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*Engineering, Environmental & Nuclear Services*

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**SUBJECT: Contract DE-EM0003976 NRC Licensed Facilities – Transmittal of Revision 1 of the Investigation of Measurable Radioactivity in Hydrogen Sampling Off-Gas Report (STI-NLF-RPT-038)**

Mr. Ferrara,

Spectra Tech, Inc. is attaching Revision 1 of the subject investigation report for the Department of Energy Idaho Operations Office submission to the Nuclear Regulatory Commission. The report provides an analysis of measurable radioactivity observed during hydrogen sampling of Fort St. Vrain fuel storage containers.

Please contact myself or Jay Newkirk, FSV Manager, at (970) 381-0589, if you have any questions or require clarification.

Sincerely,

David Bland
Program Manager

Attachment: STI-NLF-RPT-038, Investigation of Measurable Radioactivity in Hydrogen Sampling Off-Gas, Revision 1

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# Investigation of Measurable Radioactivity in Hydrogen Sampling Off-Gas

STI-NLF-RPT-038 Revision 1

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Prime Contract DE-EM0003976
NRC Licensed Facilities

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## 1.0 Introduction

The Fort St. Vrain (FSV) Independent Spent Fuel Storage Installation (ISFSI) Aging Management Program (AMP), as described in the Safety Analysis Report (SAR), Section 9.8, requires sampling of randomly selected Fuel Storage Containers (FSC) for hydrogen. Hydrogen sampling was performed on four FSCs in May 2017. During the sampling of three of the four FSCs, measurable amounts of radioactivity were detected in the sampling system off-gas. This report documents an investigation of the measurable radioactivity. This investigation is being tracked in the Management Issue Tracking System (MITS) as Issue Number MITS-17-044.

## 2.0 Discussion

The documented Safety Evaluation (SE) of the FSV Nuclear Generating Station described the fuel as consisting of fissile and fertile fuel particles. Each particle had a multiple-layer (TRISO) coating of porous pyrocarbon, silicon carbide, and highly impervious high density isotopic pyrolytic carbon. The TRISO coating was described as the primary fission product barrier and served to contain fission gasses released from the fuel kernel. TRISO coated fuel particles performed well in various tests with no significant coating failures and very low fission product release however, as in the case for water reactors, the Fort St. Vrain design provided for some coating or cladding failure. Based on experimental evidence it was expected that less than 1% of the fuel coatings would crack or fail in normal operations.

The Final Safety Analysis Report (FSAR) for the FSV Nuclear Generating Station further described the TRISO coating as giving the 200 to 500-micron diameter fuel particles excellent fission product retention characteristics. The FSAR further described the active fuel as coated fuel particles being closely packed and contained in an array of 0.5-inch diameter holes in the graphite fuel block. Fuel particles were described as having a four-layer TRISO coating; an inner layer of a porous pyrolytic carbon buffer, an intermediate layer of high density isotopic pyrolytic carbon, a thin layer of silicon carbide which was highly impervious to metallic fission products, and an outer layer which was a strong high density isotopic pyrocarbon.

The FSAR (Section 3.7, Fission Product Control) described the fuel element structural graphite as an effective secondary confinement barrier for metallic fission products due to its retention and sorption properties. The primary barrier to noble gas fission product release was the silicon carbide coating on each individual fuel particle. The high retention of coated particles was successfully demonstrated in operating reactors (UHTREX, Dragon, AVR), the FSV proof test element in Peach Bottom, and several in-pile loop test elements operated by GGA, ORNL, and the Dragon Project. Some of these experiments on the high retention characteristics of coated particles established that, for very retentive particles with multi-layered coatings such as TRISO particles, the release of fission products was primarily from failed particles and from particles having some amount of uranium contamination in the pyrolytic carbon coating. It is important to note that adequate cooling was, and continues to be, provided during storage of the spent fuel to prevent it from reaching temperatures at which air-graphite reaction could become significant and far below temperatures at which increased fission product release from the fuel elements would result.

The current revision of the FSV ISFSI SAR (Section 3.1.1.1) describes the fuel in further detail. The confinement barriers are also further discussed (Section 3.3.2.1).

The Safety Evaluation Report (SER) for transfer of the FSV ISFSI license to the Department of Energy stated that no gaseous waste was generated during normal operation of the FSV ISFSI and no gaseous releases were expected during accident conditions, but the facility had provision for installing HEPA filters to manage any airborne particulate radioactive material that might be released from an FSC during handling. It is important to note that normal operation at the time did not require breach of the FSC confinement barrier to sample for hydrogen. The SER further stated that although the TRISO coating on the fuel particles was considered the primary confinement barrier, the FSCs provided the confinement capability which the Nuclear Regulatory Commission (NRC) staff evaluated for compliance with 72.122(h) requirements. Acknowledged was the fact that the confinement design included a sealed port through the FSC lid which provided a means for leak testing and sampling the FSC main storage space.

Performed in preparation for license renewal for the FSV ISFSI, results of the aging management review of the fuel in storage indicated no aging effects that required an AMP of the fuel in storage. In the SER for the FSV ISFSI license renewal the NRC found that no AMP was required for the graphite fuel elements.

The FSV Nuclear Generating Station FSAR presented a summary of time-dependent radionuclide inventories in typical reload segments, a segment being one sixth of a core load, as a function of time in the reactor core. The inventories were limited to the predominant noble gas (Kr-85) and several particulates. Absent from the inventories were any halogens. After 27 years of radioactive decay, the list of predominant radionuclides would be expected to be limited to Kr-85 and a handful of particulates with Cs-137 and Sr-90 being the most predominant.

The Radiological Release Assessment documented in the FSV ISFSI SAR is an analysis of the releasable inventory of fission products from a single FSC containing six maximum powered fuel blocks of 600-day decay, defined as 0.001% of the solids (i.e. Cs-137 and Sr-90), 50% of the halogens (none predominant) and 100% of the noble gases (predominantly Kr-85) of a failed fuel particle fraction of 0.25%. Such a puff release at the charge face following an FSC lid removal would result in a whole-body dose largely resulting from the inhalation of Sr-90, and an external gamma dose mainly due to the presence of Kr-85. The external dose is found to be negligible compared to the inhalation dose. The point in presenting this analysis is that a release from an FSC with an assumed fraction of failed fuel would not be limited to Kr-85, but would be expected to have a detectable level of particulate activity (Sr-90 and Cs-137) associated with it, unless the fuel element structural graphite fully retains the metallic fission products.

The potential for fission product activity in the air space of an FSC was recognized, but not necessarily anticipated during the hydrogen sampling. The need to control the potential radiological risk was addressed in both the documented ALARA review (FSV-2016-01) and in the sampling procedure (STI-NLF-OPS-015, Fuel Storage Container Gas Sample/Vacuum Purge). Appropriate radiological controls were also incorporated into the Radiation Work Permit (RWP FSV-17-2/0, FSV ISFSI FSC Gas Sampling) and the sampling procedure.

During the hydrogen sampling the presence of radioactivity was indicated by an increased count rate on an adjacent Ludlum 177L benchtop count rate meter. The count rates for the various sample locations were in the hundreds of counts per minute (cpm) range with the detector probe on contact with or immediately adjacent to sampling tubing. No general area radiation level increase was indicated on an Eberline RO-20 gamma and/or beta radiation survey instrument. Count rates rapidly diminished when sampling stopped and fittings were disconnected. No removable radioactive contamination was detected at any point in the process. It is also important to note that;

- radioactivity was not removed by the in-line HEPA filter,
- the radioactivity dissipated quickly when sampling stopped and fittings were disconnected, and
- no residual contamination was detected.

With the absence of detectable levels of particulate activity (Sr-90 and Cs-137) in the in-line HEPA filter, another potential cause for the interim increase in detectable radioactivity is the presence of naturally occurring radon and thoron in the FSC internal air space and not fission product activity attributed to failed fuel despite the recognized potential for it being detected. The description of the Special Nuclear Material in the Materials License SNM-2504 for the FSV ISFSI indirectly implies 6.85% of the uranium in the fuel is a mix of U-238 and U-234. Both of these isotopes of uranium are in the Uranium Series decay chain. Radioactive decay of these two isotopes can ultimately contribute to the presence of radon (Rn-222) in a sealed FSC. The thorium in the fuel is in the Thorium Series decay chain. Radioactive decay of thorium (Th-232) can ultimately contribute to the presence of thoron (Rn-220) in a sealed FSC.

## 3.0 Conclusion

Historical safety basis documents for the FSV Nuclear Generating Station (SE and FSAR) and license basis documents for the FSV ISFSI (SER and SAR) discussed in Section 2.0 recognized the exceptional fission product primary confinement barrier provided by the TRISO coating of fuel kernels and the secondary confinement barrier by the FSC, but confinement capability was never thought to be absolute. There has always been a potential for TRISO coating failures and fission of uranium contamination in the pyrolytic carbon coating with some fission product release. Accidental release analyses documented in the FSV ISFSI SAR that involve an FSC breach assume a small fraction of failed fuel is contributing to a fission product release. The analyses may be conservative, but recognize the potential for fission product activity in the air space inside an FSC.

The analyses assume the unlikelihood of noble gas being detected during an FSC confinement barrier breach without the detection of the more predominant particulate fission products. The radiation and contamination survey results did not indicate the presence of residual particulate radioactivity. It is possible that a failed silicon carbide coating and/or a spontaneous fission of uranium contamination in the pyrolytic carbon coating could result in fission product release from a fuel particle, with the metallic fission products simultaneously being retained in the fuel element structural graphite, thus limiting the fission product release from the graphite to noble gases. Just as plausible is the release of radon and thoron gases attributed to the natural Uranium and Thorium series decay.

This investigation concludes that the interim but relatively low increase in detectable radioactivity associated with the hydrogen sampling off-gas may be attributed to either radon and thoron gases, a long-lived noble gas such as Kr-85 from uranium contamination fission in in the pyrolytic carbon coatings of some fuel kernels or the TRISO coating failure of some fuel kernels with full sorption of metallic fission products in the fuel element structural graphite, or a combination thereof. No further investigation is warranted, but observation of this occurrence should be recognized as a potential radiological hazard in any subsequent work planning for future planned seal leak testing or sampling of FSCs.

## 4.0 References

FSV ISFSI SAR, Section 9.8, Aging Management Program, Revision 14, May 11, 2017

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FSV ISFSI SAR, Chapter 3, Principle Design Criteria, Revision 8, July 29, 2010

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