UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON, TX 76011-4511

October 28, 2016

Ken J. Peters, Senior Vice President
and Chief Nuclear Officer
Attention: Regulatory Affairs
TEX Operations Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 – NRC EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORTS 05000445/2016007 and 05000446/2016007

Dear Mr. Peters:

On September 29, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Nuclear Power Plant, Units 1 and 2. On October 19, 2016, the NRC inspectors discussed the final results of this inspection with Mr. T. Hope, Manager, Regulatory Affairs, and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented one finding of very low safety significance (Green) in this report. This finding involved a violation of NRC requirements. Additionally, NRC inspectors documented one Severity Level IV violation with no associated finding. The NRC is treating these violations as non-cited violation (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Comanche Peak Nuclear Power Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at the Comanche Peak Nuclear Power Plant.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's "Rules of Practice and Procedure," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Enclosure:
Inspection Report 05000445/2016007 and
05000446/2016007 w/Attachment:
Supplemental Information

cc w/encl: Electronic Distribution

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice and Procedure," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Enclosure:
Inspection Report 05000445/2016007 and
05000446/2016007 w/Attachment:
Supplemental Information

cc w/encl: Electronic Distribution

Distribution:
See next page

ADAMS ACCESSION NUMBER: ML16302A456

☒ SUNSI Review By: GGeorge		ADAMS ☒ Yes ☐ No	☒ Publicly Available ☐ Non-Publicly Available		☒ Non-Sensitive ☐ Sensitive		Keyword: NRC-002
OFFICE	SRI:EB1	SRI:EB1	RI:EB2	C:EB1	C:PBA	C:EB1	
NAME	GGeorge	RLatta	NOkonkwo	TFarnholtz	JGroom	TFarnholtz	
SIGNATURE	/RA/	/RA/	E-mail	/RA/	/RA/	/RA/	
DATE	10/19/16	10/19/16	10/18/16	10/28/16	10/20/16	10/28/16	

OFFICIAL RECORD COPY

Letter to Ken J. Peters from Thomas R. Farnholtz, dated October 28, 2016

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 – NRC EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORTS 05000445/2016007 and 05000446/2016007

Electronic Distribution by RIV:
Regional Administrator (Kriss.Kennedy@nrc.gov)
Deputy Regional Administrator (Scott.Morris@nrc.gov)
DRP Director (Troy.Pruett@nrc.gov)
DRP Deputy Director (Ryan.Lantz@nrc.gov)
DRS Director (Anton.Vegel@nrc.gov)
DRS Deputy Director (Jeff.Clark@nrc.gov)
Senior Resident Inspector (Jeffrey.Josey@nrc.gov)
Resident Inspector (Rayomand.Kumana@nrc.gov)
Branch Chief, DRP/A (Jeremy.Groom@nrc.gov)
Senior Project Engineer, DRP/A (Ryan.Alexander@nrc.gov)
Project Engineer, DRP/A (Thomas.Sullivan@nrc.gov)
Project Engineer, DRP/A (Matthew.Kirk@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Project Manager (Margaret.Watford@nrc.gov)
Team Leader, DRS/IPAT (Thomas.Hipschman@nrc.gov)
Project Engineer, DRS/IPAT (Eduardo.Uribe@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
ACES (R4Enforcement.Resource@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
RIV Congressional Affairs Officer (Angel.Moreno@nrc.gov)
RIV/ETA: OEDO (Jeremy.Bowen@nrc.gov)
RIV RSLO (Bill.Maier@nrc.gov)

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	05000445, 05000446
Licenses:	NPF-87; NPF-89
Reports:	05000445/2016007; 05000446/2016007
Licensee:	TEX Operations Company LLC
Facility:	Comanche Peak Nuclear Power Plant, Units 1 and 2
Location:	Glen Rose, Texas
Onsite Inspection Dates:	September 12 through September 29, 2016
Inspectors:	G. George, Senior Reactor Inspector, Engineering Branch 1, Lead R. Latta, Senior Reactor Inspector, Engineering Branch 1 N. Okonkwo, Reactor Inspector, Engineering Branch 2
Approved By:	Thomas R. Farnholtz Chief, Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000445/2016007; 05000446/2016007; 09/12/2016 – 09/29/2016; Comanche Peak Nuclear Power Plant; Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications.

This report covers a two-week announced baseline inspection on evaluations of changes, tests, and experiments and permanent plant modifications. The inspection was conducted by Region IV based engineering inspectors. One Green finding was identified by the inspectors. One Severity Level IV violation was identified by the inspectors. The findings were considered non-cited violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Aspects Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated August 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6, dated July 2016.

A. NRC-Identified Findings and Self-Revealed Findings

Cornerstone: None

- SL-IV. The inspectors identified a Severity Level IV violation of 10 CFR 50.71(e) which states, in part, that the licensee "shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. The submittal shall include the effects of all changes made in the facility or procedures as described in the final safety analysis report, or all safety analyses and evaluation performed by the licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with 10 CFR 50.59 (c)(2)." Specifically, from October 9, 2012, to September 29, 2016, the licensee did not include the effects of changes to the K300 voltage relay setpoint or the safety evaluation in submittals to the Final Safety Analysis Report, Section 8.3.1.1.11, that supported the conclusion that the changes did not require a license amendment. In response to this issue, the licensee planned a corrective action to initiate a licensing document change request to update the final safety analysis report. This finding was entered into the licensee's corrective action program as Condition Report CR-2016-008177.

  The inspectors determined that the licensee's failure to initiate a Licensing Document Change Request, in accordance with Procedure STA-116, "Maintenance of CPNPP Licensing Basis Documents, Operating License conditions and Technical Specifications," Revision 14, Instruction 6.1, to update the Final Safety Analysis Report, Section 8.3.1.1.11, for the setpoint revision of the K300 voltage relays was a performance deficiency. In accordance with Inspection Manual Chapter 0612,

Appendix B, “Issue Screening,” dated September 7, 2012, this was determined to be a minor performance deficiency. This violation was evaluated using the traditional enforcement process because it had the potential for impacting the NRC’s ability to perform its regulatory oversight function. The reactor oversight process’s significance determination process does not consider violations that impact the NRC’s regulatory oversight function. This violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d.3 of the NRC Enforcement Policy, dated August 1, 2016. Specifically, the licensee failed to update the final safety analysis report as required by 10 CFR 50.71(e), but the lack of up-to-date information has not resulted in any unacceptable change to the facility or procedures. The inspectors determined that this violation did not have a cross-cutting aspect because traditional enforcement violations are not assessed for cross-cutting aspects. (Section 1R17.2.b)

Cornerstone: Initiating Events

- Green. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, that “measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components.” Specifically, from November 25, 2014, to September 15, 2016, the licensee failed to appropriately evaluate the suitability of polytetrafluoroethylene (PTFE) gaskets in pressure indicator diaphragm assemblies that form the pressure boundary of the chemical and volume control system. In response to this issue, the licensee immediately isolated all affected diaphragm seal assemblies from the safety-related pressure boundary of the chemical and volume control system. This finding was entered into the licensee’s corrective action program as Condition Reports CR-2016-008180 and CR-2016-008215.

  The inspectors determined that the failure to adequately address the suitability of polytetrafluoroethylene (PTFE) gasket material used in an increased ionizing radiation area following an accident, in accordance with 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, in the event of an accident with one percent core damage, the high radiation environment of the centrifugal charging pump rooms would cause degradation to PTFE gaskets in pressure indicator diaphragm assemblies, which would potentially cause an intersystem loss-of-coolant accident through the safety-related chemical and volume control system pressure boundary. In accordance with Inspection Manual Chapter 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 1, “Initiating Events Screening Questions,” the finding screens to a detailed risk evaluation because, after a reasonable assessment of degradation, the finding could have an effect on systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function (e.g., intersystem loss-of-coolant accident). The senior risk analyst determined that the finding was of very low safety significance (Green). This finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation because the licensee failed to thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. [P.2]. (Section 1R17.2.b)

B. <u>Licensee-Identified Violations</u>

No findings of more-than-minor significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R17 Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications (71111.17T)

.1 Evaluations of Changes, Tests, and Experiments

a. Inspection Scope

The inspectors reviewed 9 evaluations performed pursuant to Title 10 of the Code of Federal Regulations (CFR), Part 50, Section 59, to determine whether the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 22 screenings, where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, and experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests, and experiment was resolved;
- the licensee conclusions for evaluations of changes, tests, and experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000. The list of evaluations, screenings, and/or applicability determinations reviewed by the inspectors is included as an Attachment to this report.

This inspection constituted 9 samples of evaluations and 22 samples of screenings and/or applicability determinations as defined in IP 71111.17-04.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed 11 permanent plant modifications that had been installed in the plant during the last three years. This review included in-plant walkdowns for portions of the selected safety-related systems. The modifications were selected based upon risk significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an Attachment to this report.

This inspection constituted 11 permanent plant modification samples as defined in IP 71111.17-04.

.1 Increase the Upper Voltage Limit of the K300 Voltage Relays for Emergency Diesel Generator Voltage

The inspectors reviewed Final Design Authorization FDA-2012-000121-01, implemented to change the allowable setpoints for emergency diesel generator automatic voltage regulator voltage monitor K300 voltage relay to +/-0.05 percent. The original emergency diesel generator automatic voltage regulator voltage monitor setpoints were too restrictive, resulting in the emergency diesel generator transferring from the digital automatic voltage regulator to analog magnetic voltage regulation. This resulted in the diesel being declared "inoperable;" although, the automatic voltage regulator functioned as designed. This change also included updates to the test procedures and the final safety analysis report. The inspectors identified one Severity Level IV non-cited violation, as documented in Section 1R17.2.b.1.

.2 Commercial Grade Dedication of Ashcroft 200 Series Diaphragm Seals

The inspectors reviewed Procurement Document Review Summary (PDRS) 6S446227, implemented to dedicate commercial grade pressure indicator and diaphragm seals for use on the safety-related pressure boundary of the chemical and volume control system. The original diaphragm seals were subject to minor boric acid leakage. The licensee dedicated the Ashcroft 200 series pressure indicator and diaphragm seals as an alternative replacement item for the pressure indicator and diaphragm seals that were originally installed. The procurement document summary also included an evaluation of changes to the capillary and sensing medium used for the pressure indicator. The inspectors identified one Green, non-cited violation as documented in Section 1R17.2.b.2.

.3 Installation of ½-inch Isolation Valve at the Instrument Air Header Branch Connection

The inspectors reviewed Final Design Authorization FDA-2005-004990-01-04, implemented to install a ½-inch isolation valve, 1CI-1814, at the instrument air header branch connection. A new ½-inch isolation valve was installed at the instrument air header to allow for a shorter outage duration for the instrument air system. The installation did not change the normal function of the instrument air system and was therefore acceptable. The inspectors did not identify any significant concerns with the modification.

.4 Update Design Bases Document for Chemical and Volume Control System

The inspectors reviewed Final Design Authorization FDA-2013-000025-02-03, implemented to update Design Bases Document DBD-ME-255, "Chemical and Volume Control System," Section 4.3.1. This update reflected the installation of Westinghouse SHIELD thermal safe shutdown seals in the Unit 2 reactor coolant pumps. The inspectors did not identify any significant concerns with the modification.

.5 Installation of Line Breakers and Relays for Switchyard Breakers 8080/8075 for the 345 KV line to Station Service Transformer 2ST

The inspectors reviewed Final Design Authorization FDA-2010-000125-06, 14, and 20, implemented to provide K300 voltage relay protection for breaker 8075 on the east switchyard bus. This engineering change involved the installation of a breaker on the east 345 KV line that would provide a second feed to transformer 2ST from the east switchyard bus. The engineering change also provided additional K300 voltage relay protection for the 2ST transformer. The inspectors did not identify any significant concerns with the modification.

.6 Beyond-Design Basis External Event (FLEX) Reactor Makeup Water Storage Tank Crosstie

The inspectors reviewed Final Design Authorization FDA-2013-000008-15-03, implemented to provide a 4-inch crosstie between the Unit 1 condensate storage tank auxiliary feedwater piping and the reactor makeup water storage tank piping. This connection established additional means to cope with a postulated beyond-design basis external event. Specifically, the crosstie will allow the water volume in the reactor makeup water storage tank to gravity drain into the auxiliary feedwater system to assist in reactor core cooling and heat removal. The inspectors reviewed the associated design documents, calculations, installation drawings, pipe support calculations and performed a system walkdown. The inspectors did not identify any significant concerns with the modification.

.7 Beyond-Design Basis External Event (FLEX) Reactor Coolant System Primary/Secondary Connections

The inspectors reviewed Final Design Authorization FDA-2013-000008-06-00, implemented to provide Unit 2 primary safety injection system connections to assist in reactor core cooling and volume makeup following the beyond design basis external event with an extended loss of ac power, coincident with a loss of normal access to the ultimate heat sink. The inspectors reviewed the associated design documents, supporting calculations, installation drawings, pipe support calculations, component installation drawings, post installation test results, and performed a system walk down. The inspectors did not identify any significant concerns with the modification.

.8 Install Fire Dampers In Battery Room Walls, Units 1 and 2

The inspectors reviewed Final Design Authorization FDA-2013-000106-01, implemented to install fire dampers in the train "A," "B," and "C" battery rooms for Units 1 and 2. The fire dampers were installed to assure there was adequate transfer/supply air flow to the battery rooms. The inspectors reviewed the associated design documents, work orders, fire damper data sheets, seismic qualification documentation, battery room exhaust fan flow requirements, performed a system walk down, and discussed the design and installation with the cognizant system engineer. The inspectors did not identify any significant concerns with the modification.

.9 Protection of Control Circuit for Pressurizer Power Operated Relief Valves and Atmospheric Relief Valves to Prevent Spurious Operation During Fire Events

The inspectors reviewed Final Design Authorization FDA-2010-000172-74-00, implemented to install Thermo-Lag on sections of the field cable routed between the containment penetration and the control source for pressurizer power operated relief valves 1-PCV-0445A and 1-PCV-0456 and atmospheric relief valves 1-PV-2326 and 1-PV-2328. These modifications were instituted to prevent spurious operation during a postulated fire event due to hot shorts which could cause a primary system transient. The inspectors reviewed the associated design documents, work orders, performed a system walk down, and discussed the design and installation of Thermo-Lag with the cognizant fire protection engineer. The inspectors did not identify any significant concerns with the modification.

.10 Rewire Control Circuits for Emergency Diesel Generator Offsite Power Breakers, Replace Field Cables With Fire Rated Cables and Install Thermo-Lag

The inspectors reviewed Final Design Authorization FDA-2010-000172-55-06, implemented to rewire the control circuits for emergency diesel generator offsite power breakers 2EA1-1 and 2EA1-2. The modification also authorized the replacing of field cables with fire rated components and installing Thermo-Lag to protect the control circuit from a postulated hot short that could cause a spurious closure of the emergency diesel generator breaker or loss of breaker opening function from the hot shutdown panel. The inspectors reviewed the associated design documents, performed a system walk down, and discussed the modification with the cognizant engineering personnel. The inspectors did not identify any significant concerns with the modification.

.11 Rewiring of Solid State Protection System Circuit Boards

The inspectors reviewed Final Design Authorization FDA-2015-000039-01-01, implemented to remove an identified single point vulnerability within the solid state protection system universal logic and safeguards driver circuit boards. This modification rewired the input power to 16 circuit boards in Unit 1. This modification removed the potential of the 48 Vdc and 15 Vdc power inputs shorting together which would cause an inadvertent plant trip, with the failure of a single auctioneering diode. Additionally, jumpers installed to eliminate nuisance indication for 15 Vdc low voltage were removed. The inspectors did not identify any significant concerns with the modification.

b. Findings

.1 Failure to Update Final Safety Analsis Report Section 8.3.1.1.11

Introduction. The inspectors identified a non-cited Severity Level IV violation for the failure to comply with the requirements of 10 CFR 50.71(e). Specifically, the licensee failed to update Section 8.3.1.1.11 of the final safety analysis report following the revision of the settings of emergency diesel generator K300 voltage relay.

Description. The inspectors reviewed the Final Design Authorization FDA-2012-000121 which implemented changes to setpoints of the emergency diesel generator K300 voltage relays, 27-59/1EG1, 27-59/1EG2, 27-59/2EG1, and 27-59/2EG2. This modification ensured that the emergency diesel generator automatic voltage regulator was not unnecessarily disconnected when emergency diesel generator voltage was at or below the Technical Specification 3.8.1 upper limit.

The emergency diesel generator voltage is required to be maintained within Technical Specification 3.8.1 limits of 6480 – 7150 V. Each emergency diesel generator uses a safety-related digital excitation/voltage regulator or automatic voltage regulator. To obtain and provide defense-in-depth and diversity for protection against software common mode failure, the non-digital time delayed K300 voltage relay was installed. This K300 voltage relay also monitors the emergency diesel generator voltage to automatically disconnect the faulty automatic voltage regulator. The controller then operates in isochronous mode with the magnetics portion of the system maintaining the emergency diesel generator voltage within the Technical Specification limits.

A potential existed during surveillance testing that when an emergency diesel generator was paralleled with the grid, grid voltage fluctuations, emergency diesel generator voltage meter accuracy of +/- 2.0 percent, and the effect of the power factor when loading the diesel, would cause the emergency diesel generator voltage to exceed the K300 voltage relay setting voltage limit of 7114 V for greater than 3 seconds. Upon exceeding the voltage limit, the K300 voltage relay would actuate, resulting in the automatic K300 voltage relay being disconnected prematurely. This would result in the emergency diesel generator being declared "inoperable;" although, the automatic voltage regulator performed as designed.

On October 9, 2012, the licensee implemented a modification to revise the K300 under voltage nominal setting to 6447.6 V with a setting range of 6415 – 6480 V, and over voltage nominal setting to 7185.75 V with a setting range of 7150 – 7222 V.

The Comanche Peak Nuclear Power Plant, Final Safety Analysis Report, Section 8.3.1.1.11, states,

> "If Diesel Generator voltage goes outside the desired limits, the digital control system is automatically disconnected. The controller then operates in isochronous mode with the magnetics portion of the system maintaining the Diesel Generator voltage within the Technical Specification limits."

The modification's 10 CFR 50.59 screen 59SC-2012-000121-01-00 recognized that the changes to the K300 voltage relay setpoints safety function would place the emergency diesel generator in range of operation outside of Technical Specification limits, which was adverse to the Final Safety Analysis Report Section 8.3.1.1.11. The licensee then evaluated the change under revised 10 CFR 50.59 evaluation 59EV-2001-001255-01-01, ultimately determining that the change could be made without obtaining a license amendment. However, subsequent to the change, the licensee failed to create a licensing documentation change request to reflect the change in the final safety analysis report. This was a failure to implement Procedure STA-116, "Maintenance of CPNPP Licensing Basis Documents, Operating License conditions and Technical Specifications," Revision 14, Instruction 6.1.

The licensee entered this issue into their corrective action program as Condition Report CR-2016-008177.

Analysis. The inspectors determined that the licensee's failure to initiate a Licensing Document Change Request, in accordance with Procedure STA-116, "Maintenance of CPNPP Licensing Basis Documents, Operating License conditions and Technical Specifications," Revision 14, Instruction 6.1, to update the Final Safety Analysis Report, Section 8.3.1.1.11, for the setpoint revision of the K300 voltage relays was a performance deficiency. In accordance with Inspection Manual Chapter 0612, Appendix B, "Issue Screening," dated September 7, 2012, this was determined to be a minor performance deficiency. This violation was evaluated using the traditional enforcement process because it had the potential for impacting the NRC's ability to perform its regulatory oversight function. The reactor oversight process's significance determination process does not consider violations that impact the NRC's regulatory oversight function. This violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d.3 of the NRC Enforcement Policy, dated August 1, 2016. Specifically, the licensee failed to update the final safety analysis report as required by 10 CFR 50.71(e), but the lack of up-to-date information has not resulted in any unacceptable change to the facility or procedures. The inspectors determined that this violation did not have a cross-cutting aspect because traditional enforcement violations are not assessed for cross-cutting aspects.

Enforcement. The inspectors identified a Severity Level IV violation of 10 CFR 50.71(e) which states, in part, that the licensee "shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. The submittal shall include the effects of all changes made in the facility or procedures as described in the final safety analysis report, or all safety analyses and evaluation performed by the licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with 10 CFR 50.59 (c)(2)." Contrary to the above, from October 9, 2012, to September 29, 2016, the licensee failed to update periodically the final safety analysis report to include the effects of changes made to the facility that did not require a license amendment. Specifically, the licensee did not include the effects of changes to the K300 voltage relay setpoint or the safety evaluation in submittals to the Final Safety Analysis Report, Section 8.3.1.1.11, that supported the conclusion that the changes did not require a license amendment. In response to this issue, the licensee planned a corrective action to initiate a licensing document change request to update the final safety analysis report. This finding was entered into the licensee's corrective action program as Condition Report CR-2016-008177. Because this violation was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000445/2016007-01 and 05000446/2016007-01, "Failure to Update Final Safety Analysis Report Section 8.3.1.1.11."

.2 Failure to Evaluate the Suitability of Teflon Gaskets in a Safety-Related Pressure Boundary

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to establish measures to review for suitability of application of materials used in a safety-related system. Specifically, the licensee failed to appropriately evaluate the suitability of Teflon gaskets in pressure indicator diaphragm assemblies that form the pressure boundary of the chemical and volume control system.

Description. During a September 15, 2016, walkdown of the chemical and volume control system, the inspectors identified a polytetrafluoroethylene (PTFE) (commonly known as Teflon) gasket in the safety-related diaphragm seal assembly of Unit 1 local pressure indicator, 1-PI-0118. The safety function of the diaphragm seal assembly is to maintain the safety-related pressure boundary of the ANSI Safety Class 2 chemical and volume control system. The assembly installed in the plant is an Ashcroft 200 series diaphragm seal assembly.

Polytetrafluoroethylene is not tolerant to the effects of ionizing radiation. Degradation of the PTFE physical properties begins at approximately 1.7 x 10^4 rads and becomes rapidly brittle at 10^7 rads.[1] The licensee calculated the total integrated radiation dose of the centrifugal charging pump rooms as 2.43 x 10^7 rads, assuming 1 percent failed fuel in the reactor. Therefore, the application of the PTFE gasket created possibility of post-accident pressure boundary leakage (or an intersystem loss-of-coolant accident) from the chemical and volume control system.

Following identification of the PTFE gasket in the safety-related diaphragm seal, the licensee entered this condition into the corrective action program as Condition Report CR-2016-008215. During the operability determination, the licensee determined that there was a possibility that PTFE gaskets were installed in Units 1 and 2 in nine additional safety-related pressure indicator diaphragm seal assemblies since initial operation. This determination was based on the assembly vendor, Ashcroft, stating that PTFE gaskets were originally provided with the diaphragm seal assemblies since 1986. The possible affected diaphragm seal assemblies were associated with pressure indicators 1-PI-0118, 1-PI-0119, 1-PI-0186, 1PI-0187, and 1-PI-0188 on Unit 1; and 2-PI-0118, 2-PI-0119, 2-PI-0186, 2-PI-0187, and 2-PI-0188 on Unit 2. The licensee performed a walkdown of both systems and confirmed that PTFE gaskets were installed on the diaphragm seal assembly for 2-PI-0119. However, the licensee could not determine whether PTFE was installed in the other locations. The licensee declared Units 1 and 2 chemical and volume control systems inoperable, entered Technical Specification 3.0.3 for both Units 1 and 2, then isolated the safety-related diaphragm seal assemblies from chemical and volume control system to restore operability.

The PTFE gasket for 1-PI-0118 was approved for procurement as a non-safety-related part on February 21, 2014, under Procurement Document Review Summary 6S44841. This document evaluated the effect of a failure of the gasket on the system's safety-related function. This evaluation states that the gasket is not part of the pressure boundary in accordance with Section III of ASME Code and that the bolted design is the credited barrier to loss of pressure boundary integrity. It also stated that any leakage from the failure of the gasket would be within the plants ability to make up the loss of inventory.

The inspectors determined that this evaluation failed to address the suitability of the PTFE gasket used in an increased ionizing radiation area following an accident. The inspectors determined that the evaluation failed to quantify any leakage past the PTFE gasket if it failed. The inspectors determined that the procurement document misinterpreted and improperly applied the ASME Code to this component because the diaphragm assembly was not a component in the chemical and volume control system's

[1] Hamman, C. L. and Hanks, D. J. "Electrical Insulating Materials and Capacitors," *Radiation Effects Design Handbook*, National Aeronautics and Space Administration, Washington, D.C., Section 3, page 30, July 1971.

ASME Code pressure boundary. Additionally, the evaluation failed to address deviating from the licensee's specification for prohibition on use of PTFE in a radiation area.

Further investigation by the licensee determined that the PTFE gaskets were installed in diaphragm seal assemblies for 1-PI-0118 and 2-PI-0119 on November 24, 2014, and November 5, 2015, respectively. These gaskets were installed to correct increased boric acid leakage through the diaphragm seal assemblies. During the pre-job brief for the installation on pressure indicator 1-PI-0118, the lead technician questioned the use of the PTFE gasket in the application to the diaphragm seal assembly because of a licensee prohibition of PTFE tape in radiation areas. The licensee initiated Condition Report CR 2014-012353 to address the concern and stopped installation of the gasket. The licensee's engineering staff performed an evaluation which accepted the use of the PTFE gasket by "copy and pasting" the previous inadequate procurement document summary evaluation into Evaluation EV-CR-2014-012353. The inspectors determined this was a failure to thoroughly evaluate issues to ensure the resolution addressed the use of PTFE gaskets in a radiation area. This was the most significant contributor to the performance deficiency.

Analysis. The inspectors determined that the failure to adequately address the suitability of polytetrafluoroethylene (PTFE) gasket material used in an increased ionizing radiation area following an accident, in accordance with 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, in the event of an accident with one percent core damage, the high radiation environment of the centrifugal charging pump rooms would cause degradation to PTFE gaskets in pressure indicator diaphragm assemblies, which would potentially cause an intersystem loss-of-coolant accident through the safety-related chemical and volume control system pressure boundary. In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 1, "Initiating Events Screening Questions," the finding screens to a detailed risk evaluation because, after a reasonable assessment of degradation, the finding could have an effect on systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function (e.g. intersystem loss-of-coolant accident).

A senior reactor analyst performed a qualitative detailed risk evaluation. Since the rooms in which the PTFE gaskets were housed would require failed fuel to reach the radiation levels where the PTFE would become brittle and potentially fail, core damage was assumed to have already occurred during any pertinent sequences. Any subsequent failures of the PTFE gaskets would therefore not increase the core damage frequency. The failure of the gaskets would also present a containment bypass flow through the charging system and into the auxiliary building. While this path has a high factor in consideration for the potential increase in large early release frequency, the timing of the gasket failure would be after fuel failure and late in an accident sequence, which qualitatively would not significantly increase large early release frequency. The analyst therefore determined that the finding was of very low safety significance (Green).

This finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation because the licensee failed to thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Specifically, in November 2014, the licensee's engineering department failed to properly evaluate the effects of radiation on the PTFE gasket, as documented in Condition Report CR 2014-012353 [P.2].

Enforcement. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that "measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components." Contrary to the above, from November 25, 2014, to September 15, 2016, the licensee failed to establish measures for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Specifically, the licensee failed to appropriately evaluate the suitability of polytetrafluoroethylene (PTFE) gaskets in pressure indicator diaphragm assemblies that form the pressure boundary of the chemical and volume control system. In response to this issue, the licensee immediately isolated all affected diaphragm seal assemblies from the safety-related pressure boundary of the chemical and volume control system. This finding was entered into the corrective action program as Condition Reports CR-2016-008180 and CR-2016-008215. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000445/2016007-02 and 05000446/2016007-02, "Failure to Evaluate the Suitability of Teflon Gaskets in a Safety-Related Pressure Boundary."

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

.1 Review of Corrective Action Program Documents

a. Inspection Scope

The inspectors reviewed 18 corrective action program documents that identified or were related to the 10 CFR 50.59 program and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations of changes, tests, and experiments. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action program. The list of specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On September 29, 2016, the inspectors presented the preliminary inspection results to Mr. S. Sewell, Director, Organizational Effectiveness, and other members of the licensee's staff. On October 19, 2016, the NRC inspectors discussed the final results of this inspection with Mr. T. Hope, Manager, Regulatory Affairs, and other members of the licensee's staff. The licensee acknowledged the results as presented. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

A. Adeleye, Design Engineering and Analysis
J. Atwood, Design Engineering and Analysis
C. Corley, Manager, Modification Engineering
C. Feist, Design Engineering and Analysis
A. Hall, Outage Manager
J. Hicks, Regulatory Affairs
T. Hope, Manager, Regulatory Affairs
H. Kenyon, Security
D. Klooster, Manager, Design Engineering and Analysis
S. Kyram, Procurement Engineering
E. Lessmann, Manager, Engineering Smart Team
J. Lloyd, Manager, Operations Support
D. Mcgaughey, Director, Nuclear Operations
G. Merka, Regulatory Affairs
J. Mitchell, Quality Control
C. Montgomery, Project Engineering
S. Porter, Design Engineering and Analysis
K. Robinson, Supervisor, Quality Control
S. Sewell, Director, Organizational Effectiveness
B. St. Louis, Operations
M. Stakes, Manager, Maintenance
J. Taylor, Director, Site Engineering
K. Vehstedt, Regulatory Affairs
L. Wandall, Design Engineering and Analysis
B. Zoh, Design Engineering and Analysis

NRC Personnel

J. Josey, Senior Resident Inspector
R. Kumana, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened and Closed

05000445/05000446/2016007-01	NCV	Failure to Update Final Safety Analysis Report Section 8.3.1.1.11 (Section 1R17.2.b.1)
05000445/05000446/2016007-02	NCV	Failure to Evaluate the Suitability of Teflon Gaskets in a Safety-Related Pressure Boundary (Section 1R17.2.b.2)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it unless this is stated in the body of the inspection report.

10 CFR 50.59 Screenings

59SC-2005-004990-01-00	59SC-2010-000125-05-06	59SC-2010-000172-51-02	59SC-2012-000121-01-00	59SC-2012-000130-04-01
59SC-2013-000008-06-00	59SC-2013-000025-01-03	59SC-2013-000155-02-00	59SC-2013-000178-01-00	59SC-2013-000185-01-00
59SC-2013-000217-01-03	59SC-2015-000073-01-00	59SC-2015-000073-01-00	59SC-2015-000078-01-00	59SC-2015-000131-01-00
59SC-2016-000002-01-00	59SC-2016-000025-01-00	59SC-2016-000035-01-00	59SC-2016-000037-01-00	59SC-2016-000046-01-00
59SC-2016-000564-01-00	59SC-2016-002686-01-00			

10 CFR 50.59 Evaluations

59EV-2010-000172-01-02	59EV-2016-001706-08-00	59EV-2014-005590-02-00	59EV-CR-2012-011253-14	59EV-CR-2013-002861-03
59EV-CR-2010-004331-100	59EV-2001-001255-01-02	59EV-CR-2015-000483-03	59EV-2010-000125-01-00	59EV-2013-000025-01-00

Permanent Plant Modifications

Number	Description or Title	Revision
FDA-2005-004990-01-04	Install 1/2" Isolation Valve at the Instrument Air Header Branch Connection Which Will Be Labeled 1CI-1814 For U1 FW Pump Turbine Gland Steam Supply Header Control System	4
FDA-2010-000125-06	Installation of Line Breakers and K300 Voltage Relay for 8080/8075 for the 345KV line to 2ST	1
FDA-2010-000172-55-06	Rewire Control Circuits for Emergency Diesel Generator Offsite Power Breakers, Replace Field Cables With Fire Rated Cables and Install Thermo-Lag	6
FDA-2010-000172-74-00	Protect Control Circuit for Pressurizer Power Operated Relief Valves and Atmospheric Relief Valves to Prevent Spurious Operation During Fire Events	0

Permanent Plant Modifications

Number	Description or Title	Revision
FDA-2012-000121-01	Increase the Upper Voltage Limit of the K-300 Relays for EDG Generator Voltage	1
FDA-2013-000008-06-00	Unit-2 FLEX RCS Makeup Modification	1
FDA-2013-000008-15-03	FLEX Reactor Makeup Water Storage Tank (RMWST) Crosstie	0
FDA-2013-000025-02-03	Update DBD-ME-255 Section 4.3.1 per IAR Tracker	3
FDA-2013-000106-01-00	Install Fire Dampers in Battery Room Walls	3
FDA-2015-000039-01-01	Modify Unit 1 SSPS to Eliminate Single Point Vulnerability as Identified in Westinghouse Technical Bulletin TB-13-7 (VDRT- 4764719)	1
PDRS 6S446227	TSN 473987, Gauge, Pressure, 0 to 3500 PSIG with Diaph Seal & 40 Ft Silicon Filled Capillary	0
PEIDS NEI03262013	Gauge, Pressure, with Diaphragm Seal (Includes Filled Capillary System When Specified) (Pressure Retention Only)	1
PEIDS NEM1029	AVI Model Bailey Controls Positioners	7

Corrective Action Program Documents (Issued)

CR-2016-008167	CR-2016-008169	CR-2016-008171	CR-2016-008176	CR-2016-008177
CR-2016-008180	CR-2016-008215	CR-2016-008249	CR-2016-008375	CR-2016-008485
CR-2016-008516	CR-2016-008541	IR-2016-008121	IR-2016-008167	IR-2016-008169
IR-2016-008171	IR-2016-008562	TR-2016-008140	TR-2016-008182	TR-2016-008347

Corrective Action Program Documents (Reviewed)

CR-2009-000859	CR-2010-006325	CR-2013-001004	CR-2014-012353	CR-2015-000965
CR-2015-000970	CR-2015-001455	CR-2015-001503	CR-2016-002893	CR-2016-005552
CR-2016-007756	CR-2016-007757	CR-2016-007758	CR-2016-007799	CR-2016-007800
CR-2016-007802	CR-2016-007806	CR-2016-007812		

Calculations

Number	Description or Title	Revision
EE-CA-0008-0871	Protective Relay Setting for Safeguards Buses OV/UV Relays and Associated This Delay Relays	17

Specifications

Number	Description or Title	Revision/Date
CPES-I-1018	Installation of Piping/Tubing and Instrumentation	September 7, 2016
CPES-M-1041G	Chemical/Consumable Products	November 15, 2012
Spec Sheet 4610	Specification for Pressure Gauges	31
Spec Sheet 4619	Specification for Diaphragm Seals	17

Procedures

Number	Description or Title	Revision
STA-716	Modification Process	26
STA-707	10CFR50.59 and 10CFR72.48 Reviews	21
ABN-107	Emergency Boration	9
EPG-725	Pressure Testing	4
STA-116	Maintenance of CPNPP Licensing Basis Documents, Operating License Conditions, and Technical Specifications	14
STI-116-01	Initiating and Processing a Change to a Licensing Basis Document	1
STI-422.01	Operability Determination and Functionality Assessment Program	4
ECE-05.01-04	Technical Evaluation of Replacement Items	10
ECE-6.02-02	Engineering Review of Procurement Documents	16
ECE-6.02-03	Critical Characteristics Development	8
ECE-5.02	Specifications	12
ECE-5.08	Design Change Process	2

Drawings

Number	Description or Title	Revision/Date
2323-M1-0223	SK-0003-05-004990-01-02, Rev. 2 - Flow Diagram, Turbine Gland Steam System	9
7OA1092-1	Type 200 EP Diaphragm Seals	October 27, 1975
7OA917	45-Type 1279SS Series Lower Connection Phenol Turret Case Wall Mounted Maxisafe Duragauge	November 27, 1977
8759D75	Pressure Gage	1977
BRP-CS-1-AB-205	Chemical and Volume Control	4
BRP-GS-1-TB-017	SK-0005-05-0004990-01-00, Rev. 0, - Gland Steam	1
BRP-GS-1-TB-018	SK-0002-05-0004990-01-00, Rev. 0, - Gland Steam	1
E1-0010, Sh. B	Common Auxiliary and Control Bldgs, Safeguard 480V MCC's One Line Diagram	42
E1-0014, Sh. B	Service Water Intake Structure and Safeguard 480V MCC's One Line Diagram	9
E1-0014, Sh. C	Service Water Intake Structure and Safeguard 480V MCC's One Line Diagram	9
E1-0043	Traveling Screen 01 TAG CPX-SWTSTS-01	3
E1-2400, Sh. 193	Protective Device Settings, Emergency Diesel Generators	6
E2-2400, Sh. 193	Protective Device Settings, Emergency Diesel Generators	4
I2-2103-08-V-02	Locally Mounted Pressure Instrument	2
M1-0220, Sh. 001A	SK-0006-05-004990-01-01, Rev. 01, Flow diagram, Instrument Air, Turbine Building	27
M1-0223	SK-0002-05-004990-01-00, Rev. 3, - Gland Steam	---
M1-0255	Chemical and Volume Control System Charging and Positive Displacement Pump Trains	25
M1-2223, Sh. 3	SK-0004-05-004990-01-02, Rev. 2, - Instrumentation Control Diagram, Turbine Steam System Chanel 3910/392	8

Drawings

Number	Description or Title	Revision/Date
SK-0001-10-000172-01-00	Flow Diagram Chemical Volume Control System Charging and Positive Displacement Pump Trains	0
SK-0001-13-000008-06-00	Flow Diagram, Chemical and Volume Control System (M2-00255)	14
SK-0001-13-000008-06-00	Chemical Volume and Control System	0
SK-0002-10-000172-74-00	Conduit Support Location – Thermo-Lag Installation	0
SK-0002-13-000008-06-00	Flow Diagram, Safety Injection System (M2-0261)	16
SK-0002-13-000008-06-00	Safety Injection System	0
SK-0002-13-000106-01-03	Battery Room Fire Dampers	---
SK-0003-13-000008-06-00	Safety Injection (BRP-SI-2-SB-073) Sheet 1	4
SK-0003-13-000008-06-00	Safety Injection	0
SK-0003-13-000106-01-00	Battery Room Fire Dampers	0
SK-0004-13-000008-06-00	Safety Injection (BRP-SI-2-SB-073) Sheet 2	1
SK-0004-13-000008-06-00	Chemical and Volume Control (BRP-CS-2-SB-86)	9
SK-0004-13-000008-06-00	Safety Injection	0
SK-0005-13-000008-06-00	Chemical and Volume Control	0
SK-0007-13-000008-06-01	Small Bore Pipe Support (H-CS-2-SB-0860300-5)	1
SK-0008-13-000008-06-01	Small Bore Pipe Support	1
SK-0009-13-000008-06-00	Small Bore Pipe Support	0

Drawings

Number	Description or Title	Revision/Date
SK-0010-13-000008-06-00	Small Bore Pipe Support	0
SK-0011-13-000008-06-00	Chemical and Volume Control	0

Design Basis Documents

Number	Description or Title	Revision
DBD-EE-051	Design Bases Document, Station Auxiliary Distribution System and Connected Loads	41
DBD-EE-041	Design Bases Document, 480V and 120 V AC Electrical Power System	32
DBD-ME-028	Classification of Structures, Systems, and Components	27
DBD-ME-233	Station Service Water System	34
DBD-ME-255	Chemical and Volume Control System	45

Work Orders

3988676	4372240	4541525	4887435	4889306
4889314	4896426	5038276		

Miscellaneous

Number	Description or Title	Revision/Date
	CPNPP - Self Assessment Report on Implementation of 10 CFR 50.59 Program	February 1, 2016
	Permanent Plant Modification 10 CFR 50.59 Targeted Self-Assessment	August 31, 2016
N1979-SEIS-1	Qualification Report for Luminant Generation Co. Fire Dampers	00
TR-Q1403.1	Environmental Test for Motor Circuit Protector	0
VDRT-4817924	Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties	February 3, 2015