UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 7, 2019

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
Braidwood
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 – REGULATORY AUDIT SUMMARY REGARDING LICENSE AMENDMENT REQUEST TO UTILIZE TVEL TVS-K LEAD TEST ASSEMBLIES; (EPID L-2019-LLA-0208)

Dear Mr. Hanson:

By letter dated July 19, 2018, as supplemented by letter dated October 19, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18204A169 and ML18296A288, respectively), Exelon Generation Company, LLC (EGC, licensee) submitted a license amendment request to add a license condition to the Braidwood Station, Units 1 and 2, Operating Licenses, which would authorize the use of up to eight Joint Stock Company TVEL (TVEL), which is the fuel company subsidiary of Rosatom, TVS-K lead test assemblies (LTAs) in non-limiting reactor core locations for operation and evaluation.

As part of its review, the U.S. Nuclear Regulatory Commission (NRC) staff determined that an audit would be the most effective approach to enable the NRC staff to confirm that the conclusions presented in the EGC July 19, and October 19, 2018, letters represent a complete and accurate interpretation of the underlying technical assessments and analyses, and to identify information, if any, needed to be placed on the docket that is necessary for the NRC staff to make a regulatory finding. The audit plan issued to the licensee by e-mail dated February 7, 2019 (ADAMS Accession No. ML19031C845), contains further background and details on the regulatory bases for the audit.

The audit was held in three phases. The first phase was conducted at the Global Nuclear Fuel-America, LLC (GNF-A) facilities in Wilmington, NC, from February 11 to 14, 2019. The second phase was conducted at the Westinghouse Electric Company, LLC offices in Rockville, MD, on March 14, 2019. The third and final phase was conducted at the General Electric – Hitachi (GEH) offices in Washington, DC on April 30, and May 1, 2019. A summary of the audit results is enclosed.

Enclosure 2 transmitted herewith contains sensitive unclassified information. When separated from Enclosure 2, this document is decontrolled

If you have any questions, please contact me at 301-415-6606 or via e-mail at Joel.Wiebe@nrc.gov

Sincerely,

/RA/

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosures:
1. Audit Report (Non-proprietary)
2. Audit Report (Official Use Only – Proprietary)

cc w/encl 1: via Listserv

ENCLOSURE 1

AUDIT REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING LICENSE AMENDMENT REQUEST TO

UTILIZE TVEL TVS-K LEAD TEST ASSEMBLIES

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

Proprietary information pursuant to
Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390
has been redacted from this document.
Redacted information is identified by blank space enclosed within double brackets
as shown here [[ ]].

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AUDIT REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING LICENSE AMENDMENT REQUEST TO

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DOCKET NOS. 50-456 AND 50-457

1.0 INTRODUCTION

By letter dated July 19, 2018, as supplemented by letter dated October 19, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18204A169 and ML18296A288, respectively), Exelon Generation Company, LLC (EGC, the licensee) submitted a license amendment request to add a license condition to the operating licenses for Braidwood Station, Units 1 and 2 (Braidwood), which would authorize the use of up to eight Joint Stock Company TVEL (TVEL), which is the fuel company subsidiary of Rosatom, TVS-K lead test assemblies (LTAs) in nonlimiting reactor core locations for operation and evaluation.

As part of its review, the U.S. Nuclear Regulatory Commission (NRC) staff determined that an audit would be the most effective approach to enable the NRC staff to confirm that the conclusions presented in the EGC July 19, and October 19, 2018, letters represent a complete and accurate representation of the underlying technical assessments and analyses, and to identify information, if any, needed to be placed on the docket that is necessary for the NRC staff to make a regulatory finding. The audit plan issued to the licensee by e-mail dated February 7, 2019 (ADAMS Accession No. ML19031C845), contains further background and details on the regulatory bases for the audit.

The audit was held in three phases. The first phase was conducted at the Global Nuclear Fuel-Americas, LLC (GNF-A) facilities in Wilmington, NC, from February 11 to 14, 2019, and covered most of the evaluations and analyses performed for the TVEL LTAs. The scope of this audit included:

- the quality assurance controls used to ensure that the TVEL inputs are consistent with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B,
- the fuel rod thermal mechanical methodology and qualification,
- the thermal limit evaluation methodology and critical heat reflux (CHF) correlation,

- the structural analyses for seismic and loss-of-coolant accident (LOCA) events,
- the criticality analyses to support fuel storage, and
- the radiological source term assessments.

The NRC staff who participated in this audit were Scott Krepel (lead reviewer), Jonathan Ortega-Luciano (quality assurance (QA) reviewer), Paul Clifford (thermal mechanical methods reviewer), and Reed Anzalone (thermal hydraulic methods reviewer). Upon completion of this phase, the NRC staff identified some areas where information was not available to allow the NRC staff to complete their audit. Therefore, two follow-up phases were scheduled. The second phase occurred at the Westinghouse Electric Company, LLC offices in Rockville, MD, on March 14, 2019, and focused on the Westinghouse licensing basis methodologies for analyzing fuel assembly structural response during seismic and LOCA events to confirm that the evaluation of the TVEL LTAs is consistent with the licensing basis analyses. The third and final phase occurred at the General Electric-Hitachi (GEH) offices in Washington, DC, and focused on the remaining evaluations and reports for the fuel rod thermal mechanical evaluation and thermal hydraulic evaluation that had not yet been completed at the time of the February audit phase. Note that GNF-A is a subsidiary of GEH. Mr. Krepel and Mr. Clifford were the only NRC staff members supporting the latter two audits.

A full list of the attendees for the February 11 entrance meeting is provided in Section 2.0, though many of the attendees did not participate in the full audit.

2.0 AUDIT ENTRANCE MEETING ATTENDANCE

The participants for the initial entrance meeting, along with their affiliations, are listed below.

Name	Affiliation
Scott Krepel	NRC, Office of Nuclear Reactor Regulation (NRR)/Division of Safety Systems (DSS)/Nuclear Performance and Code Review Branch (SNPB)
Jonathan Ortega-Luciano	NRC, NRR/Division of Inspection and Regional Support (DIRS)/Quality Assurance and Vendor Inspection Branch (IQVB)
Paul Clifford	NRC, NRR/DSS
Reed Anzalone	NRC, NRR/DSS/Reactor Systems Branch (SRXB)
Bob Close	EGC
Rebecca Steinman	EGC
Thomas Rodack	RBH Associates
Brian Mount	Dominion Energy
Amir Vexler	GNF-A
Steve Shelton	GNF-A
Russell Stachowski	GNF-A
Carmen Alonso	GNF-A
Lee Madel-Toner	GNF-A
Scott Pfeffer	GNF-A
John Hannah	GNF-A
Paul Cantonwine	GNF-A

Name	Affiliation
David Pribyl	GNF-A
Gerry Latter	GNF-A
Brandon Schoonmaker	GNF-A
Randy Jacobs	GNF-A
Jesus Diaz-Quiroz	GNF-A
Kevin Ledford	GNF-A
Brian R. Moore	GNF-A
Scott Bowman	GNF-A
Scott C. Swoope	GNF-A
Rich Augi	GNF-A
Michelle Catts	GNF-A
Kent Halac	GNF-A

3.0 REGULATORY AUDIT SUMMARY

The first day of the February 2019, audit primarily consisted of a series of presentations by GNF-A staff to provide an overview and relevant background regarding the licensee's July 19, 2019, request as well as each of the six technical areas that the audit plan identified as specific areas of interest. The remainder of the audits consisted of a combination of document review and discussion sessions with key technical staff from EGC, GNF-A, or Westinghouse, as appropriate. At the end of each audit phase, a brief exit meeting was conducted to summarize the NRC staff's review, any open items, and preliminary requests for additional information (RAIs) that the NRC staff expected to send to the licensee. When all audit phases were complete, the NRC staff determined that some RAIs would be necessary. By NRC e-mail dated May 2, 2019 (ADAMS Accession Nos. ML19128A278 and ML19128A236), the RAIs were transmitted to the licensee.

A full list of all documentation made available to the NRC staff for review during the audits can be found in Section 4.0.

3.1 Quality Assurance Program

EGC and GNF-A presented information about the QA program associated with the TVS-K LTA program. The NRC staff was particularly interested in clarifying how this program worked due to the sharing of responsibility for the various aspects of the LTAs and overall reload licensing for the proposed Braidwood core loading between four different entities (EGC as the licensee, Westinghouse as the licensing methodology vendor, TVEL as the LTA manufacturer, and GNF-A as the contractor performing the safety evaluations for the LTAs).

First, EGC provided some general context. Braidwood has already conducted an LTA program with AREVA so they are familiar with the general requirements for evaluation of LTAs. James Wingfield was appointed as the lead QA auditor of the Russian fabrication facilities, and GNF-A was contracted to provide additional support in verifying that the information provided by TVEL would meet applicable NRC requirements. EGC is the primary party responsible for ensuring that the QA requirements are met, but GNF-A plays a significant role in performing the leg work to confirm this (more discussion on this topic is included below). Westinghouse does not play a role except to specify information that they need to model the TVS-K LTAs within any applicable aspects of their licensing analyses (most notably the structural response analyses for seismic/LOCA events).

EGC also stated that the TK-C69 shipping container certificate had been submitted to the U.S. Department of Transportation (DOT)/NRC for approval in August 2018. Specifically, by letter dated July 27, 2018 (ADAMS Accession No. ML18256A136), GNF-A requested the DOT to revalidate the Russian certificate for the Model TK-C69 shipping container, and by letter dated August 28, 2018 (ADAMS Accession No. ML18256A140), the DOT requested the NRC to review and make a recommendation regarding the DOT revalidation of the TK-C69 for import and export use. At that time, the shipping container was expected to be used for shipment of the fuel to Braidwood and EGC was expecting to obtain revalidation approval prior to shipping the fuel. Prior to the end of the third audit phase, EGC informed the NRC that it was delaying plans to insert the LTAs to spring 2020 because of the delay in obtaining approval for revalidation of the shipping container. Subsequent to the end of the third audit phase, EGC informed the NRC that it was considering other options for shipment of the LTAs.

Next, GNF-A provided a series of presentations describing its involvement in the different aspects of QA. GNF-A stated that TVEL had International Organization for Standardization (ISO) 9001, "Quality Management Systems – Requirements,"[1] certification and an established record as a nuclear fuel manufacturer and that AREVA had previously implemented a QA program at one of the TVEL facilities. TVEL built on this to establish Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B and 10 CFR Part 21, compliant programs for its order entry, gadolinia fabrication, zirconium component fabrication, uranium pellet manufacture, and TVS-K fuel assembly fabrication programs. GNF-A performed an audit of TVEL's quality programs to identify gaps in Appendix B compliance, and will provide surveillance and quality oversight during fabrication, as well as supporting TVEL in any potential Part 21 reportability issues. As part of the auditing and surveillance process, GNF-A and EGC visited TVEL facilities several times and confirmed that TVEL had a strong ISO 17025:2005, "Testing and Calibration Laboratories,"[1] program that was already accredited with an International Laboratory Accreditation Cooperation (ILAC) signatory, with material traceability, knowledgeable personnel, and organized facilities. GNF-A identified some issues, which TVEL is addressing through establishing appropriate controls and internal testing of raw material that was obtained from a supplier that was not Appendix B compliant. GNF-A checked key engineering fields against ASTM International (ASTM) (formerly known as the American Society for Testing and Materials) or other appropriate standards (not Russian standards), and verified that the TVEL QA manual, RKK(USA)-1-2018, in use at all fabrication facilities was adequate.

For the data used to support the safety analyses, GNF-A used a design validation process that leveraged three different approaches to confirm that the information was accurate. The approaches were to: (1) assess the test facility to determine how well the facility is following appropriate standards; (2) corroborate the data from other industry sources; and (3) confirm the data via independent analyses. GNF-A verified all data using at least two of these approaches, and in doing so, brought the information within its own Appendix B compliant program for safety-related calculations. As an example, the critical heat flux (CHF) correlation used in the thermal margin calculation was verified by: (1) confirming that the test facility procedures were appropriate to obtain the desired data, (2) comparing the data to similar data from independent tests performed at the Heat Transfer Research Facility (HTRF) at Columbia University, and (3) independently performing the validation calculations using the VIPRE thermal hydraulic subchannel analysis code.

[1] Available from the International Organization for Standardization at https://www.iso.org/store.html.

3.2 TVS-K LTA Thermal Mechanical Analysis Methodology

3.2.1 E110opt Cladding Material Properties

The NRC staff reviewed the GEH qualification plan and empirical database supporting the E110opt material characterization. A discussion of the NRC staff's review and key observations regarding the E110opt material properties used to support the thermal mechanical evaluation is provided below.

First, the NRC staff reviewed GNF-A calculation 004N5013, Revision 0, "Material Properties Quality Plan." This calculation provides direction under which input information provided by TVEL can be used by GNF-A as design input and was completed in accordance with GEH's approved QA program, NEDO-11209-A (ADAMS Accession No. ML17144A357).

As discussed in Section 3.1, above, GNF-A's process includes assessment, corroboration, and independent confirmation approaches to validate the data for use in safety-related activities. These three approaches were applied to the E110opt material properties through assessment of the test plan, procedures used, equipment calibration, and qualification of technicians; corroboration through available literature or data generated by an independent lab; and confirmation by optional testing and measurements.

The GNF-A documentation provided the "plan" for each of the material properties. Most of the material properties did not include confirmation activities. The following properties were deemed "critical to quality" and considered to be key properties affecting integrity, and thus, independent confirmation was deemed to be necessary by GNF-A:

- Creep
- Stress-strain
- Young's modulus
- Irradiation growth
- High temperature oxidation

During the audit, the NRC staff asked EGC if it had compared Framatome's M5 cladding material properties to those TVEL provided for E110opt. Being nearly chemically identical, and both recrystallized annealed (RXA) alloys, the NRC staff would expect nearly identical properties. EGC replied that it had not compared the material properties.

During the audit, the NRC staff reviewed GNF-A calculation 004N6490, Revision 1, "Evaluation of TVEL Reports on Properties of E110opt, E635, E110opt Models, Qualification Data for Thermal-Mechanical Code, and Experimental Substantiation of the VVER-1000 Fuel Rod Design with Fuel without Central Hole." This calculation describes the TVEL data, comparisons to available data, and sensitivities to parameters such as irradiation, alloying, and corrosion. The following key discussions and conclusions were noted.

- Displacements per atom (dpa) is used as the basis for conversion between different criteria for fast neutron fluence. The conversion between dpa and fast neutron fluence for the BOR-60 test reactor (and others such as the Halden Reactor and the Advanced Test Reactor (ATR)) and neutron energy spectra was evaluated as part of the Nuclear Fuel Industry Research (NFIR) project.

- Electric Power Research Institute, Inc. (EPRI) CA:2009, 1019098[2], "The NFIR-V Dimensional Stability Project: A Method for Transposing Test Reactor Irradiation Data for PWR [pressurized water reactor] and BWR [boiling water reactor] Applications."

- For the BOR-60 test reactor, the conversion was 1 dpa = 0.6×10^{21} neutrons per centimeter squared (n/cm^2) fast neutron fluence with E > 1 million electron volts (MeV). This is the average of calculations for the 40 percent BWR void fraction case performed using the ENDF/B-V, ENDF/B-VI, and ENDF/B-VII dpa cross-section libraries. The conversion for a PWR is 1 dpa = 0.64×10^{21} n/cm^2 fast neutron fluence with E > 1 MeV.

- The TVEL irradiation free growth data for material E110opt and E635 was compared with several other zirconium alloys, including M5® and ZIRLO®. The proposed E110opt growth model (f=0.064) includes a term for accelerated growth at high fluence. The E110opt growth model bounds all the E110opt data and all but 2 data points for E110. There is some discussion that the iron content in more recent E110 alloys has been evolving to delay the onset of accelerated growth, and that E110opt behaves more like the modern E110. The proposed E635 growth model does not include an accelerated component, based on observed growth. The model bounds all the data.

- The Young's modulus, shear modulus, and Poisson's ratio measurements from TVEL were compared against GNF-A's material handbook and MATPRO database. Irradiation effects and impacts associated with temperature, oxygen, coldwork, and crystallography were considered. A comparison of the data shows good agreement with existing handbooks and the MATPRO database.

- The TVEL creep data was reviewed for unirradiated and irradiated E110 and E110opt under internal pressure (tension). TVEL also supplied a creep correlation and creep data for unirradiated and irradiated guide tubes under biaxial compression. Plots comparing the predictions from the correlation and measured data show reasonable agreement.

- TVEL stress-strain data was provided for unirradiated and irradiated conditions. A comparison with data from open literature, including M5 and ZIRLO, shows good agreement.

- TVEL conducted breakaway oxidation tests under LOCA conditions. Early onset of breakaway was shown not to be a concern for the Kroll process (sponge) E110opt.

- TVEL provided high temperature steam oxidation and post quench ductility (PQD) measurements.

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[2] Available from EPRI, 3420 Hillview Avenue, Palo Alto, California 94304.

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- TVEL measured the phase transition temperature for E110opt.
    - [[ ]]
    - [[ ]]
- Several other properties were dispositioned in the calculation, but given time constraints, were not reviewed in depth as part of this audit:
    - Thermal conductivity
    - Oxide conductivity
    - Specific heat
    - Density
    - Thermal expansion
    - Fatigue
    - Corrosion and hydriding
    - Hydride solubility

3.2.2 Fuel Pellet Specification

The NRC staff reviewed TVEL's fuel pellet manufacturing specification to identify and disposition any differences relative to commercial pellets. A discussion of the NRC staff's review and key observations regarding the fuel pellet specifications used to support the thermal mechanical evaluation is provided below.

GNF-A stated that the TVEL fuel pellets are manufactured in accordance with the following ASTM Specifications:

- ASTM C776-17[3], *Standard Specification for Sintered Uranium Dioxide Pellets for Light Water Reactors*
- ASTM C922-14[3], *Standard Specification for Sintered Gadolinium Oxide-Uranium Dioxide Pellets*

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[3] Available from ASTM International, 100 Barr Harbor Drive, PO Box C700, West Conshohocken, PA, 19428-2959 USA

The current approval of the PRIME methodology is limited to "UO_2 and UO_2-Gd_2O_3 fuel pellets with no additives beyond nominal trace elements (ASTM specifications)." During the audit, the NRC staff reviewed GNF-A calculation DBR-0043623, "TVS-K Fuel Pellet Specification Evaluation Against ASTM Specification." The stated objective of this calculation was to "provide a case for concluding the current PRIME models for fuel pellets are applicable to the TVS-K PWR fuel design."

The GNF-A calculation provided a comparison of ASTM to TVEL pellet specifications for both UO_2 and (U,Gd)O2 fuel pellets. The maximum concentrations for all individual impurity elements in the TVEL specifications are at or below ASTM C776 levels. While the total impurity content is not specified in TVEL documents, a prior chemical analysis of TVEL fuel pellets manufactured to the same specification as the fuel loaded in the Ringhals reactor shows total impurity level "far below the ASTM limit" of 1500 micrograms per gram (µg/g) U. The GNF-A calculation concludes that the PRIME03 models and fuel melting temperature are applicable to the TVEL fuel pellets.

GNF-A stated that the TVEL pellet specification for grain size is given as **[[ ]]**. For current commercial fuel designs (UO2 with no additives), grain size is generally 8-15 microns. **[[ ]]**

GNF-A stated that the measured grain size was between **[[ ]]** for the TVEL manufactured pellets. GNF-A believes the grain sizes are reported as **[[ ]]**. GNF-A appears to adjust **[[ ]]**; whereas TVEL may be adjusting **[[ ]]**. GNF-A staff verbally stated that TVEL can achieve **[[ ]]**, but the specification for the TVEL fuel is controlled based on **[[ ]]**. No specified surveillance, sampling, or testing requirements have been identified during the audit.

During the audit, the NRC staff talked with GNF-A personnel involved in the audit of the fuel fabrication facilities in Russia. GNF-A, EGC, and Dominion Energy, all participated in the audit. The audit did not investigate sampling data for pellet quality control, nor did the audit investigate furnace performance. GNF-A personnel noted that TVEL tends to change over their fabrication components more frequently than GNF-A, to retool or improve their processes.

3.2.3 PRIME Qualification

The NRC staff reviewed the PRIME model changes to support modeling of the E110opt cladding, separate-effects and integral testing database, and validation for the TVS-K fuel and cladding. A discussion of the NRC staff's review and key observations regarding the PRIME qualification used to support the thermal mechanical evaluation is provided below.

Section 3.5.1 of the licensee's letter dated July 19, 2018, states the following:

> The TVS-K LTA fuel rod design will be evaluated using a version of the PRIME03 code modified to include the E110opt cladding material characteristics. New material models for E110opt, based upon significant material data including irradiation creep and rod growth, are being implemented in PRIME03.

The licensee's letter dated July 19, 2018, describes the original qualification of PRIME, especially with respect to the PWR fuel rod empirical database:

> Both historical assessments affirm the ability of PRIME to predict PWR fuel behavior, and specific E110opt based integral rod validations will affirm the ability of PRIME to predict fuel performance characteristics, based on the E110opt cladding properties provided by TVEL's cladding experimental basis.

The current NRC approval of the PRIME methodology includes specific limits on the applicability of the methodology. During the audit, the NRC staff compared the range of applicability to the TVS-K design specifications. The TVS-K fuel rod design specification falls within the PRIME applicability range for cladding diameter, thickness, pellet diameter, and other key parameters. PRIME's grain size range is 0.0001 – 0.007 cm (1-70 microns).

During the audit, the NRC staff reviewed GNF-A calculation DBR-0032104, Revision 3, "PRIME03P, Revision 15, Software Requirements Description, Revision 6." The report and underlying engineering software calculations were completed in accordance with GNF-A's Appendix B program. The following observations were noted.

- Fifteen 15 changes were incorporated to expand the flexibility within the code to model E110opt cladding (the same changes can be used to model FeCrAl cladding in other applications).
- Change CR-1976 allows the material model selection and material property coefficients to be input via a CEDAR library file.
- The following model polynomials were updated to accommodate new cladding materials:
  - Young's modulus
  - Specific heat
  - Thermal conductivity
  - Irradiation growth
  - Creep
  - Stress-strain
  - Thermal expansion
  - Poisson's ratio
- The flexibility of the methodology was improved by adding the following inputs:
  - Clad density
  - Fatigue coefficients
  - Fuel pellet density

During the audit, the NRC staff reviewed GNF-A calculation DBR-0004997, Revision 2, "PRIME03P ECP Revision Test Report, Revision 0." The report and underlying engineering software calculations were completed in accordance with GNF-A's 10 CFR Part 50, Appendix B, program. The test report documents a series of regression tests conducted to ensure that the revised PRIME03P computer program still meets product technical requirements, unintended adverse effects were not introduced, the program still produces correct results, and all software changes have been correctly implemented.

During the audit, the NRC staff reviewed GNF-A EH calculation 005N1222, "Braidwood TVS-K LTA Technical Evaluation Report Input: Section A.2 TVS-K PRIME03 Validation Summary." The following observations were noted.

- The PRIME validation database consists of **[[ ]]** experimental rods, including both BWR (Zry-2) and PWR (Zry-4) cladding.

- New experimental validation cases with E110-type cladding were added to the validation database (see Table 1, below).

- Fuel Centerline Temperature (FCT)

    - As shown in Table 1, two separate, multi-rod Halden Reactor irradiated fuel assembly (IFA) test irradiation programs with E110-type cladding measured FCT.

    - The GNF-A calculation states, "A bias in the E110 predictions is observed compared to legacy predictions, but in the conservative direction." Based upon review of predicted and measured FCT data, the NRC staff agrees with this assessment.

    - IFA-676 contained fuel rod segments with different grain size and poison concentrations. Predicted versus measured FCT plots were provided that segregated the data for the **[[ ]]** grain sizes. Examination of this plot reveals a slight under-prediction relative to this limited database.

    - The IFA-700 ramp testing included FCT measurements. Comparison of the predicted and measured data suggests that PRIME conservatively over-predicts fuel temperature during these ramp tests.

- Cladding Axial Deformation

    - As shown in Table 1 below, several Halden Reactor test irradiation programs with E110-type cladding measured cladding axial strain. In addition, post irradiation examination (PIE) were performed on the Ringhals' LTAs.

    - The GNF-A calculation states, "Full length rod measurements are underpredicted, whereas predictions on rodlets more closely align with historical accuracy. It's understood that the underprediction of axial growth in full length rods will have some resultant impact on other fuel rod parameter predictions, but it is judged that these secondary effects are appropriately small considering the adequate predictions of other fuel rod parameters."

        - Axial growth is a small component of the determination of void volume and uncertainties in this prediction have an insignificant impact on rod internal pressure.

  - The PRIME predictions of axial length are not used in design calculations associated with the shoulder gap (differential fuel rod to guide tube growth).

- Cladding Diametral Deformation

  - As shown in Table 1 below, several Halden Reactor test irradiation programs with E110-type cladding measured cladding diametral strain.

  - The GNF-A calculation states, "PRIME predictions of E110 experimental validation cases with steady-state diametral deformations are adequate, as they generally fall in the historic predictive capability of PRIME with other cladding materials." Based on its review of predicted and measured cladding diametral strain data, the NRC staff agrees with this conclusion.

  - As shown in Table 1 below, ramp tests were conducted at the MIR reactor facility on high burnup fuel irradiated at a commercial reactor (NGx-x test series). Eight ramp tests were conducted to identify failure thresholds. In each test, the power instantly ramped to the terminal power and held steady for **[[ ]]**. The residual plastic strain was measured post-test.

    - The predicted and measured data suggests that PRIME underpredicts plastic strain. All the measurements were underpredicted.

      - For example, PRIME predicted **[[ ]]**.
      - For larger measured strains, PRIME predicted strains that were **[[ ]]**.

    - PRIME's underprediction may be an artifact of the experimental protocols **[[ ]]**.
      GNF-A plans to gather more ramp test data to further investigate.

- Fission Gas Release (FGR)

  - As shown in Table 1 below, FGR measurements were taken on the long-term base irradiation rods in IFA-676. Rod average burnup ranged from approximately 58 – 72 Gigawatt days per metric ton of uranium (GWd/MTU) on the 5 FGR measurements.

  - The predicted and measured FGR data suggests that PRIME does a reasonable job (uncertainties are within the span of existing database).

  - The IFA-676 test contained fuel rod segments with different grain size and poison concentrations. Predicted versus measured FCT plots were provided that segregated the data for the **[[ ]]** grain sizes. Examination of this limited data set did not reveal any issues with PRIME's ability to predict FGR.

- Rod Internal Pressure (RIP)
  - As shown in Table 1 below, RIP measurements were taken on the long-term base irradiation rods in IFA-676 and during ramp testing in IFA-700.
  - The predicted and measured RIP data suggests that PRIME does a reasonable job (uncertainties within span of existing database).
  - The IFA-676 test contained fuel rod segments with different grain size and poison concentrations. Predicted versus measured RIP plots were provided that segregated the data for the **[[** **]]** grain sizes. Examination of this limited data set did not reveal any issues with PRIME's ability to predict RIP.

Table 1: TSV-K E110 Empirical Database for PRIME Validation

Rod Name	Experiment	Burnup (GWd/MTU)	Qualification Data				
			Axial Strain	Diametral Strain	FGR	FCT	RIP
IFA-610-11	Creep	51				X	
IFA-650-11	LOCA Burst	56	X				X
IFA-676-1	Base	0 - 60			X	X	X
IFA-676-2			X			X	X
IFA-676-3					X	X	X
IFA-676-4			X		X		
IFA6-76-5						X	
IFA-676-6			X		X	X	
IFA-699-2-L	Creep	Not Specified		X			
IFA-700-1	Ramp	54	X			X	
IFA-700-2						X	X
IFA-700-9			X			X	X
IFA-720.1	Base	55	X		X	X	X
IFA-728	Oxidation	Not Specified	X	X	X		
IFA-741-2	Creep	fresh		X			
NG1-1	Ramp	41	X	X			
NG1-2			X	X			
NG1-3			X	X			
NG5-1	Ramp	55	X	X			
NG5-2			X	X			
NG5-3			X	X			
NG5-4			X	X			
NG5-5			X	X			
Ringhals	Commercial LTA	2nd Cycle	X				

3.2.4 PRIME Fuel Rod Thermal/Mechanical Evaluation

The NRC staff reviewed GNF-A's calculations related to TVS-K fuel rod thermal/mechanical design and to confirm that it satisfies all acceptance criteria. A discussion of the NRC staff's review and key observations regarding the thermal mechanical evaluation is provided below.

During the audit, the NRC staff reviewed GNF-A calculation 005N0786, "Braidwood TVS-K LTA Technical Evaluation Report Input: Section A.3 – PRIME Application Summary." This calculation summarizes the PRIME calculations which support the TVEL LTAs for Cycle 22. The following observations were taken from this calculation, as well as cited underlying GNF-A calculations.

- Rod Internal Pressure Calculation (005N0843, Revision 1)
  - Based on projected rod power histories (not traditional bounding curve) and includes a 5 percent increase.
  - RIP calculated for each fuel rod in the LTA.
  - Limiting rod location 1,1 (inboard corner fuel rod)
  - For Cycle 22, PRIME predicted RIP remains below system pressure for the nominal and 95 percent confidence levels.
    - Cycle 22 maximum (95/95) RIP equals **[[** **]]** below 2250 pounds per square inch (psi) system pressure.
    - Cycle 22 maximum (normal) RIP approximately **[[** **]]**.
  - RIP margin (95/95) **[[** **]]**.
  - No DNB propagation methodology.

- Fuel Handling Accident (FHA) Post-shutdown RIP Calculation (005N0843, Revision 1)
  - Braidwood FHA RIP limit is 1200 psig (pounds per square inch gauge)
  - For end of Cycle 22, PRIME predicted post-shutdown RIP is **[[** **]]**.

- Cladding ovality and creep collapse calculation shows margin to design criteria (005N2546, Revision 0).

- Cladding fatigue calculation shows margin to design criteria (005N0926, Revision 0).

- Anticipated Operational Occurrence (AOO) Power-to-Melt Calculation (005N0844, Revision 1)
  - Power-to-melt as a function of exposure predicted with PRIME (at a 95 percent confidence level) and provided to Westinghouse.

    UO_2 – **[[**

]]

(U,Gd)O_2 - [[

]]

- AOO Cladding Stain Calculation (005N0839, Revision 1)
    - Westinghouse provided AOO overpower scenarios that were used with PRIME to predict incremental cladding strain (worst tolerance basis).
    - PRIME methodology is to report [[

      ]].
    - For Cycle 22, [[
      ]] strain is compared to the 1.0 percent strain limit.
        - Based on the beneficial cladding oxidation (and hydrogen uptake) of E110-type cladding relative to Zry-4 or ZIRLO cladding, the NRC staff believes that the 1.0 percent stain limit is reasonable for this application.
    - [[
      ]]

3.2.5 FRAPCON-4 Confirmatory Calculations

As part of the audit, the NRC staff completed confirmatory design calculations for the TVS-K fuel rod design using the FRAPCON-4 thermal-mechanical fuel rod code. Fuel design specifications were obtained from GNF-A calculation DBR-0041896, Revision 0, "T-M TVS-K Fuel Rod Inputs."

Fuel Rod Internal Pressure

A clear majority of the empirical database used to calibrate and validate FRAPCON-4's fission gas release model is comprised of fuel with grain size ranging from 8-15 microns. As a result, fuel grain size is not an input parameter and has been hard-wired to a value of 10 microns. To help investigate the TVEL fuel rod performance, a modified version of FRAPCON-4 was created that allows a user defined grain size.

The limiting Cycle 22 fuel rod power history was provided by GNF-A and corresponded to the fuel rod located in the corner of the TVS-K assembly (i.e., rod 1,1). A sample calculation was performed to ensure that all the important parameters with respect to RIP were consistent with the TVS-K fuel rod (as modelled in PRIME). The predicted initial, hot void volume in

FRAPCON-4 was slightly smaller than PRIME's predictions; which is in the conservative direction.

FRAPCON-4 calculations were performed with the standard 10 micron grain size and a larger grain size, which bounds that expected in the TVS-K fuel rods. Calculations were also performed at the limiting Cycle 22 power history and at a power history reflecting a 10 percent increase in power at all times. In addition, calculations were performed using best-estimate FGR models and a 2-sigma multiplier on the diffusion coefficient in the FGR model. In total, eight FRAPCON-4 calculations were performed. Table 2 lists the results of these calculations. In all cases, RIP remains below RCS pressure (2250 psia). These independent calculations confirm GNF-A's assertion that the RIP will remain below system pressure throughout Cycle 22.

Power-to-Melt

The local fuel rod power at fuel centerline melt conditions is provided as a function of fuel rod design and exposure. Starting with the expected limiting Cycle 22 fuel rod power profile, the rod power was manually iterated until fuel melting was observed. This exercise was conducted at three burnup steps using nominal melting temperature and a 2-sigma melting temperature. FRAPCON's integral assessment conducted in accordance with NUREG/CR-7022, Volume 2 "FRAPCON-3.5: Integral Assessment," Revision 1[4] determined a 1-sigma uncertainty of 4.7 percent on the model's ability to predict fuel centerline temperature. The 2-sigma melting temperature as a function of burnup is shown in Figure 1.

In total, six FRAPCON-4 predicted power-to-melt local powers were reported. Table 3 lists the predicted local power at melt conditions. Figure 1 graphically depicts the trend in power-to-melt along with the GNF-A PRIME predictions. In all cases, the GNF-A PRIME predictions are more conservative.

Anticipated Operational Occurrence Cladding Strain

GNF-A provided the NRC staff with the limiting Cycle 22 AOO overpower scenario at several burnup steps. FRAPCON-4 calculations were performed at four burnup steps using nominal fuel thermal expansion model and a 2-sigma multiplier on fuel thermal expansion. In total, eight FRAPCON-4 cases were performed. Table 4 lists the predicted **[[** **]]** strain of these calculations. **[[**

]] In all cases, **[[** **]]** strain remains below 1.0 percent. These independent calculations confirm GNF-A's assertion that fuel rod cladding will not fail under AOO overpower conditions.

[4] Available for examination or purchase at the NRC's Public Document Room (PDR), Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville Maryland 20852. The PDR staff may be contacted at 1-800-397-4209, 301-415-4737, or by email to pdr.resource@nrc.gov.

Table 2: FRAPCON-4 Rod Internal Pressure Confirmatory Calculations

Case	Rod Power / Average Burnup	Grain Size	Applied Uncertainties	FRAPCON-4 Prediction
1	EOC 22[1]	[[ ]]	Nominal	[[ ]]
2			2σ FGR	[[ ]]
3		[[ ]]	Nominal	[[ ]]
4			2σ FGR	[[ ]]
5	Cycle 22[2] + 10%	[[ ]]	Nominal	[[ ]]
6			2σ FGR	[[ ]]
7		[[ ]]	Nominal	[[ ]]
8			2σ FGR	[[ ]]

Table Notes:

[1] Highest power rod (1,1) in TVK-S LTA has a projected rod average burnup of 26.4 GWd/MTU.

[2] With +10 percent multiplier on power, projected burnup increases to 29.1 GWd/MTU.

Table 3: FRAPCON-4 Power-to-Melt Confirmatory Calculations

Case	Rod Power / Average Burnup	Applied Uncertainties	FRAPCON-4 Prediction[1]
9	2.2 GWd/MTU	Nominal	[[ ]]
10		2σ FCM	[[ ]]
11	13.5 GWd/MTU	Nominal	[[ ]]
12		2σ FCM	[[ ]]
13	26.4 GWd/MTU	Nominal	[[ ]]
14		2σ FCM	[[ ]]

Table Notes:

[1] Maximum local power density (Fq) at fuel centerline melt conditions.

Table 4: FRAPCON-4 AOO Cladding Strain Confirmatory Calculations

Case	Rod Power / Average Burnup	Applied Uncertainties	FRAPCON-4 Prediction[1]
9	6.7 GWd/MTU	Nominal	[[ ]]
10		2σ Fuel Exp.	[[ ]]
11	13.5 GWd/MTU	Nominal	[[ ]]
12		2σ Fuel Exp	[[ ]]
13	22.5 GWd/MTU	Nominal	[[ ]]
14		2σ Fuel Exp.	[[ ]]
15	26.4 GWd/MTU	Nominal	[[ ]]
16		2σ Fuel Exp.	[[ ]]

Table Notes:

[1] Strain values [[ ]].

[[

]]

Local Burnup	FRAPCON Tmelt	FRAPCON 2-Sigma Tmelt
0 GWd/MTU	[[ ]]°F	[[ ]]°F
10	[[ ]]	[[ ]]
20	[[ ]]	[[ ]]
30	[[ ]]	[[ ]]
40	[[ ]]	[[ ]]
50	[[ ]]	[[ ]]
60	[[ ]]	[[ ]]
70	[[ ]]	[[ ]]

3.2.6 Design-Basis Accident Assessment

The NRC staff reviewed available and applicable LOCA integral testing, available and applicable reactivity initiated accident (RIA) integral testing, and assessed the impact on the accident source terms. A discussion of the NRC staff's review and key observations regarding the expected behavior of the TVS-K fuel and cladding during design basis accidents is provided below.

LOCA Performance:

As discussed in Section 3.2.1, research and testing of E110opt cladding under simulated LOCA conditions has been conducted. Breakaway oxidation test results suggest that E110opt is not susceptible to the same early breakaway oxidation that was earlier observed for electrolytic E110 cladding. PQD test results suggest that residual ductility is preserved up to and beyond the 2200 °F (degree Fahrenheit) and 17 percent equivalent clad reacted (ECR) analytical limits prescribed in 10 CFR 50.46. Note that the 17 percent ECR limit, based on Baker-Just as prescribed in 10 CFR 50, Appendix K, corresponds to approximately 13 percent Cathcart-Pawel (CP)-ECR. Hence, the observed nil ductility threshold of **[[**
]].

Fuel Fragmentation, Relocation, and Dispersion (FFRD) Performance:

Halden performed a series of simulated LOCA tests designed to investigate FFRD. These tests were part of the IFA-650 series. IFA-650-6 and 650-11 were integral LOCA balloon/burst tests conducted on VVER reactor fuel with E110 cladding. Fuel rod segments were previously irradiated in a commercial reactor to approximately 56 GWd/MTU. Information was not available regarding whether these rodlets had annular pellets.

Both IFA-650-6 and -11 exhibited fuel rod balloon and burst under simulated LOCA conditions. Fuel fragmentation was observed, however, dispersal of fragmented fuel particles outside the fuel rod was not reported.

Currently, there are no specific regulatory requirements associated with FFRD but FFRD research and discovery continue. IFA-650-6 and -11 results suggest that the TVEL fuel pellets are no more susceptible to FFRD-related phenomena than other fuel pellets approved for commercial use in the US.

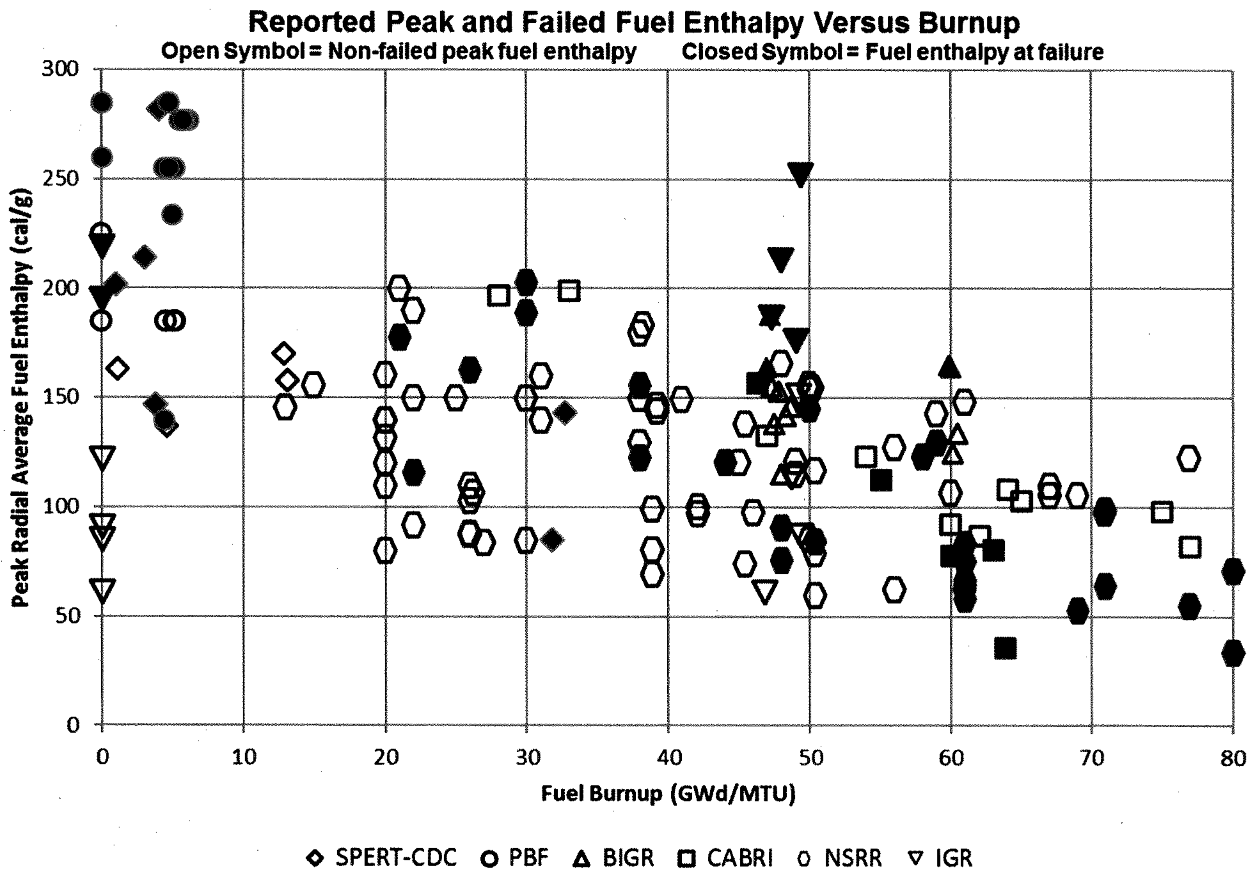
Reactivity Initiated Accident Performance:

Figure 2 depicts the entire reactivity initiated accident (RIA) empirical database and represents decades of testing conducted around the globe. RIA testing conducted on VVER reactor fuel rods with UO_2 fuel pellets and E110 cladding at the Russian BIGR and IGR research facilities are shown as triangles on Figure 2. In total, 26 RIA tests were performed at these Russian facilities. These test results have been deemed applicable by the NRC staff and have been used to generate the Interim RIA Acceptance Criteria and Guidance provided in Standard Review Plan, Chapter 4.2, Appendix B, Revision 4, dated March 2007 (ADAMS Accession No. ML070740002). Hence, E110 cladding is already addressed in existing guidance by virtue of inclusion of E110 testing in the database used to develop the current guidance.

Radiological Source Term:

As described in Section 3.2.2, the TVS-K UO_2 fuel pellets are manufactured to ASTM specifications. There is no evidence which suggests that the chemical composition and species of the various radionuclides would be different from current commercial fuel pellets. Furthermore, given the relative low power of these LTAs, the fractional release of nuclides from the pellet is expected to be small at the end of Cycle 22.

Figure 2: RIA Empirical Database



3.3 Core Thermal Hydraulic Analysis Methodology

The NRC staff reviewed the basis for the assertion that the TVS-K LTAs would be hydraulically compatible with the co-resident VANTAGE+ OFA fuel from Westinghouse. A discussion of the NRC staff's review and key observations regarding the hydraulic compatibility evaluation of the TVS-K fuel is provided below.

The TVS-K fuel contains a smaller flow area and different loss coefficients for the various fuel assembly structural elements (nozzles, spacer grids, etc.). GNF-A performed a thermal hydraulic compatibility evaluation that focused on two main ways that the two fuel assembly design types can affect each other due to variations in their thermal hydraulic performance characteristics.

The first area evaluated was the expected variation in coolant properties as a result of the differences in fuel assembly designs.

- The NRC staff verified that the "K/A^2 approach" described in the licensee's July 19, 2019, letter is essentially just an adjustment to the loss coefficients based on flow areas to obtain a consistent comparison basis by which the impact on coolant flow characteristics could be estimated. The overall flow resistance for the TVS-K fuel assembly is about [[ ]] than the co-resident VANTAGE+ OFA fuel.

- The major coolant flow characteristics—total core flow, pressure drop, flow distribution, and so on—were predicted to change by no more than [[ ]] for the planned loading of 8 TVS-K LTAs in the Braidwood reactor.

- The initial compatibility evaluation was performed using loss coefficients for the TVS-K fuel that was loaded in Ringhals, with a [[ ]] uncertainty associated with measurement and an additional [[ ]] uncertainty to provide additional margin for redesigns. The TVS-K LTA design planned for insertion in Braidwood contains several small redesigns, and the redesigned fuel assembly went through testing that concluded in late 2018. The redesigned fuel assembly was shown to have very similar characteristics, with a total change in loss coefficient of about [[ ]] across the entire fuel assembly. Therefore, the findings from the compatibility evaluation are applicable to the TVS-K LTAs planned for insertion in the Braidwood reactor.

The second area evaluated was the potential for crossflow between adjacent fuel assemblies. The differences in flow area and loss coefficients would serve to drive some crossflow between fuel assemblies, which may affect local thermal hydraulic characteristics.

- Crossflow was evaluated using the same VIPRE model that was used for the thermal margin evaluation. The calculation showed expected trends, with crossflow occurring in the direction of less flow resistance, essentially moving back and forth between the TVS-K and VANTAGE+ OFA fuel assemblies due to variations in flow resistance at the spacer grid locations as well as the difference in flow areas between spacer grids.

- The maximum cross flow was calculated at the TVS-K anti-fretting grid as [[ ]]; this is a somewhat extreme case because the VANTAGE+ OFA fuel assembly does not have a corresponding grid in this location.

- The NRC staff observed that there was no attempt to specifically validate VIPRE for this purpose, for the specific fuel assembly designs under consideration. However, given the reasonableness of the results and the relatively minor differences between the two fuel assembly designs, the general conclusion appears to be reasonable for the purpose of loading 8 TVS-K LTAs in the Braidwood reactor.

3.4 CRK Critical Heat Flux Correlation

The NRC staff reviewed TVSK-DR-EX-25, Revision 1, which provided TVEL's documentation on the CRK CHF correlation for the TVS-K fuel assembly, the supporting experimental testing, and the VIPRE modeling used to develop the correlation. GNF-A reports 004N8457, 004N9369, 004N9516, 0049485 and 004N9475, documented GNF-A's validation of the experimental data and the VIPRE models GNF-A used to develop the final correlation as it will be applied in U.S. reactors. A discussion of the NRC staff's review and key observations regarding the development and validation of the CRK CHF correlation is provided below.

Correlation and Supporting Testing

Tests supporting the development of the CRK CHF correlation were performed at the L-186 test facility at OKBM Afrikantov in Nizhny Novgorod, Russia; the SVD-2 facility at the Institute of Physics and Power Engineering (IPPE) in Obninsk, Russia; and a test facility at the Kurchatov Institute (KI) in Moscow, Russia. Primary testing was performed at L-186 and SVD-2 on the spacer grids originally developed for the TVS-K fuel loaded in the Ringhals reactor. GNF-A compared the data from SVD-2 to data from the former Heat Transfer Research Facility (HTRF) at Columbia University, and found the results to be consistent with the HTRF, or conservative. CHF testing performed at L-186 and SVD-2 was described at a high level in the supplemental information described in the licensee's October 19, 2019, letter and was not audited in further detail.

One of the redesigns for the TVS-K fuel planned for loading in the Braidwood reactor, relative to the TVS-K fuel previously operated in the Ringhals reactor, was a redesign in the spacer grids to add mixing vanes to improve CHF performance near the guide tubes. Testing at KI used the updated spacer grids developed for the TVS-K fuel to be loaded in the Braidwood reactor, and was used to confirm that the CRK CHF correlation based on the L-186 and SVD-2 data was conservative relative to the spacer grids planned for use. GNF-A obtained engineering drawings from TVEL to share with the audit team showing the changes to the spacer grids used in the KI testing. The audit team verified that the spacer grids used in testing incorporated the updates to the spacer grid that were made after the design of the Ringhals LTAs, but prior to insertion of the Braidwood LTAs. Testing was done in pressure ranges of **[[ ]]** psia and mass fluxes between **[[ ]]** Mlbm/hr-ft^2. Results calculated using the CRK CHF correlation developed prior to the spacer grid modifications appear to be conservative.

SVD-2 testing was also used to characterize the thermal diffusion coefficient for use in VIPRE. The tests put highly peaked rods on one side of the test bundle and low peaked rods on the other side. Tests were carried out with and without intermediate flow mixing (IFMs). Coolant temperatures calculated using VIPRE-01 were compared to measured coolant temperatures, and the value of the thermal diffusion coefficient that minimized the error between measured and predicted was selected for use. This process resulted in a thermal diffusion coefficient value of **[[ ]]**, selected as the most conservative (lower) value from the IFM and non-IFM tests.

The CRK CHF correlation as provided in TVSK-DR-EX-25, Revision 1, took the following form:

[[ ]]

Where:

[[

]]

[[

]]

- [[ ]]

  [[

  ]]

A linear least-squares regression was performed against the experimental data to determine the empirical coefficients B_i.

The following table represents the range of applicability of the CRK correlation, based on the test data that was taken:

Parameter	Minimum	Maximum
Pressure (psia)	[[ ]]	[[ ]]
Mass velocity (Mlbm/hr-ft^2)	[[ ]]	[[ ]]
Quality (-)	[[ ]]	[[ ]]
Heated Length (in)	[[ ]]	[[ ]]
Rod diameter (in)	[[ ]]	
Guide tube diameter (in)	[[ ]]	
Rod pitch (in)	[[ ]]	
Hydraulic diameter (in)	[[ ]]	[[ ]]
Axial grid spacing (in)	[[ ]]	[[ ]]

The W-3 correlation is to be applied for TVS-K CHF calculations outside of the CRK CHF correlation applicable range, and in the non-mixing grid portion of the bundle.

3.4.2 TVEL VIPRE Model and Data Treatment

TVEL performed VIPRE calculations in order to validate the CRK CHF correlation. The NRC staff review of the CRK CHF correlation provided greater weight to similar work performed by GNF-A (discussed in Section 3.4.3, below), but the TVEL report provides some useful independent verification of GNF-A's conclusions. A discussion of the NRC staff's review and key observations regarding the TVEL validation of the CRK CHF correlation is provided below.

Calculations were performed in VIPRE-01 MOD-02.6. The modeling methodology used the dummy rod model approach, in which heat flux is a direct input to the coolant. Turbulent friction was calculated using the **[[** **]]**. The nodalization was standard (i.e., each subchannel had its own node), **[[** **]]**.

The ratio of the experimental to calculated CHF was provided in plots versus each parameter. No significant trends were observed.

Outliers were removed from the dataset by doing a z-test where values beyond a critical value were removed. The critical value was based on a percentage uncertainty defined by $\alpha' = 1 - (1-\alpha)^{1/n}$, where α is the original significance level (i.e., 95 percent) and n is the number of points in the dataset. No outliers were found.

Normality tests were also performed for various subsets of the data: one for each test series, one for each of the IFM, non-IFM, 4x4, 5x5, non-guide tube, guide tube, uniform, and non-uniform tests. All the normality tests passed. No significant deviations were observed between the statistics of the subsets.

3.4.3 GNF-A VIPRE Model and Data Treatment

NRC staff reviewed GNF-A Report 004N8457 for details on the VIPRE model used by GNF-A to generate/re-validate the CRK CHF correlation. A discussion of the NRC staff's review and key observations regarding the GNF-A validation of the CRK CHF correlation is provided below.

The GNF-A VIPRE model was very similar to the TVEL model, with some notable differences. **[[** **]]** Axial power shapes were interpolated linearly from the powers given by TVEL from the testing. GNF-A used **[[** **]]**, instead of the **[[** **]]** used by TVEL. The **[[** **]]** was used for turbulent friction, **[[** **]]** at Reynolds numbers of approximately 10^5. Pressure loss coefficients for spacer grids, IFMs, and simple support grids were taken from experimental data. **[[** **]]** based on the testing performed by TVEL at SVD-2 (this value was validated as acceptable by GNF-A in GNF-A Report 004N9369). To implement the CRK CHF correlation in VIPRE, GNF-A took a dynamic linked library containing the correlation and linked it to VIPRE.

The GNF-A report stated that the predicted CHF used to compare to the measurement was that predicted by VIPRE on the same rod and axial location as the test measurement. The VIPRE manual advises against this approach. However, correspondence between GNF-A and TVEL identified that this method was used in the original development of the CRK CHF correlation, and correspondence between GNF-A and Dominion identified that the practice is common in the industry. The GNF-A report concluded that the approach was acceptable for the LTA program, but that it would be re-evaluated and better understood by GNF-A staff prior to reload applications. Some studies were included in the report to determine the sensitivity of the correlation to different values of the "predicted" CHF.

The correlation statistics were calculated by GNF-A using **[[** **]]**. GNF results were essentially the same as TVEL, resulting in a correlation limit of 1.14.

GNF-A performed studies with VIPRE to determine the sensitivity to various modeling approaches. One study was on the axial power shape, which found that there were significant differences in the axial power profiles that GNF-A and TVEL originally used. When GNF-A copied TVEL's method for determining the axial power shape, results were more similar. From these sensitivity studies, GNF-A also found that TVEL adjusted the grid span term to better reproduce the axial location of CHF when it was found to occur in the middle of the grid span (rather than where it typically is found, just below the spacer grid). GNF-A was able to confirm that this adjustment was made in the TVEL analyses and that it explained the differences between the TVEL and GNF-A results.

GNF-A also included in the report an approach for finding outliers, in which a point was determined to be an outlier if it was outside of a two-sided 95/95 tolerance limit (i.e., points whose z-score is greater than ± 1.960 based on a significance level of 0.05). The NRC staff does not find this approach to be acceptable and communicated this to the GNF-A staff. GNF-A staff then clarified that the approach was not employed. Instead, the approach used was the same as discussed in the TVEL TVSK-DR-EX-25, Rev. 1, report, which identified two outliers.

NRC staff audited GNF-A Report 004N9369, "Thermal Diffusion Coefficient Validation," which presented GNF-A's independent assessment of the thermal diffusion coefficient developed by TVEL. GNF-A concluded that TVEL's value of **[[ ]]** was acceptable.

NRC staff also audited GNF-A Report 004N9516, "VIPRE-01 Columbia/HTRF W-3L Correlation Benchmark," which presented GNF-A's benchmark of the W-3L correlation using VIPRE-01. GNF-A agreed with the correlation limit of 1.3, which is to be applied when W-3L is used in analysis of TVS-K fuel assemblies.

NRC staff also audited GNF-A report 0049485, "VIPRE-01 KI CRK Correlation Benchmark," reviewed the testing performed by TVEL at the KI thermal-hydraulic test facility using the updated spacer grids that are planned for use at Braidwood. In the report, GNF-A agreed with TVEL's conclusion that the KI data supports the conservatism of the correlation for the added spacer grid configuration.

Finally, the NRC staff audited GNF-A Report 004N9475, "CHF Engineering Evaluation Report," which summarized much of the information provided in the reports discussed above.

3.5 TVS-K LTA Thermal Limits Analysis Methodology

GNF-A prepared calculations for EGC to determine the power cutback needed to ensure the TVS-K assemblies were not limiting relative to the co-resident fuel at Braidwood. Calculations were documented in GNF-A Report 004N9129, "Braidwood Nuclear Power Station: VIPRE-01 Power Margin Calculation for TVS-K Fuel Assembly Power Cutback," and GNF-A Report 004N9080, "VIPRE-01 Power Margin Calculation for Braidwood LTA Power Cutback." The analysis approach was outlined by Westinghouse in EGC TODI NF183751, "Transmittal of Statepoints for TVS-K Power Margin Calculation," as discussed in a GNF-A task scoping document.

As discussed in GNF-A report 004N9129, Westinghouse provided EGC with thermal-hydraulic statepoints, which were then used as inputs to VIPRE-01 analyses of a core with mixed TVEL and Westinghouse fuel assemblies. Analyses were initially carried out with a 14-channel VIPRE model (typical of industry calculations), containing the TVS-K assembly in the middle, surrounded by Westinghouse VANTAGE+ OFA assemblies, with TVS-K assemblies in the rest of the core. This approach produced "unrealistically favorable" results for the TVS-K assemblies which caused GNF-A to switch from the 14-channel VIPRE model to a more detailed 25-channel VIPRE model.

GNF-A report 004N9080 discussed the VIPRE model in more detail, as well as comparing the initial VIPRE results obtained by GNF-A to those calculated by TVEL. The 25-channel model was similar to the 14-channel model but included explicit modeling of two more rows of rods and subchannels in the central fuel assembly. The report also applied the VIPRE model to evaluate crossflows between adjacent TVS-K and VANTAGE+ OFA assemblies, which is discussed in Section 3.3, above.

Results for the power margin calculation for TVS-K assemblies were provided in the report. Tom Rodack, contractor for EGC, obtained and shared with the audit team the equivalent results from Westinghouse for the co-resident VANTAGE+ OFA fuel which demonstrated that the TVS-K assemblies were non-limiting for the statepoints evaluated.

3.6 Seismic Analysis Methodology and TVS-K LTA Inputs

The approach used by the licensee to address the mixed-core effects on the structural performance of fuel loaded in the Braidwood reactor under external applied loads was to have Westinghouse perform the analyses based on specified inputs for the TVS-K LTAs. Each fuel vendor then compared the results to their respective acceptance criteria to verify that the structural performance of their fuel assembly design would be acceptable. As a result of this approach and the need to avoid mixing proprietary information from competing vendors, the NRC staff performed separate audits for the GNF-A/TVEL and the Westinghouse sides of the evaluation. A discussion of the NRC staff's review and key observations regarding the analysis of the TVEL TVS-K LTAs under external applied loads and the resulting impacts on the co-resident fuel is provided below.

During the first part of the audit, GNF-A described the process for addressing seismic and LOCA applied loads. In summary, Westinghouse provided a data request (e.g., frequency, stiffness, damping) to fully characterize the TVEL TVS-K LTA performance under external applied loads. TVEL then conducted testing based upon the specified parameters. The results of these tests were supplied to Westinghouse. GNF-A did not perform any finite element analysis modeling to simulate seismic- or LOCA-driven fuel accelerations and impacts.

Using their currently approved models and methods, Westinghouse calculated horizontal and vertical loads on the LTA parameters. These parameters were transmitted through EGC to GNF-A. During the audit, the NRC staff asked EGC about any assumptions either fuel vendor was making about the performance of the other vendors fuel design. EGC stated that an Owner Acceptance Review was performed on all GNF-A and Westinghouse transmittals and calculations. All assumptions regarding fuel characterization and performance were identified, assessed, and dispositioned.

TVEL completed testing to define acceptance criteria for the LTA components. In relation to the co-resident Westinghouse VANTAGE+ OFA fuel bundle, the TVS-K has a higher assembly stiffness, higher grid stiffness, and a significantly higher grid crush strength (over 200 percent). Grid crush testing was performed at room temperature and hot conditions, as well as accounting for simulated end-of-life conditions. When Westinghouse supplied maximum loads from their analyses, GNF-A compared them to the allowable limits based on TVEL testing and showed positive margin to all acceptance criteria.

During the second part of the audit, Westinghouse described their mixed-core fuel seismic analysis. Below are some observations regarding the overall mixed-core analysis methodology and how it accounts for the LTAs:

- Westinghouse-approved models and methods were employed.

- End-of-Life conditions are not specifically modeled, which is consistent with the current approved methods for Westinghouse.
  - No credit is taken for flowing water damping.

[[

]]

- The presence of the TVS-K LTAs has [[ ]] on the co-resident VANTAGE+ OFA fuel.
  - The OFA mid-grid margin [[ ]].
  - The OFA IFM grid margin [[ ]].
  - The impact on OFA component margins (e.g. guide tube, fuel rod) was insignificant.

- Westinghouse VANTAGE+ OFA fuel satisfies all design and regulatory criteria for mixed core assessment
  - Rod Cluster Control System insertion maintained (guide tubes not deformed)
  - No significant deformation (applied loads below Pcrit)
  - Coolable geometry maintained

The TVEL TVS-K and co-resident VANTAGE+ OFA fuel assembly grids do not perfectly line up axially, in location and height. The NRC staff reviewed Westinghouse calculations CN-NFPE-18-45 and CN-NFPE-18-65 to better understand the scaling factor used to adjust grid strength and stiffness to account for misalignment.

CN-NFPE-18-45 provides dimensions and elevations for the grids, from which the following observations are made:

- With the exception of the TVEL TVS-K grid at the 13 inch elevation, all grids match up. While there may be some overlap (above or below), impact loads are transferred through the grid straps and not grid-to-fuel rod.

- The accelerations and impact at the extreme top and bottom are expected to be minimal.

CN-NFPE-18-65 provides undergage grid impact test results, which was used to scale the impacts as follows:

- Impacts which contact less than the full height of a grid (i.e., undergaged impacts) affects both grid strength and stiffness

- Three different undergage impact conditions were tested:
  - [[ ]]
  - [[ ]]
  - [[ ]]

- The test data was used to scale the calculated results.

- Based on how the TVEL TVS-K and VANTAGE+ OFA grids line up, in all cases, the smaller grid will contact the adjacent grid fully across its entire surface area. Therefore, the adjustment is applied to the larger grid of any two grids impacting each other, because the applied load will not be distributed over the entire surface area of the larger grid.
  - The VANTAGE+ OFA mid grid is 2.25 inches in height, whereas the TVEL TVS-K mid grid is 1.772 inch in height (1.772/2.25 = 79 percent overlap ratio).
  - As a result, the mid grid impacts are adjusted as follows:
    - The VANTAGE+ OFA mid grid stiffness and strength is adjusted downward, because the TVEL load is not distributed across the entire grid surface area.
    - The TVEL TVS-K mid grid stiffness and strength is not adjusted because the overlap ratio is greater than 1.0 (applied load distributed to entire surface area).

During the audit, Westinghouse and NRC staff discussed the impact of TVEL TVS-K fuel relative to previous mixed-core evaluations. [[ ]] Westinghouse also noted that as an added conservatism, the maximum penalty factor is conservatively applied to the maximum loading location from the homogeneous core analysis to assess compliance with acceptance criteria.

3.7 TVS-K LTA Fuel Handling and Storage

The NRC staff discussed the work performed by the licensee to confirm that other aspects of the planned use of TVS-K LTAs at Braidwood that are subject to 10 CFR 50 regulations were adequately addressed, such as fuel handling and storage. A discussion of the NRC staff's review and key observations for the fuel storage criticality and FHA analyses is provided below.

3.7.1 Spent Fuel Pool and New Fuel Vault Criticality Analyses

The approach adopted by the licensee was to perform an evaluation of the TVS-K fuel lattice and the design basis fuel lattice from the licensee's analysis of record, to establish that the TVS-K fuel was bounded by the current criticality analyses of record. Nominal parameters were used for the fuel bundle geometry inputs, and the TVS-K fuel lattice was evaluated with an enrichment of 4.2 weight percent (w/o) U-235. The actual TVS-K fuel will be fabricated with an enrichment of 3.9 w/o U-235, so this represents additional margin inherent in the evaluation.

The TVS-K fuel was modeled as fresh, unpoisoned fuel in an infinite array to maximize its reactivity. Different sensitivity studies were run for water temperature, boron-10 areal density for the neutron absorbers in the spent fuel pool racks, fuel assembly pitch, flux trap gap, optimum moderation in the new fuel vaults, and other relevant characteristics to identify the limiting cases.

The results showed that the analysis of record bounded the TVS-K fuel by more than 4000 percent mille (pcm) for the spent fuel pool configuration and by more than 2000 pcm for the new fuel vault configuration. No fuel tolerances were evaluated, since the TVS-K fuel tolerances are consistent with the tolerances for other designs used within the United States and there is significant margin to account for any small variations.

3.7.2 Fuel Handling Accident

As discussed in Section 3.2.6, above, the radiological source term from the TVS-K fuel pellets is expected to be consistent with that of other commercial fuel designs currently used in the US. However, the higher density of the TVS-K fuel pellets may increase the amount of available radioisotope inventory for release during an FHA. In discussion with EGC and GNF-A, the NRC staff confirmed that GNF-A will evaluate the gap inventory for the TVS-K fuel at the end of its operating life, accounting for the higher mass of heavy metal in addition to the lower power due to the 5 percent cutback. EGC would then compare the result to their current licensing basis, which includes failure of a specified number of fuel rods that are all operated at the maximum allowed fuel rod power for their entire lifetime. Based on preliminary calculations, they believe that the TVS-K fuel will be bounded. If that does not turn out to be the case, then they would take additional measures or report to the NRC, as appropriate.

4.0 DOCUMENTS REVIEWED DURING AUDITS

A list of the primary documents that the NRC staff reviewed during the audits is provided below.

4.1 QA Documents

EGC Documents

- EGC Contract No. 653908, "Contract for the Braidwood Nuclear Station Lead Test Assembly Program, Lead Test Assemblies Supply and Associated Service between Joint Stock Company TVEL and Exelon Generation Company, LLC," July 15, 2016.

- "Amendment No. 4 to the Contract for the Braidwood Nuclear Station Lead Test Assembly Program, Lead Test Assemblies Supply and Associated Services between the Joint Stock Company TVEL and Exelon Generation Company, LLC dated 15 July 2016," and June 14, 2018.

- Letter from J.R. Wingfield (EGC) to O. Mastakov (TVEL), dated June 6, 2018, "Exelon Generation Audit of TVEL CMP Exelon Audit No. SR-2018-14."

- Letter from J.R. Wingfield (EGC) to I.V. Petrunin (TVEL), dated August 24, 2018, "Exelon Audit No. SR-2018-31, Exelon Generation Audit of TVEL Moscow."

- Letter from J.R. Wingfield (EGC) to I.V. Petrunin (TVEL), dated August 24, 2018, "Exelon Generation Audit of TVEL Moscow Exelon Audit No. SR-2018-31" (Audit package).

- Letter from J.R. Wingfield (EGC) to I. Leshukov (TVEL), dated October 10, 2018, "Exelon Generation Audit of TVEL-Novosibirsk Chemical Concentrates Plant (NCCP), JSC Exelon Audit No. SR-2018-34."

- Letter from J.R. Wingfield (EGC) to I. Leshukov (TVEL), dated February 4, 2019, "Exelon Generation Audit of TVEL-Machine Engineering Plant (MSZ), PJSC Exelon Audit No. SR-2018-35."

- Letter from J.R. Wingfield (EGC) to I.V. Petrunin (TVEL), dated February 6, 2019, "Results of Exelon Surveillance (#NF 184031) of Chepetsky Mechanical Plant (CMP)."

- Letter from J.R. Wingfield (EGC) to I.V. Petrunin (TVEL), dated February 12, 2019, "Deficiency/Corrective Action Closure & Status Exelon Audit Number: SR-2018-14 of Chepetsky Mechanical Plant (CMP)."

- Chepetsky Surveillance Package dated February 12, 2109.

- NO-AA-10, Revision 94, "Quality Assurance Topical Report."

<u>GNF-A Documents</u>

- GNF-A Opportunity 323661, "Purchased Service Agreement between TVEL and GNF-A for the Engineering Support of the US TVS-K LTA Program and Other Related Services," Revision 0, December 23, 2016 (a.k.a. EGC Contract No. 638740).

- GNF-A Opportunity 323574, "GNF-A and Exelon TVS-K LTA Program Agreement for the Braidwood Nuclear Station", Revision 0, March 20, 2017.

- PLM 004N5794, "TVS-K Braidwood LTA Project Plan", Revision 0, November 9, 2017.

- PLM DOC-0009-0707, "TVS-K PWR Fuel LTA Program and Reload Licensing Plan," Revision 2, September 26, 2018.

<u>TVEL Documents</u>

- RKK(USA)-1-2018, "Fuel Company Quality Manual for Operation at the American Market", Revision 2, September 20, 2018.

4.2 <u>Fuel Rod Design Documents</u>

<u>GNF-A Documents</u>

- 005N122-Revision 1 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section A.2 TVS-K PRIME03 Validation Summary."

- DBR-0043293-Revision 0 "PRIME03 Validation: E110opt Qualification Cases."

- DBR-0039777-Revision 0 "PRIME03 Halden Reactor Group E110 Qualification Test Suite."
- DBR-0043423-Revision 0 "PRIME Modeling and Measured Cedar File Creation for TVEL DR-09 Cases."
- DBR-0041896-Revision 0 "T-M TVS-K Fuel Rod Inputs."
- 005N0786-Revision 1 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section A.3 – PRIME Application Summary."
- DBR-0045096-Revision 1 "PRIME03 TVS-K LTA Evaluations."
- 005N0926-Revision 0 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section 4.1.1.2.2 – Fatigue."
- DBR-0041346-Revision 0 "PRIME Fatigue Analysis for E110opt Cladding."
- 005N0842-Revision 1 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section 4.1.1.5.2 – Rod Dimensional Changes."
- 005N0843-Revision 1 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section 4.1.1.6 – Rod Internal Gas Pressure."
- 005N2546-Revision 0 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section 4.1.2.2 – Cladding Collapse."
- DBR-0045171-Revision 1 "TER Section 4.1.2.2 – Creep Collapse Analysis for E110opt Design Basis."
- 005N0844-Revision 1 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section 4.1.2.4 – Overheating of Fuel Pellets."
- DBR-0043671-Revision 0 "Braidwood LTA Power-to-Melt Analysis."
- 005N0839-Revision 1 "Braidwood TVS-K LTA Technical Evaluation Report Input: Section 4.1.2.6 – Pellet/Cladding Interaction."
- DBR-0043623-Revision 0 "TVS-K Fuel Pellet Specification Evaluations Against ASTM Specification."
- DBR-0004164-Revision 5 "PRIME03P – Revision 15 ECP Software Change Plan (SCP) Revision 3."
- DBR-0032104-Revision 3 "PRIME03P – Rev. 15 Software Requirements Description Rev.6."
- DBR-00049987-Revision 2 "PRIME03P ECP Revision Test Report (RTR) Revision 0."

- 004N5921-Revision 1 "PRIME03 Software User's Manual Revision 15."
- DBR-0041593-Revision 0 "E110opt Material CEDAR file for use with PRIME03"

4.3 Thermal Hydraulics Documents

GNF-A Documents

- 004N4758, Revision 0, "TVEL CHF Quality Plan."
- 004N6094, Revision 2, "TVEL CHF and T/H Engineering Assessment Report for the Afrikantov OKB Mechanical Engineering (OKBM) testing facilities."
- 004N6189, Revision 1, "TVEL CHF Engineering Assessment Report for the Institute for Physics and Power Engineering (IPPE) testing facilities."
- 004N6190, Revision 1, "TVEL CHF Engineering Assessment Report for the Kurchatov Institute (KI) testing facilities."
- 004N7296, Revision 0, "Procured Acceptance Testing Documentation for VIPRE-01."
- 004N7693, Revision 0, "TVEL CRK Correlation Shared Library Update."
- 004N8457, Revision 0, "TVEL TVSK-DR-EX-25-REVISION 0 Evaluation Analysis – VIPRE-01 OKBM/IPPE CRK Correlation Benchmark."
- 004N9369, Revision 0, "TVEL TVSK-DR-EX-25-REVISION 0 Evaluation Analysis – Thermal Diffusion Coefficient Study."
- 004N9475, Revision 0, "TVEL CHF Engineering Evaluation Report."
- 004N9485, Revision 0, "VIPRE-01 KI Benchmark."
- 004N9516, Revision 0, "TVEL TVSK-DR-EX-25-REVISION 0 Evaluation Report - Columbia CHF Comparisons with VIPRE."
- 004N9558, Revision 0, "TVEL TVSK-DR-EX-25 Rev. 1 with Cover Sheet."
- DBR-0040182, Revision 0, "Validation of the F-test for Means Equality (Section 5.3.2.3) – Design Notes & ECP Files."
- 004N4759, Revision 0, "TVEL Thermal-Hydraulics Quality Plan."
- 004N9080, Revision 1, "TVEL – VIPRE-01 Power Margin Calculation for Braidwood LTA Power Cutback."
- 004N9129, Revision 0, "TVEL – Braidwood Nuclear Power Station; VIPRE-01 Power Margin Calculation for TVS-K Fuel Assembly Cutback."

- 004N9700, Revision 0, "TVEL TVS-K Thermal-Hydraulic Design Report, TVSK-DR-EX-21, Revision 0 and Cover Sheet," JSC TVEL Report TVSK-DR-EX-21, Thermal Hydraulics Design Report, Revision 0, March 2019.

- 005N0919, Revision 0, "Braidwood TVS-K LTA Technical Evaluation Report Input: Thermal-Hydraulics Sections 3.2, 4.2, and Appendix B," (file 005N0919_REVISION 0_TER_THSections_TVEL_LTA_Braidwood.pdf).

- 005N1684, Revision 0, "Engineering Evaluation of TVEL Thermal Hydraulic Design Report – TVSK-DR-EX- 21."

- 005N1954, Revision 0, Response to NRC Audit Open Items to Utilize TVEL TVS-K Lead Test Assemblies at Braidwood, Critical Heat Flux Supplement."

4.4 Seismic Analysis Documents

Westinghouse Documents

- Calculation Note CN-NFPE-18-45.

- Calculation Note CN-NFPE-18-65.

SUBJECT: BRAIDWOOD STATION, UNIT NOS. 1 AND 2 – REGULATORY AUDIT REPORT REGARDING LICENSE AMENDMENT REQUEST TO UTILIZE TVEL TVS-K LEAD TEST ASSEMBLIES; (EPID L-2019-LLA-0208) DATED AUGUST 7, 2019

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