UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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| In the Matter of<br><br>NUCLEAR INNOVATION NORTH AMERICA, LLC<br><br>(South Texas Project Units 3 and 4) | Docket Nos. 52-012-COL<br>52-013-COL |

**ORDER**

**(Transmitting Pre-Hearing Questions)**

On October 13, 2015, the Commission issued a notice that it would convene an evidentiary hearing at its Rockville, Maryland headquarters on November 19, 2015, pursuant to section 189a. of the Atomic Energy Act of 1954, as amended, to receive testimony and exhibits in the uncontested portion of the captioned proceeding.[1] In connection with that hearing, pursuant to my authority under 10 C.F.R. § 2.346(a) and (j), Nuclear Innovation North America LLC (NINA) and the NRC Staff should file written responses to the questions provided in the table below. Responses should be filed by **October 29, 2015**.

[1] *See* In the Matter of Nuclear Innovation North America LLC, Combined Licenses for South Texas Project, Units 3 and 4; Notice of Hearing, 80 Fed. Reg. 61,492 (Oct. 13, 2015).

**Table of Questions**

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| 1 | Safety | General | Staff | There are a number of departures from the ABWR certified design and exemptions from NRC regulations. What consideration did the Staff give to any cumulative impacts from these departures and exemptions? |
| 2 | General | 10 C.F.R. § 52.41 | Staff<br>Applicant | 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 design.<br><br>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? |
| 3 | General | License | Staff<br>Applicant | 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 update after issuance of the license.<br><br>If the required information can be provided with the first FSAR update, why can it not be provided now?<br><br>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. |

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| | | | | 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). |
| 4 | General | License<br><br>10 C.F.R. § 73.54 | Staff<br>Applicant | 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.<br><br>Please describe whether the Staff has found that all of the cyber security requirements necessary for licensing are met at this time.<br><br>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). |
| 5 | General | FSAR, Ch.1:<br>§ 1.4 | Applicant | NINA would be the licensee responsible for design and construction of STP Units 3 and 4. STPNOC 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).<br><br>Describe how the transition to operation will take place. Will there be duplicate programs running in parallel, such as the corrective action program? |
| 6 | Safety | SER, Ch. 1:<br>§ 1.4S<br><br>10 C.F.R. § 52.73, | Staff<br>Applicant | 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. |

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| | | Part 52, Appendix A.III | | 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. |
| 7 | General | FSAR Rev. 12 Tier 2, Section 1.4.4.3 and 1.4.4.4<br><br>SER, Ch.1:<br>§ 1.4.2 | Staff<br>Applicant | 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.<br><br>a. For NINA: Both Toshiba and Shaw are identified as responsible for the EPC. Please clarify.<br><br>b. For the Staff: Why is Shaw not discussed in the SER? |
| 8 | Safety | SER, Ch. 1:<br>§ 1.11S.5 | Staff | NINA requested an exemption from the current financial qualification requirements in 10 C.F.R. §§ 52.77, 50.33(f), and Part 50, Appendix C, to allow the use of a financial qualification standard similar to that in 10 C.F.R. Part 70, in accordance with the Staff Requirements Memorandum on SECY-13-0124, "Policy Options for Merchant (Non-Electric Utility) Plant Financial Qualifications," and the Draft Regulatory Basis for the proposed financial qualifications rulemaking. With its exemption request, the applicant addressed the standards in 10 C.F.R. §§ 52.7 and 50.12 and submitted an Applicant Financial Capacity Plan with proposed license conditions. The NRC Staff concluded that NINA's proposed license conditions (with minor revisions) meet the intent of the Draft Regulatory Basis. Please describe what actions would be required by the prospective licensee to satisfy the proposed financial qualification license conditions. |
| 9 | Safety | SER, Ch. 1:<br>§ 1.5S | Staff<br>Applicant | 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 |

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| | | | | 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... ."<br><br>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? |
| 10 | Safety | SER, Ch.1, Ch. 6, Ch. 19 | Staff Applicant | 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.<br><br>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? |
| 11 | Safety | SER, Ch. 1 | Staff Applicant | 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 |

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| | | | | certifications issued after October 16, 2003. The ABWR was certified in 1997.<br><br>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). |
| 12 | Safety | SER, Ch. 6:<br>§ 6.2.1.4 | Staff<br>Applicant | 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? |
| 13 | Safety | SER, Ch. 1, Ch. 6, Ch. 7<br><br>10 C.F.R.<br>§ 50.44(c)(4)<br><br>RG 1.97 | Staff | In § 1.11S.1.3 of the SER, regarding STD DEP T1 2.14-1, the NRC staff states:<br><br>The containment hydrogen and oxygen monitoring functions of the Containment Monitoring System are no longer required to function for the mitigation of a design basis LOCA [loss of coolant accident]. Consequently, the containment hydrogen and oxygen monitoring functions are no longer classified as Category 1, as defined in Regulatory Guide (RG) 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Revision 4. The RG 1.97 classification of containment hydrogen and oxygen |

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| | | | | monitoring functions are changed to Category 3 for hydrogen monitoring, and Category 2 for oxygen monitoring, allowing these instruments to be reclassified as nonsafety-related.<br><br>Regulatory Guide 1.97, Revision 4 (June 2006), removed the terminology of Categories 1, 2, and 3 and instead provided performance-based criteria for use in selecting variables.<br><br>a. Please explain how the Staff applied Regulatory Guide 1.97.<br><br>b. Please explain how the Staff determined NINA's proposed hydrogen and oxygen monitors meet the criteria of 10 C.F.R. §§ 50.44 (c)(4)(i) and (c)(4)(ii).<br><br>c. Further, please explain the Staff's acceptance of the deletion of the words "during accident conditions" in reference to the monitors for oxygen levels in FSAR § 6.2.5.2.1. |
| 14 | Safety | SER, Ch. 2: § 2.2S.3<br><br>FSAR Table 2.2S-10<br><br>RG 1.91 | Staff Applicant | 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? |
| 15 | Safety | SER, Ch. 2: § 2.3S.1.4.3.2<br><br>NUREG/CR-4461 | Staff Applicant | 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? |

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| | | | | 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 Response 03.05.01.06-1). Why is it reasonable to use the 200 ft characteristic dimension provided in NUREG/CR-4461? |
| 16 | Safety | SER, Ch. 2: § 2.5S.1.4.2 | Staff Applicant | 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.<br><br>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?<br><br>For the Staff: Does the recent extraction of hydrocarbons in the Eagle Ford Shale change the Staff's assessment of Effects of Human Activities (under Site Engineering Geology Evaluation in Section 2.5S.1.4.2 Site Area Geology)? |
| 17 | Safety | SER, Ch. 2: § 2.5S.2.4.4 | Staff | As noted in Section 2.5S.2.4.4 of the SER, NINA compared the results of probabilistic seismic hazard analysis (PSHA) calculations for two of six Electric Power Research Institute-Seismicity Owners Group (EPRI-SOG) earth science teams (ESTs) source models as part of its PSHA software validation. The Staff issued RAI 02.05.02-11 regarding NINA's PSHA software validation. In response, NINA provided a comparison of calculations for all six EPRI-SOG ESTs. The SER notes significant differences in the validation results between the original 1989 EPRI-SOG PSHA calculation and NINA's PSHA calculation used for the combined license application for all the ESTs except for the Bechtel and Weston |

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| | | | | ESTs. Describe the process used by the Staff to review the validation results for the PSHA software. |
| 18 | Safety | SER, Ch. 2: § 2.5S4.4.8<br><br>FSAR § 2.5S.4.8 | Staff Applicant | 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.<br><br>a. Why was volumetric strain evaluated for only soil layers with a factor of safety less than 1.4?<br><br>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?<br><br>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? |
| 19 | Safety | SER, Ch. 2: § 2.5S.4.2.8 | Staff Applicant | 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 |

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| | | | | of this peer-reviewed research, explain your conclusion that there is no liquefaction potential for the clayey strata at the STP site. |
| 20 | Safety | SER, Ch. 3:<br>§ 3.3.2 | Staff<br>Applicant | 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 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. |
| 21 | Safety | SER, Ch. 3:<br>§ 3.5.1.6.4<br><br>FSAR § 2.2S.2.7.2<br><br>RG 1.206 | Applicant | 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. |
| 22 | Safety | SER, Ch. 3:<br>§ 3.7 | Staff<br>Applicant | 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 non-conservative. 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. |

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| 23 | Safety | SER, Ch. 3: § 3.7.2.4.19 | Staff Applicant | 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? |
| 24 | Safety | SER, Ch. 3: § 3.8.4.4.1 | Staff Applicant | 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.<br><br>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?<br><br>b. Why was it not necessary for NINA to take a departure from the standard design for the site-specific lateral seismic earth pressures? |
| 25 | Safety | SER, Ch. 3: § 3.8.4.4.5 | Staff Applicant | 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. |
| 26 | Safety | SER, Ch. 3: § 3.9.2.4.1 | Staff | In SER Section 3.9.2.4.1, the Staff indicated that it questioned the applicability of the Kashiwazaki-Kariwa Plant (referred to as RJABWR in the SER) as a prototype plant for demonstrating the design of reactor |

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| | | | | internals against flow-induced vibration. However, the Staff evidently accepted the operating experience of the RJABWR steam dryer as evidence of safe design of the STP steam dryer (see, for example, first paragraph in Section 3.9.2.4.1.1.2). What were the reasons behind the Staff's initial skepticism of using the Japanese plant as a prototype? Why did the Staff ultimately conclude that the Japanese plant was an appropriate prototype? |
| 27 | Safety | SER, Ch. 5: § 5.3.1<br><br>10 C.F.R. Part 50, Appendix H | Staff Applicant | What edition(s) of ASTM E-185, and the ASTM standards referenced therein, will be used to demonstrate compliance with Appendix H requirements? |
| 28 | Safety | SER, Ch. 5: § 5.3.1<br><br>10 C.F.R. Part 50, Appendix H | Staff | Did the Staff review the technical report, "Reactor Pressure Vessel Material Surveillance Program," Toshiba Corp., Apr. 2009 (UTLR-0003, Rev. 0), referenced in section 5.3.5 of FSAR, Rev. 12, to determine if NINA's surveillance program complies with Appendix H and ASTM requirements referenced therein? If so, what did the Staff conclude? |
| 29 | Safety | SER, Ch. 5<br><br>10 C.F.R. Part 50, Appendix H | Staff | Did the Staff evaluate the dosimetry measurement criteria proposed in the combined license application that are based solely on measurements taken from dosimeters located outside of the reactor vessel surveillance capsules? If so, discuss the review, in view of ASTM E-185-82. That standard requires that a full set of neutron dosimeters be included inside of the surveillance capsules and performance of dosimetry measurements of these dosimeters when the capsules are removed in accordance with the reactor vessel surveillance capsule withdrawal schedule. |

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| 30 | Safety | SER, Ch. 5:<br>§ 5.3.1 | Staff<br>Applicant | 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? |
| 31 | Safety | SER, Ch. 5:<br>§ 5.3.1<br><br>10 C.F.R. Part 50, Appendix G | Staff<br>Applicant | 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? |
| 32 | Safety | SER, Ch. 5 | Staff<br>Applicant | 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."<br><br>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 |

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| | | | | 10 C.F.R. § 50.55a that require Staff approval before issuance of a combined license?<br><br>For the Staff: How and where does the SER evaluate NINA's response to COL Action Item 14.1.3.3.2.1-1? |
| 33 | Safety | SER, Ch. 6<br><br>Order EA-13-109 | Staff<br>Applicant | 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.<br><br>a. Is the COPS opening pressure low enough to prevent significant hydrogen leakage into the secondary containment?<br><br>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? |
| 34 | Safety | SER, Ch. 6<br><br>FSAR, § 6C.1<br><br>10 C.F.R.<br>§ 50.46(b)(5)<br><br>RG 1.82 | Staff<br>Applicant | 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). |

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| 35 | Safety | SER, Ch. 6:<br>§ 6.2.1.4<br><br>10 C.F.R.<br>§ 50.46(b)(5) | Staff<br>Applicant | 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 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). |
| 36 | Safety | SER, Ch. 6<br><br>FSAR, § 6C.5<br><br>RG 1.82<br><br>NUREG/CR-6224<br><br>NUREG/CR-6808 | Staff<br>Applicant | 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."<br><br>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 |

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| | | | | 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."<br><br>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. |
| 37 | Safety | SER, Ch. 7:<br>§ 7.9S.4<br><br>RAI 07.09-5<br><br>Branch Technical Position (BTP) 7-21 | Staff<br>Applicant | 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.<br><br>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- |

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| | | | | 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."<br><br>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 . . . ."<br><br>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?<br>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 actuation system functions) in the required time assumed in the safety analysis? |
| 38 | Safety | SER, Ch. 7<br><br>Branch Technical Position (BTP) 7-21 | Staff Applicant | 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., |

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| | | | | 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.<br><br>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? |
| 39 | Safety | SER, Ch. 7 | Staff<br>Applicant | 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. |
| 40 | Safety | SER, Ch. 8:<br>§ 8.3.1.4 | Staff | In FSAR Subsection 8.3.1.1.4.1, NINA discusses the design of the 120 VAC Class 1E instrument power system and provides the following information:<br><br>Individual regulating transformers supply 120 VAC to the four divisions of instrument power (Figure 8.3-2). Each Class 1E divisional transformer is supplied from a 480V MCC in the same division, except for the Division IV transformer, which is supplied from the 480V MCC of Division II. There are three divisions (I, II, and III), each backed up by its associated divisional diesel generator as the source when offsite source is lost. Division IV is backed up by the Division II diesel generator, when the offsite source is lost. Power is distributed to the individual loads from distribution panels, and to logic level circuits through the control room logic panels. Transformers are sized to supply their respective distribution panel instrumentation and control loads. |

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| | | | | Departure STD DEP T1 2.12-2, I&C Power Divisions, provides for a fourth instrument power division as opposed to the three divisions in the ABWR certified design. The fourth division is not a separate entity from the original three. The fourth is also powered by Division II. Whenever two redundant and independent entities are powered from the same division, the concern arises as to the possibility of common cause failure of both entities. Discuss how the Staff's single failure analysis for this aspect of the design took into account potential common cause mechanisms. |
| 41 | Safety | SER, Ch. 9: § 9.1.1.4<br><br>SRP § 3.2.1<br><br>RG 1.29 | Staff | SER Section 9.1.1.4 (COL License Information Item 9.5) states that NINA provided a new diagram to demonstrate that the new fuel inspection platform would meet safe shutdown earthquake (SSE) criteria. Does this piece of equipment meet the criteria to be classified as Seismic Category 1, and if so, how did the Staff verify that the equipment would perform its intended function during a SSE event? |
| 42 | Safety | SER, Ch. 9<br><br>FSAR § 9.5.1.1.7<br><br>STD DEP T1 3.4.1 | Staff<br>Applicant | 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? |
| 43 | Safety | SER, Ch. 9: § 9.5.1.5<br><br>FSAR § 9.5.1.1.9 | Staff<br>Applicant | 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? |

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| 44 | Safety | SER, Ch. 10<br><br>ACRS Letter dated 2/19/15<br><br>FSAR, Rev. 12 | Staff<br>Applicant | 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:<br><br>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.<br><br>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? |
| 45 | Safety | SER, Ch. 12<br><br>10 C.F.R.<br>§§ 20.1601 (d), 20.1602 | Staff<br>Applicant | 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.<br><br>How are the 10 C.F.R. §§ 20.1601 and 20.1602 requirements met for radiation protection doors in containment? |
| 46 | Safety | FSAR,<br>§ 13.1.1.1.3 | Applicant | 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? |

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| 47 | Safety | FSAR § 13.1.1.2 | Applicant | 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? |
| 48 | Safety | SER, Ch. 13, § 13.2<br><br>FSAR § 13.2.3.1 | Staff<br>Applicant | 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. |
| 49 | Safety | SER, Ch. 13:<br>§ 13.6 | Staff<br>Applicant | In the draft license, the Staff proposes the following license condition:<br><br>"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)."<br><br>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."<br><br>a. Why is this license condition necessary? Were there similar conditions in previously-issued combined licenses?<br><br>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? |

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| 50 | Safety | SER, Ch. 13:<br>§ 13.8 | Staff<br>Applicant | In the draft license, the Staff proposes the following license condition:<br><br>"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."<br><br>a. Why is this license condition necessary? Were there similar conditions in previously-issued combined licenses?<br><br>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? |
| 51 | Safety | SER, Ch. 17<br><br>FSAR<br>§ 17.4S.1.1.2 | Staff<br>Applicant | 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.<br><br>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.<br><br>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. |

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| 52 | Safety | SER, Ch. 19 Attachment A<br><br>FSAR Part 11 (Mitigative Strategies Report) | Staff Applicant | 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).<br><br>Explain why an action in a Corrective Action Program is being credited for a licensing basis document. |
| 53 | Safety | SER, Ch. 19 Attachment A | Staff Applicant | 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.<br><br>a. Can this issue be evaluated prior to issuance of a combined license?<br><br>b. How does this commitment demonstrate an acceptable strategy in the licensing basis? |
| 54 | Safety | SER, Ch. 16 Tech Specs.<br><br>COLA, Part 4, Tech Specs. | Staff Applicant | Part 4 of the application, Limiting Condition for Operation (LCO) 3.8.11 states:<br><br>"The following AC electrical power sources shall be OPERABLE:<br><br>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;"<br><br>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. |

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| | | | | APPLICABILITY: MODE 4 and MODE 5 with water level in the refueling cavity < 7.0 meters above the reactor pressure vessel flange."<br><br>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.<br><br>Are there surveillance requirements for MODE 4? |
| 55 | Safety | SER, Ch. 2, Ch. 19 | Staff | Please provide more detail as to why NINA's response to the following RAIs resulted in regulatory commitments in the FSAR as opposed to license conditions.<br><br>1. In response to RAI 19-30 dated July 28, 2010 (ML102110184), and as discussed in Appendix 19R of the SER, NINA revised COL FSAR § 2.4S.10, "Flooding Protection Requirements," to state that all watertight doors and hatches are normally closed.<br><br>2. The Staff concluded that the issues associated with Open Item 19-9 (RAI 19.01-31) are resolved based on: the results of the quantitative assessment and sensitivity analyses that satisfy the requirements of 10 C.F.R. § 52.79(d)(1); completion of the specific bulleted requirements (referenced in FSAR § 19.4.6, "Shutdown Risk") in the abnormal operating procedures to address hurricane preparations, which will assure that the risk from hurricanes for STP Units 3 and 4 remains below the Commission goals; and completion of these requirements, which will assure that the STP Units 3 and 4 design has levels of defense-in-depth. |

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| 56 | Safety | SER, Ch. 22 | Staff Applicant | 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:<br><br>> 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.<br><br>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.<br><br>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. |
| 57 | Environmental | FEIS (General) | Staff | NINA proposed several exemptions to otherwise-applicable regulatory requirements in its application. Please discuss how the Staff's environmental review in the FEIS captured the exemptions. |
| 58 | Environmental | FEIS (General) | Staff | The Staff's environmental standard review plan is currently being updated to reflect environmental requirements established after 2007. The FEIS was published February 2011. What, if any, review was done to ensure the FEIS addresses all current requirements? |

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| 59 | Environmental | FEIS (General) | Staff | In SECY-15-0123 (page 30), the Staff describes its process for searching for new and significant information with respect to the FEIS. The Staff notes that it conducted an audit of NINA's process for identifying new and significant information and references an Audit Report from 2015 on this topic. However, the Audit Report notes several follow up actions and unanswered questions. Specifically the audit team intended to conduct further reviews of cumulative impacts analyses, tribal consultations, the Colorado River salt wedge, and whether any additional information provided by NINA required supplementation of the FEIS. Please provide additional updates on these activities. Has the Staff completed its review of whether any of this information constituted new and significant information requiring a supplement to the FEIS? |
| 60 | Environmental | FEIS (General) | Staff | Throughout the FEIS, the Staff frequently references NUREG-1437, the NRC's generic environmental impact statement for nuclear power plant license renewals. Since the publication of the FEIS, the agency has issued a revised version of NUREG-1437. Has the Staff taken steps to ensure that the revisions to NUREG-1437 did not impact the analyses in the FEIS? |
| 61 | Environmental | FEIS (General, Continued Storage) | Staff | If the Staff had directly considered the environmental impacts of spent fuel storage, as described in the Continued Storage GEIS, NUREG-2157, what effect, if any, would that consideration have had on the benefit-cost balance described in Chapter 10, section 6, and the evaluation of alternatives in Chapter 9? |
| 62 | Environmental | FEIS (General) | Staff | Highlight major themes from the comments on the Draft Environmental Impact Statement (DEIS), and generally describe the Staff's responses to those comments. |
| 63 | Environmental | FEIS (General) | Staff Applicant | 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? |

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| 64 | Environmental | FEIS (General) | Staff Applicant | 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. |
| 65 | Environmental | FEIS, USACE permit | Staff Applicant | 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. |
| 66 | Environmental | FEIS, Ch. 3, § 3.2.2.2 | Staff Applicant | 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.<br><br>Please describe the status of this issue. |
| 67 | Environmental | FEIS, Ch. 4, § 4.3.1.3 | Staff Applicant | 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? |

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| 68 | Environmental | FEIS, Ch. 4; § 4.6, Historic and Cultural Resources | Staff | A letter dated June 4, 2008 (ML081610296) describes a commitment by STPNOC (NINA's predecessor) to develop a procedure for discovery of cultural or historical artifacts during construction. Where will this commitment be captured, and how will the Staff ensure the commitments are met? |
| 69 | Environmental | FEIS, Ch. 5: § 5.2, Water-Related Impacts | Staff Applicant | 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. |

IT IS SO ORDERED.

For the Commission

**NRC SEAL**

**/RA/**

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland,
this 16th day of October, 2015.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing **ORDER (Transmitting Pre-Hearing Questions)** have been served upon the following persons by the Electronic Information Exchange.

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[Original signed by Brian Newell ]
Office of the Secretary of the Commission

Dated at Rockville, Maryland
this 16th day of October, 2015