailed plant engineering and design.

b. FSAR Section 17.4S.1.4 describes the process for determining whether SSCs are risk-significant when they are not modeled in the PRA. In summary, the process consists of the following tasks:

- Identification of the system function(s) supported by that component.
- Development of a risk categorization of the component based on deterministic insights.
- Identification of critical attributes for components determined to be safety/risk significant.

The deterministic risk ranking process is described in detail in FSAR Section 17.4S.1.4.2. In implementing the risk ranking process, the expert asks the following questions and ranks the SSC based upon the answers:

- Is the function used to mitigate accidents or transients?
- Is the function specifically called out in the Emergency Operating Procedures (EOPs) or Emergency Response Procedures (ERPs)?
- Does the loss of the function directly fail another risk-significant system?
- Is the loss of the function safety significant for shutdown or mode changes?
- Does the loss of the function, in and of itself, directly cause an initiating event?

Question 52: SER, Ch. 19, Attachment A notes that NINA commits to locate spare batteries and chargers in suitable areas. In FSAR Part 11 (Mitigative Strategies Report, Fire Fighting Response Strategy), NINA states that an action has been added in the Corrective Action Program to locate communication device's spare batteries and chargers near the Control Room or other suitable areas (Commitment: 08-18140-11).

Explain why an action in a Corrective Action Program is being credited for a licensing basis document.

NINA Response: The FSAR (where the commitment is made) is the licensing basis document that requires completion of the commitment. NINA's Commitment Management Program requires documenting commitments in the ABWR Corrective Action Program (CAP) in order to track the commitment to completion. The action in the CAP is not credited for a licensing basis document.

NRC commitments are uniquely identified in the CAP and can be easily retrieved and processed. In addition, procedure U7-P-AD02-0003, "ABWR Corrective Action Program", contains controls to ensure the timely completion of all actions, including commitments.

Question 53: STP Application Part 11-49 (Rev. 12) contains Commitment: 08-18140-56, which provides that the considerations for equipment survivability and personnel accessibility within plant areas will be evaluated.
a. Can this issue be evaluated prior to issuance of a combined license?
b. How does this commitment demonstrate an acceptable strategy in the licensing basis?

NINA Response:

a. It is not possible to complete these considerations at this time as the final plant design has not yet been completed.

b. The Staff's FSER on the Mitigating Strategies Report (COLA Part 11) stated:

> …the applicant has adequately followed the guidance of DC/COL-ISG-016; NEI 06-12; and the February 25, 2005, guidance letter. The staff finds that the applicant provided sufficient information at the COL application stage, including commitments made in the MSD, to meet the requirements of 10 CFR 52.80(d) and to provide reasonable assurance that the requirements in 10 CFR 50.54(hh)(2) will be met prior to the initial fuel load of the STP Units 3 and 4, respectively.

COLA Part 11 and the guidance documents referenced above provide the acceptance criteria that must be met during the development of the detailed plant design and procurement documents. As indicated in the quotation provided above, this provides reasonable assurance that the requirements of 10 C.F.R. § 50.54(hh)(2) will be met.

<u>Question 54</u>: Part 4 of the application, Limiting Condition for Operation (LCO) 3.8.11 states:
"The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystems required by LCO 3.8.10, "Distribution Systems – Shutdown;"

b. Two or more diesel generators (DGs) capable of supplying the required OPERABLE features via the onsite Class 1E AC electrical power distribution subsystems required by LCO 3.8.10.

APPLICABILITY: MODE 4 and MODE 5 with water level in the refueling cavity < 7.0 meters above the reactor pressure vessel flange."

Surveillance Requirements SR 3.8.11.1 state: "For AC sources required to be OPERABLE, the SRs of Specification 3.8.2 are applicable." However, Specification 3.8.2, AC Sources – Refueling is applicable for MODE 5 with water level in the refueling cavity > 7.0 meters above the reactor pressure vessel flange.

Are there surveillance requirements for MODE 4?

<u>NINA Response</u>: Yes, Surveillance Requirement (SR) 3.8.11.1 requires that all of the SRs required by LCO 3.8.2 must also be met when LCO 3.8.11 is applicable in "MODE 4 and MODE 5 with water level in the refueling cavity < 7.0 meters above the reactor pressure vessel flange."

LCO 3.8.2, 'AC Sources – Refueling,' and LCO 3.8.11, 'AC Sources – Shutdown (Low Water Level),' establish requirements for the minimum number of qualified offsite circuits and diesel generators required to be operable when either of these LCOs is applicable. The SRs and SR frequencies that verify operability of the qualified offsite circuits and diesel generators in LCO 3.8.2 and 3.8.11 are identical. The reason different LCOs are established for refueling (LCO 3.8.2) and shutdown (low water level) (LCO 3.8.11) is that different actions are required when the LCO is not met.

All SRs and required frequencies for qualified offsite circuits and diesel generators are established in LCO 3.8.1, 'AC Sources – Operating.' LCO 3.8.2, 'AC

Sources – Refueling,' identifies the subset of those SRs that must be met when in the refueling mode in SR 3.8.2.1 by stating that the SRs of Specification 3.8.1 are applicable and listing the exceptions. LCO 3.8.11, 'AC Sources – Shutdown (Low Water Level),' identifies SRs that must be met when in shutdown with low water level by stating that the SRs of Specification 3.8.2 are applicable. This approach ensures that the requirements of each SR is identified in a single location and implemented by a single procedure which significantly reduces the potential for error when tracking SR performance and reduces the administrative burden when changes are made.

Question 56: In Chapter 22, the Staff states that NTTF Recommendation 5.1, "Reliable Hardened Vents for Mark I and Mark II Containments," is not applicable "because it applies to boiling-water reactor (BWR) type plant designs with Mark I and Mark II Containments, which differ significantly from the Advanced Boiling Water Reactor (ABWR) containment." In SER § 1.4S.4.1, the Staff states, "the ABWR containment combines design features of Mark II and Mark III containments." Section 6.2.1.1 of the SER states:

> This primary containment design basically uses combined features of the Mark II and Mark III designs, except that the drywell consists of UD [upper drywell] and LD [lower drywell] volumes. The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark II wetwell.

In addition, the drywell and wetwell free volume of the ABWR is approximately 0.47 E6 ft^3 and the containment design pressure is 45 psig. The BWR Mark II design parameters are a free volume of 0.4 E6 ft^3 and containment design pressure of 45 psig.

Please describe in more detail any significant design differences between the ABWR and BWR Mark II containments that justify no further review by the Staff of NTTF Recommendation 5.1.

NINA Response: Although the ABWR primary containment design uses features of the Mark II and Mark III designs, there are differences. As noted, the drywell consists of UD and LD volumes, and the vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. NUREG-1503, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, Main Report,

p. 19-10 notes that the drywell head has an increased pressure capacity which allows a higher COPS actuation pressure, since the drywell head is the limiting containment component. The most important difference, however, is that the ABWR already had a design for a hardened containment vent which met the requirements of GL 89-16, "Installation of a Hardened Wetwell Vent," as discussed in NUREG-1503 p. 6-33.

Question 63: The area where STP is located has been in a drought since 2010 and flows in the Colorado River have been affected. How have the drought conditions been taken into account to ensure that the STP site is still the obviously superior site?

NINA Response: Drought conditions are not uncommon in Texas and were considered during the original engineering, design and permitting of STP. The site was originally designed to accommodate four operating units and the Main Cooling Reservoir (MCR) was sized accordingly. Also, sufficient senior water rights were procured to ensure that four units could operate even under severe drought conditions.

The most recent drought episode, while significant, did not surpass the drought of record for the Lower Colorado River that occurred during the 1950s. STPNOC was able to obtain sufficient water from the run of river throughout the drought and did not have to exercise its senior water rights for water from the upland reservoirs.

When the two new units are placed into operation, additional water use is anticipated. However, no new appropriations of water are required to support the needs of the new units. Because of the unique design of the MCR, the availability of firm water through its senior water rights, and the use of water management strategies, the STP site remains the obviously superior site, in part because of its ability to sustain operations during severe drought episodes.

<u>Question 64</u>: Explain the analysis, if any, that was completed to show that the drought referenced in Question 63 had no impacts on terrestrial/ecology conclusion in the FEIS [Final Environmental Impact Statement].

<u>NINA Response</u>: The site, situated within the Western Gulf Coast Plains ecoregion, contains vegetative assemblages and associated species that are suitably adapted to the local climatic regime. Droughts are not uncommon and while individual organisms that utilize the vegetation for forage and shelter may be impacted by a given drought event, populations as a whole, tend to endure. The recent drought was not as severe as the drought of record discussed in the FEIS, and the conclusions in the FEIS remain valid for drought conditions.

Regulatory agencies have not required ecological monitoring of the STP site or its associated transmission corridors since the period of reservoir filling (mid 1980s) and there is no ongoing formal monitoring of terrestrial resources (aside from various bird counts). STPNOC informally monitors site environs for changing conditions and ecological conditions have remained generally unchanged and satisfactory during the recent drought. The proposed location of Units 3 and 4 consists primarily of previously developed lands (warehouses, parking lots, laydown yards, etc.), a mowed field, and a relatively open shrubland area. None of these areas offers particularly attractive habitat for the terrestrial species that inhabit the site.

Some non-disturbed areas within the site do offer suitable habitat including bottomland hardwood forests, upland forests, shrub/scrub assemblages and wetlands. While not formally monitored, no obvious changes in these habitats (such as dying trees or other vegetation) occurred as a result of the drought. In addition, no obvious changes in the animal populations that utilize these areas have been noted.

Question 65: In a letter dated September 16, 2010 (ML103020111), the U.S. Army Corps of Engineers (USACE) provided a copy of the Unified Stream Methodology assessment conducted on relatively permanent waters along the Colorado River, in Matagorda County, indicating that STP should submit a compensatory mitigation plan for the proposed dredging and filling associated with STP Units 3 & 4. Has NINA since received the USACE permit under Clean Water Act § 404? If not, discuss the expected schedule for issuance.

NINA Response: Permit Number SWG-2007-00786 was issued by the USACE - Galveston District on February 6, 2012. This permit authorizes STPNOC to conduct maintenance dredging and expansion of two existing barge slips located on the Colorado River and to place six culverts and fill material in waters of the United States while constructing a heavy-haul road from the barge slip to the construction site.

Question 66: In 2014, the Environmental Protection Agency (EPA), under Clean Water Act § 316(b) issued regulations for industrial cooling systems that may apply to new units at existing facilities. FEIS § 3.2.2.2, at 3-7, states that the review team concludes that Units 3 and 4 and the intake structure on the Colorado River that supports their operation would not qualify as a "new facility," but rather qualify as an "existing facility." The Staff also acknowledged that the EPA was developing regulations that address cooling water systems for existing facilities, which would be applicable to STP Units 3 and 4. EPA issued those regulations in 2014.
Please describe the status of this issue.

NINA Response: The EPA recently released Clean Water Act (CWA) regulations regarding Section 316(b). The final rule does not apply to the new facilities currently being licensed by NINA at the STP site.

The new 316(b) rule does apply to the existing Reservoir Makeup Pumping Facility (RMPF) operated by the STPNOC that withdraws water from the Colorado River and conveys it to the MCR where it is used for cooling purposes in a closed-cycle recirculating system. This is a regulated activity under CWA § 316(b) and STPNOC has operated, and will continue to operate, the RMPF in full compliance with applicable regulations under CWA § 316(b). While currently compliant with impingement requirements under the new rule, the facility may be required to demonstrate compliance with entrainment requirements of the rule.

The State of Texas, through the Texas Commission on Environmental Quality (TCEQ), has recently met with existing-facility stakeholders to discuss implementation of the new 316(b) regulations. The rule will be applied through the existing Texas Pollutant Discharge Elimination System (TPDES) permit program as administered by TCEQ. While the regulatory program has yet to be implemented, the rule requires that TCEQ establish site-specific Best Technology Available (BTA) entrainment requirements for existing facilities on a case-by-case basis. Existing facilities are required to develop and submit an Entrainment Characterization Study for use by the State in establishing these site-specific BTA. However, since the RMPF fills the MCR (which meets the definition of closed-cycle recirculating system), this requirement can be reduced or waived by the State because the facility has already minimized adverse environmental impacts.

Question 67: Have there been any new threatened or endangered species listed by State or Federal entities since the FEIS was completed in 2011? If so, how have they been or will they be addressed?

NINA Response: Since the completion of the FEIS in 2011, the Rufa Red Knot (*Calidris canutus rufa*) has been listed as federally "Threatened" by the U.S. Fish & Wildlife Service. In Matagorda County, this species has only been sighted at survey locations near the coast associated with tidal flats and beaches (which are its preferred foraging habitat). It has never been identified at the STP site because the MCR, as well as other water bodies on site, constitute poor habitat for its use. Consequently, the potential for occurrence on site is very low with no impacts anticipated during the construction or operation of the existing or new facilities. Due to the low probability of occurrence on site, no special protective measures are warranted.

Question 69: In February 2010, the State of Texas waived its authority under Title 30, Texas Administrative Code, Chapter 279.2(b)(4) (ML100500926) to act on STPNOC's request for water quality certification. Please explain how this action meets the requirement to have the Clean Water Act § 401 certification required for the NRC to issue a license.

NINA Response: As provided in CWA § 401: "If the State, interstate agency, or Administrator, as the case may be, fails or refuses to act on a request for certification, within a reasonable period of time (which shall not exceed one year) after receipt of such request, the certification requirements of this subsection shall be waived with respect to such Federal application." Thus, a waiver does comply with the CWA, and enables the NRC to issue the COLs.

By waiving water quality certification under CWA § 401, the TCEQ acknowledged that there would be no direct water quality impacts associated with the NRC's federal action of simply issuing a license. Impacts to water quality would occur during specific activities associated with the construction and operation of the new facilities (authorized by that license); however each of these activities would be regulated under separate and specific permits required to perform the activity. For example, water quality impacts associated with dredge or fill activities would be regulated under permits issued by the USACE under CWA § 404. Impacts associated with discharge of wastewater and stormwater would be regulated under the National Pollutant Discharge Elimination System (NPDES) program authorized through CWA § 402. In each instance, the TCEQ would issue a water quality certification prior to permit issuance.

CERTIFICATE OF WITNESS

I certify that NINA's responses to the Commission's questions were prepared by me or under my direction; that the responses are true and correct to the best of my information, knowledge and belief; and that I adopt these responses as part of my sworn testimony in this proceeding.

Executed in Accord with 10 CFR § 2.304(d)

/s/ Scott M. Head
Scott M. Head

DECLARATION OF WITNESS

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

Executed on November 12, 2015.

Executed in Accord with 10 CFR § 2.304(d)

/s/ Scott M. Head
Scott M. Head
Manager, Regulatory Affairs
Nuclear Innovation North America LLC
122 West Way, Suite 405
Lake Jackson, TX 77566
Phone: 979.316.3011
E-mail: SMHead@ninallc.net

Respectfully submitted,

Executed in Accord with 10 C.F.R. § 2.304(d)

/s/ *Steven P. Frantz*
Steven P. Frantz
John E. Matthews
Stephen J. Burdick
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Avenue, N.W.
Washington, D.C. 20004
Phone: 202-739-3000
Fax: 202-739-3001
E-mail: sfrantz@morganlewis.com

Counsel for Nuclear Innovation North America LLC

Dated in Washington, D.C.
this 12th day of November 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	Docket Nos. 52-012-COL
	)	52-013-COL
NUCLEAR INNOVATION NORTH AMERICA LLC	)	
	)	
(South Texas Project Units 3 and 4)	)	November 12, 2015
	)	

CERTIFICATE OF SERVICE

I hereby certify that on this date a copy of the "Applicants' Responses to Commission's Pre-Hearing Questions" was submitted through the NRC's E-filing system.

Signed (electronically) by Steven P. Frantz
Steven P. Frantz
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Avenue, N.W.
Washington, D.C. 20004
Phone: 202-739-5460
Fax: 202-739-3001
E-mail: sfrantz@morganlewis.com

Counsel for Nuclear Innovation North America LLC

DB1/ 85087303.5