OUT-OF-PILE VERIFICATION OF TRITON11™ BWR FUEL

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ABSTRACT

This paper presents the TRITON11 out-of-pile verification scope and methods, with examples from structural tests and analyses and thermal hydraulic loop testing as well as experiences from fuel service and manufacturing tests. This verification ensures that the TRITON11 fuel product fulfils all requirements for continued in-pile verification by irradiation of Lead Test Assemblies in a nuclear reactor.

1. Introduction

Westinghouse has been developing the **TRITON11™** fuel product, a new 11×11 BWR fuel design offering large fuel cycle cost savings and significant fuel reliability improvements for the global market. The TRITON11 fuel design was presented at the Top Fuel 2016 conference where detailed descriptions of its mechanical, nuclear and thermal hydraulic design features were given [1]. Since then, the final development stage of design verification by out-of-pile testing and analysis has been completed. Licensing of Lead Test Assemblies (LTAs) is near completion for two Nordic BWR plants, and the first LTAs are planned to be inserted in these plants in the spring and fall of 2019.

The TRITON11 product is the greatest leap in Westinghouse BWR fuel innovation since the introduction of the SVEA fuel concept in the early 1980’s. The fuel assembly mechanical design for Nordic ASEA-type reactors is shown in Fig 1. In total there are 109 fuel rods: 91 fuel rods of full length, 10 part-length rods (PLRs) of approximately 1/3 length and 8 PLRs of approximately 2/3 length. All fuel rods are resting freely on the bottom tie plate (BTP) and are laterally supported by 10 spacer grids of the same sleeve-type design as used in SVEA-96 **OPTIMA 3™** fuel. Three cylindrical water channels referred to as water rods (WRs) provide non-boiling water for improved moderation in the interior part of the fuel bundle. The three WRs are connected to both the handle and the BTP and thereby constitute the load bearing structure for handling of the fuel bundle. The spacer grids are secured from moving axially by welded capture heads on the WRs. A spherical balance plate is distributing the handling load between the three WRs. The TRITON11 design features an advanced fuel channel with varying thickness both laterally and axially to optimize mechanical and nuclear performances as well as an expansion of the inner dimension in the upper 1/3 region (coinciding with the end of the 2/3 PLRs) to reduce two-phase pressure drop. The design includes a full-size **TRIPLEWAVE+**™ debris filter of the same type as used in Optima 3 fuel, situated below the BTP and welded to the inlet transition piece.

Verification of the TRITON11 mechanical design has focused on the new design features and included structural and functional tests as well as detailed analysis by finite element (FE) methods; see Section 2. The thermal hydraulic testing scope was significantly increased compared to previous BWR fuel products to gain further insight and improve modeling accuracy; see Section 3. Manufacturing process development and testing have been completed prior to LTA production; see Section 4. New fuel inspection and repair tooling concepts have been developed and tested; see Section 5.

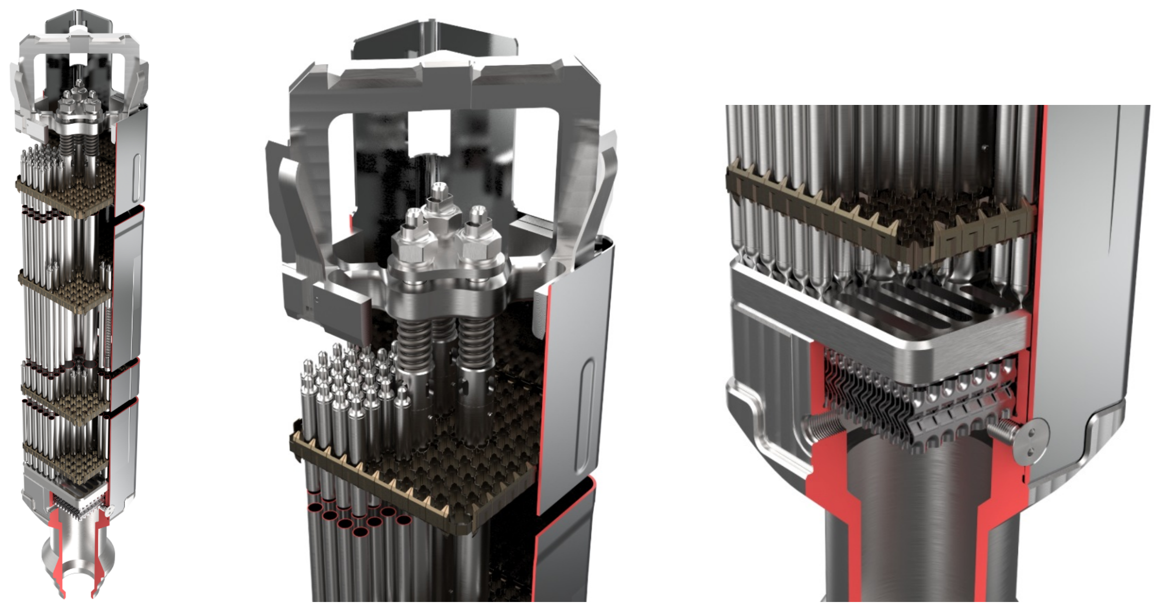

Fig 1. TRITON11 fuel design.

2. Mechanical Design Verification

The verification of the mechanical design has been conducted using test, analysis or in several cases both test and analysis. It has been considered of great importance to understand all different mechanical aspects of the new design features. Hence, it was decided early on to conduct both testing and FE analyses for those new features. Data from testing and analyses were used in different ways depending on the purpose; two independent evaluations confirming each other, one giving input to the other or a test validating a FE computational model. Loading conditions during the entire life of the fuel assembly have been considered, including manufacturing, transport, reactor operation, and handling. The tests and analyses include:

- Fuel assembly compatibility with other core internals, fuel assemblies, handling equipment, and storage positions. Fuel assembly components compatibility within the fuel assembly.
- Handling loads at normal and accident conditions, see examples in Sections 2.1 to 2.4.
- Endurance tests of complete fuel assembly at single and two-phase reactor conditions, considering anticipated geometrical and material changes during reactor operation.
- Operational loads under normal and accident conditions, such as pressure cycling of channel and WRs, and lateral load tests for spacer grids and channels.
- Transport loads from handling and transport test with complete fuel assembly (bundle within channel).

2.1 Handling Lift of Fuel Assembly

As an example of a confirmatory evaluation using both test and analysis, handling lift of the fuel assembly is considered. In the design for Nordic reactors, the fuel channel is fixed to the inlet transition piece including the TripleWave+ filter as one unit and the fuel bundle, including BTP and handle, is loaded into the channel from the top. The entire fuel assembly is lifted in two lifting lugs at the top end of the channel, but the handle of the fuel bundle is also designed to be secured to the grapple of the fuel preparation machine for redundancy. The fuel assembly lift is verified for a handling load of 7.5 kN at 60°C.

Two geometrical areas were identified as possibly being limiting when determining the lifting capacity; the channel lifting lugs and the connection between the channel and the inlet transition piece, including the four countersunk screws in the lower part of the assembly. For each of these areas, two standalone verifications were conducted; one by tensile test and the other by FE analysis.

The test was conducted in an electrical tension machine and was designed for determining collapse load in accordance to Appendix II of [2]. The FE model was created in ANSYS Workbench using 3D solid elements and 1/8 of the geometry. Plastic analyses according to the ASME Boiler and Pressure Vessel Code [2] were conducted using non-linear geometry and non-linear material modeling (Ramberg-Osgood). Worst case tolerance outcome and minimum specified material data at beginning of life were considered.

The two standalone verifications were in the end compared and evaluated. Fig 2 shows the results in the lower area. During the verification using FE analysis the screw connection was conservatively assumed to be frictionless. It was found that using a friction coefficient of 0.3 gave a very close match between the test and FE analysis.

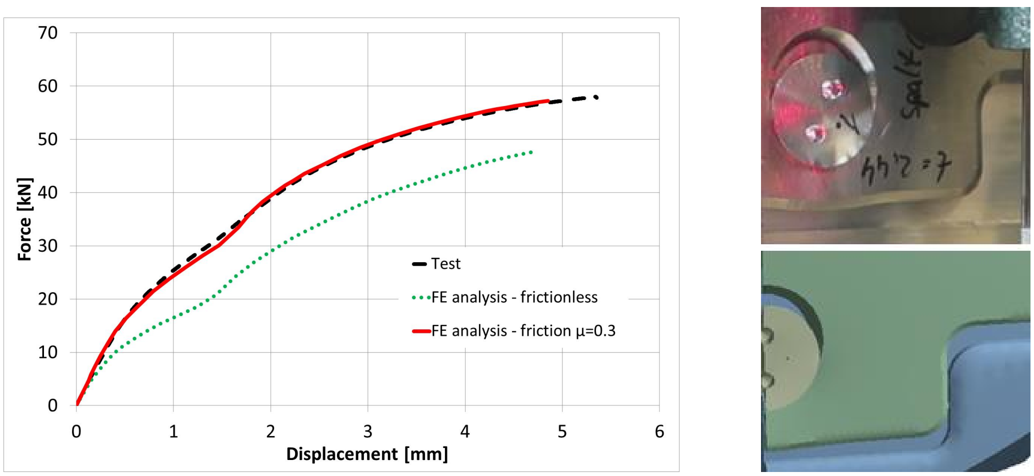

Fig 2. ***Left***: Comparison of test and analysis results assuming different friction coefficients for the channel screw joint. The test and analysis configurations included two opposite screws out of four. ***Right***: Photo after test with 45 kN and picture of the FE simulation.

2.2 Handling Lift of Fuel Bundle

A functional test of the fuel bundle handling lift has been conducted. The lifting load during handling of the fuel bundle is applied at the handle lifting bail and distributed to the WRs via the balance plate and the WR nuts. The functions of the top components and distribution of force between the WRs were of main interest. The test was conducted for a number of different conditions, including maximum expected differential WR growth and a postulated broken WR, i.e. lifting in two WRs only. Force sensors were incorporated in the WRs for measuring of the distribution of forces between them.

The test showed that the balance plate works as intended. The force distribution between the WRs was found to be satisfactory under all conditions. The test results were used for validation of the FE model.

2.3 Tensile Test of Water Rods

The WRs have been designed with the tube as the limiting component in tensile strength. Testing confirmed that the bottom and top end plugs, including their welds, are stronger than the tube. The rupture occurred in the tube section for all eight samples as shown in Fig 3. The evaluation of the force-strain curves was done in accordance with Appendix II of [2]. The results, shown in Fig 3, have been scaled by the ratio of the minimum specified yield strength at actual temperature to the test yield strength at the test temperature. The results show an allowable tension load for one WR of more than 10 kN at 60°C (normal handling conditions).

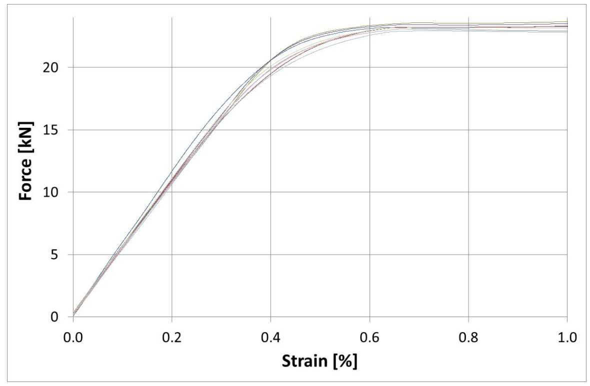

Fig 3. ***Left***: Photo of the test specimen after testing. ***Right***: Resulting force-strain curves.

2.4 Handling Accident

An accidental drop of a fuel assembly could be caused by breakage within the fuel assembly structure, the fuel grapple or the grapple cable/mast, or by improper grappling. Energy from the dropped assembly would be transmitted to the impacted fuel assembly/assemblies. A portion of the energy is absorbed by the dropped assembly and a portion is absorbed by the impacted assembly/assemblies. Energy absorption by the fuel rod cladding can cause cladding failure and the release of fission products.

This postulated handling accident has been investigated using nonlinear transient dynamic FE analysis using explicit time integration with the simulation software LS-DYNA. The main purpose of the analysis was to provide better insights into the postulated event and assisting the development of a robust design for TRITON11 fuel.

3. Thermal-Hydraulic Design Verification

Pressure drop and lift force tests on a shorter, but otherwise fully realistic, mock-up of the TRITON11 fuel assembly have been performed under single-phase flow conditions at the Westinghouse FRODE loop [3]. The main objectives of the tests were to determine:

- Pressure loss coefficients for fuel components that are not tested in the Westinghouse FRIGG loop (see below), i.e. inlet transition piece, BTP, TripleWave+ filter, and handle; see Section 3.1.
- Hydraulic lift force acting on the entire fuel bundle; see Section 3.2.
- WR flow dependence on WR inlet orifice diameter; see Section 3.3.

The following tests have been performed under realistic BWR conditions in the Westinghouse FRIGG loop [3] using a full heated section of a TRITON11 fuel assembly:

- Single- and two-phase pressure drops of individual fuel components; see Section 3.1.
- Dryout during steady-state and transients; see Section 3.4.
- Thermal-hydraulic instability limits; not presented in this paper.
- Local void and flow velocity by use of optical probes and Pitot tubes; reported elsewhere for scoping geometries, [4] and [5].

3.1 Pressure Drop

By installing pressure taps at different elevations in the FRODE and FRIGG test sections, differential pressure drops were measured across individual fuel components including bare rod friction in both the bottom and top sections of the TRITON11 fuel assembly (having different flow areas). A best-estimate pressure drop model at components level, including friction model, two-phase friction multiplier and spacer losses in various PLR regions, was developed by utilizing all detailed axial pressure drop measurements from FRODE and FRIGG. The conditions in FRODE covered temperatures from 20 to 80°C and flow rates up to 25 kg/s, corresponding to Reynolds numbers in the range of $(1.0 – 8.0)\times10^4$ (using the local flow area as reference area). The test conditions in FRIGG varied from low temperature single-phase to dryout conditions at reactor pressures, thus covering Reynolds numbers in the range of $(0.46 – 25)\times10^4$.

Example results from the validation of the TRITON11 best-estimate pressure drop model are shown in Fig 4. The prediction results are unbiased at the level of each fuel component and the total pressure drop is predicted with accuracy better than 2%. Additional verification of the pressure drop model was performed by CFD calculations, simulating the FRODE test section geometry and conditions including detailed modeling of the TripleWave+ filter, as shown in Fig 4. The CFD comparison of pressure drops showed excellent agreement.

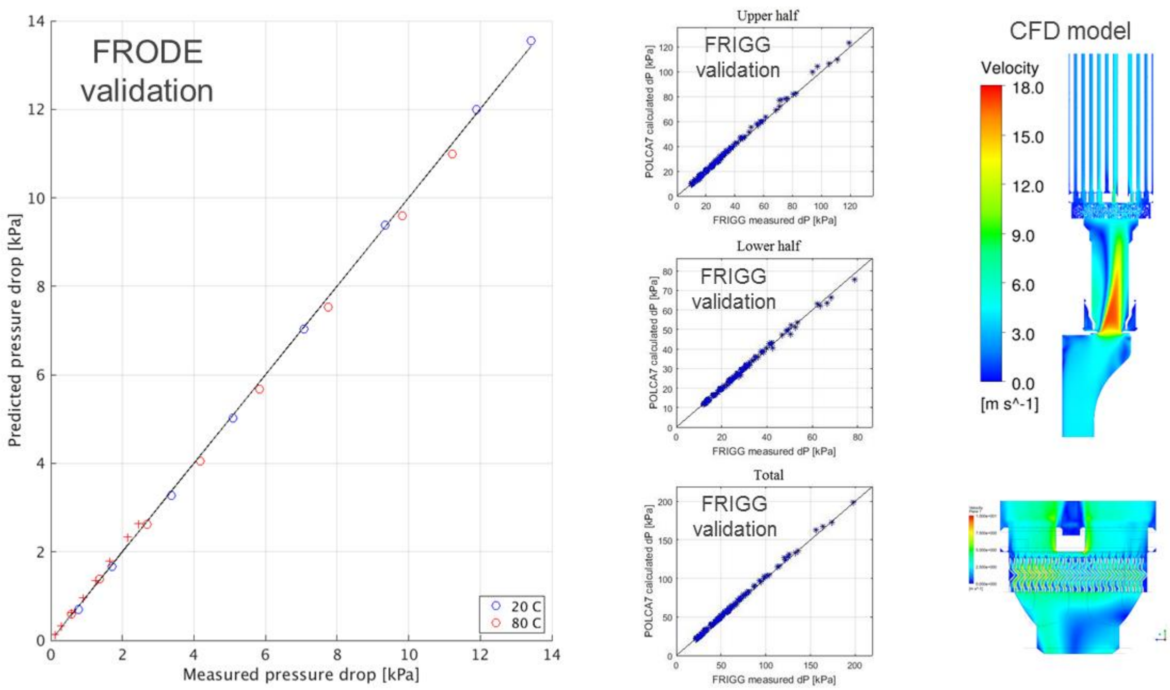

Fig 4. ***Left***: Example comparison of measured vs. predicted pressure drop in FRODE. ***Center***: FRIGG validation of TRITON11 pressure drop model implemented in the Westinghouse 3D core simulator, POLCA7. The example compares integrated pressure drops in the upper and lower halves of the heated part of the test bundle, as well as total pressure drop, at a wide range of conditions covering normal reactor operation and anticipated operational occurrences. ***Right***: Cutouts of CFD model of FRODE test section.

3.2 Bundle Lift Force

The hydraulic lift force acting on the fuel bundle only, disregarding the inlet components attached to the fuel channel, is a dimensioning parameter in T/H design of fuel for the Nordic ASEA-type reactors. The lift force results from geometrical characteristics and pressure forces acting on the bundle, i.e. friction shear stresses, local forces due to obstructions (BTP, spacers

and handle) and buoyancy. In addition, the lift force is sensitive to the distribution of flow in the active channel versus the WRs. Since the TripleWave+ filter is part of (welded to) the inlet transition piece, its pressure loss does not contribute to the bundle lift force. This adds important lift force margin for the TRITON11 design.
The lift force is directly derived from pressure drop calculations and one may question whether pressure drop validation only is sufficient. This assumption, not previously challenged, was assessed by performing direct measurements of bundle lift force in the FRODE loop. This was achieved by installing load cells at four contact points between the BTP and the inside corners of the inlet transition piece as shown in Fig 5. The weight of the test bundle was adjusted (increased) to obtain an appropriate lift margin by filling the cladding tubes with rodlets of stainless steel. The load cells were calibrated with measurements of the (known) bundle weight in air and water (zero flow).
Fig 5 shows a perfect agreement between measured and calculated lift force throughout the entire flow range. The only free parameter which was adjusted for the lift force calculation is the pressure loss coefficient of the BTP, since this could not be determined separately from the measurement of pressure drop over the test section inlet. Instead, the BTP loss coefficient was obtained by fitting against the measured lift force and was found to be in excellent agreement with the result from CFD calculations.

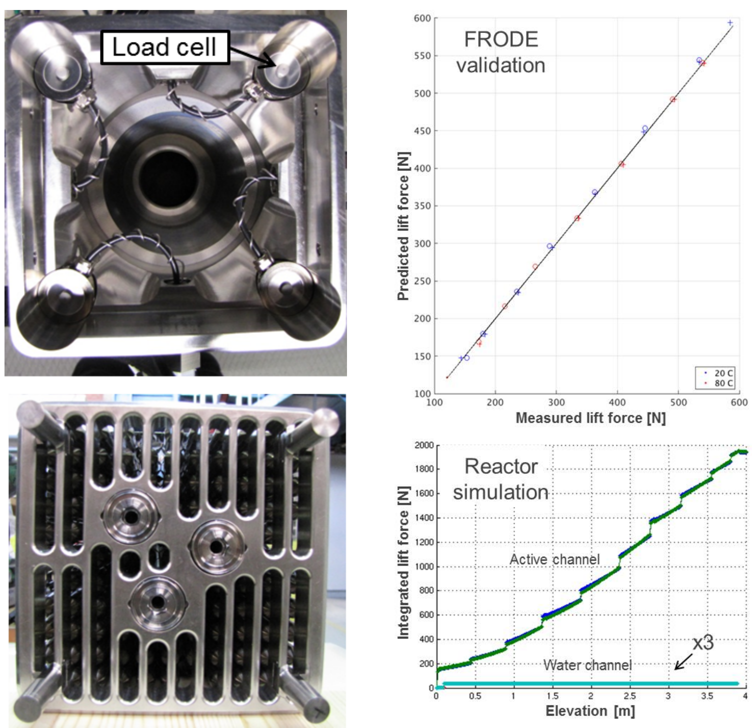

Fig 5. ***Upper left***: Inside view of inlet transition piece showing four load cells installed at the corners during the lift force tests. ***Lower left***: Special design of the BTP, standing on four legs in contact with the load cells, used during the lift force tests. ***Upper right***: Predicted vs. measured lift force in FRODE. ***Lower right***: Contributions of lift force integrated as function of axial elevation calculated for realistic reactor conditions. The incremental steps correspond to the spacer positions.

3.3 Water Rod Flow

The FRODE test bundle was equipped with WRs having different inlet orifice diameters of 4.0 mm, 5.7 mm and 7.0 mm. The flow velocity was measured inside each WR using Pitot tubes installed through the WR top end plug, see Fig 6. The cross-section average flow rate was derived from the local measurement by a turbulent velocity profile.

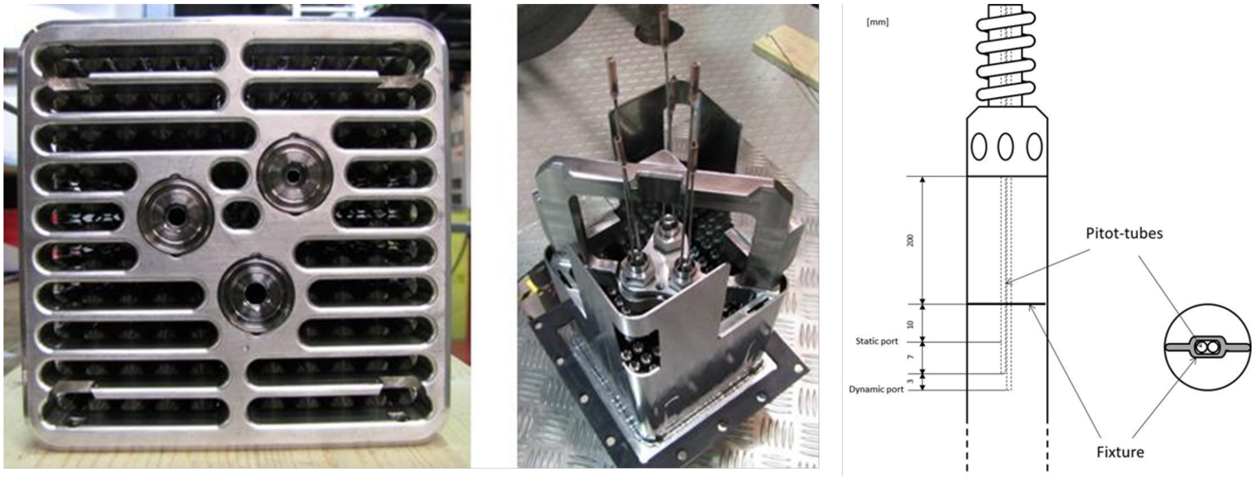

Fig 6. Experimental set-up for measurement of WR flow by use of Pitot tubes.

The WR inlet loss coefficient, K_{WR}, was determined for each test condition by iteration so that the pressure drop along the WR is equal to the corresponding pressure drop along the active channel (using the pressure drop model derived from experiments, Section 3.1). The loss coefficient was found to follow a simple relationship inferred from the Bernoulli Equation, $K_{WR} + 1 = \lambda \left(\frac{D_1}{D_2}\right)^4$, where D_1 is the WR tube inner diameter, D_2 is the inlet orifice diameter and λ is a multiplier fitted to the data.

The predicted versus measured WR flows for the three different inlet orifice diameters are plotted in Fig 7. The relative deviations are within 10%, except for a few measurement points at the smallest inlet orifice diameter which is affected by larger relative uncertainties in the Pitot measurements (reaching its lower range in dynamic pressure). This is perfectly acceptable for choosing a conservatively large inlet orifice diameter that will ensure insignificant boiling in the WRs under all relevant reactor operating conditions.

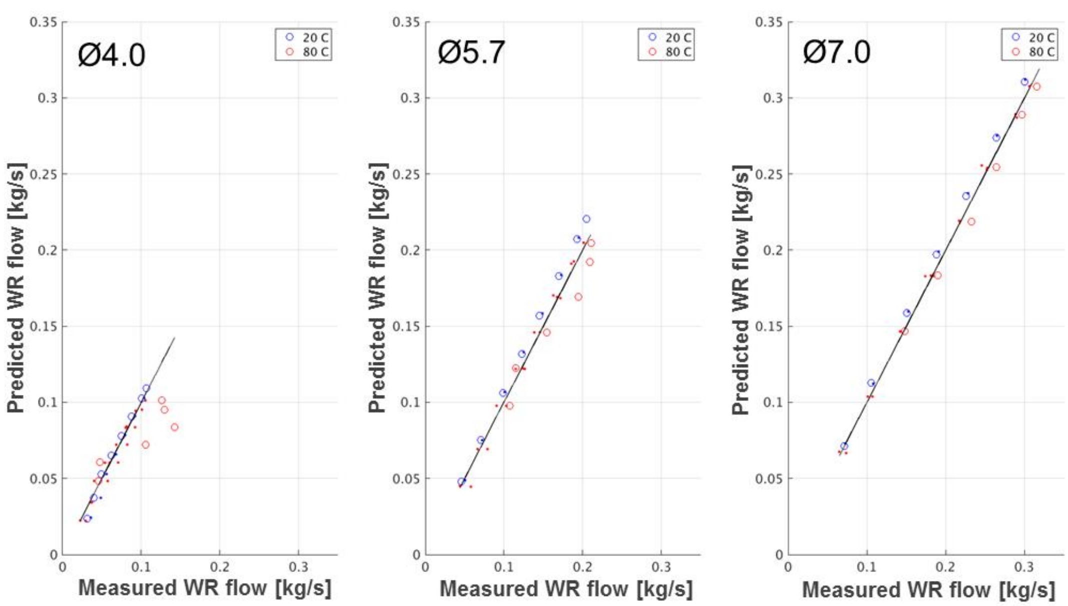

Fig 7: Predicted vs. measured WR flow for three different inlet orifice diameters.

3.4 Dryout

Dryout testing of the TRITON11 fuel design has been performed at the FRIGG loop under both steady-state and transient conditions. A full-size 11×11 test bundle was used with 109 fully-instrumented heater rods, a complete set of spacer grids with optimized vane features and a high-precision flow channel. Based on the obtained data, a new CPR correlation, referred to as "D6", has been developed for TRITON11 fuel. This section considers the validation of the D6 CPR correlation for application to transients.

During the transient tests, the heater rod thermocouple responses were recorded as function of time. The rod temperature typically varies first with heat flux (pre-dryout phase), then increases rapidly (dryout and post-dryout phase) before decreasing suddenly due to power reduction and associated rod rewetting (quenching phase). These characteristics were utilized to detect (1) if transient dryout occurred and (2) the onset and duration of dryout. In addition to the temperature traces, transient system response data were recorded in order to provide time-dependent boundary conditions for the transient system code simulations.

For testing in transient mode predefined excursions in power and/or flow are imposed on the test bundle over a limited period of time, ending with quenching of the temperature rise by a predefined power reduction. The TRITON11 transient tests can be categorized as power increase, flow reduction or combination transients for various transient power and flow histories. The tests were performed with different initial inlet flow rates and/or initial power levels in order to cover various levels of transient dryout severity, ranging from no dryout to deep dryout characterized by a high clad temperature increase. Fig 8 shows examples of transient events leading to dryout detection.

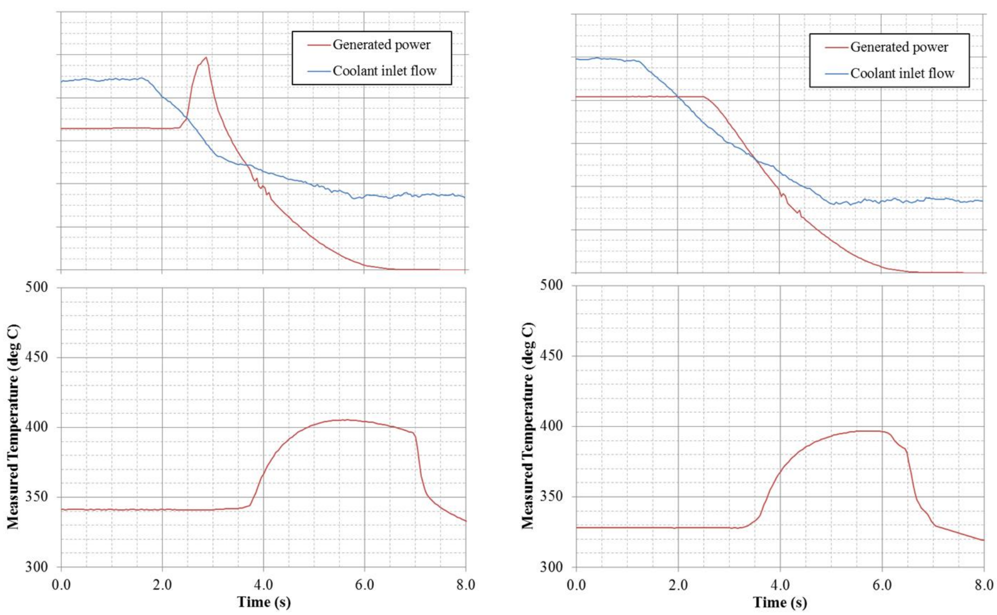

Fig 8. ***Left***: Transient dryout detection with fast power increase combined with flow reduction simulating the responses in a fast pressurization event. ***Right***: Transient dryout detection with combined power and flow reductions representative of a pump trip event.

Transient CPR validation involves confirming the capability of the transient code along with the CPR correlation to accurately predict the onset of dryout during selected simulated transient events representative of plant specific AOOs. The left panel of Fig 9 shows that with the D6 CPR correlation dryout is indeed predicted (i.e. CPR $\leq$ 1) each time that dryout was experimentally detected. The right panel of Fig 9 shows that by incorporating a wall heat transfer model in the transient code the maximum cladding temperature increase during each transient event is predicted with adequate accuracy.

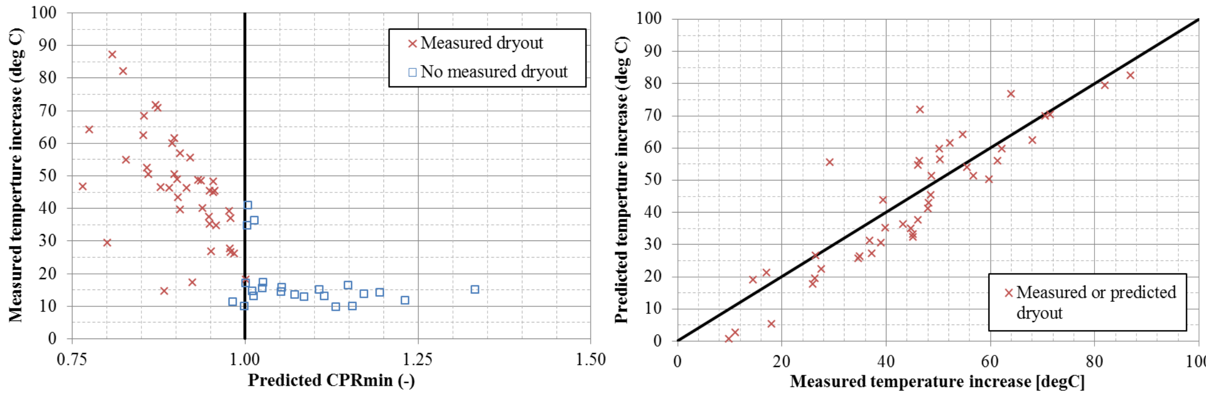

Fig 9. ***Left***: Validation of D6 CPR correlation for transient dryout prediction. ***Right***: Validation of D6 CPR correlation and transient code heat transfer model for predicting maximum post-dryout temperature increase.

4. Manufacturing Process Development and Testing

Efficient and reliable manufacturing has been a key objective during the development of the TRITON11 design. The ambition has been to maximize the usage of existing manufacturing experience as well as current processes to minimize risks and facilitate product introduction. All manufacturing processes for the first LTA deliveries have been tested, verified and qualified to meet the high quality standards and efficiency recognized in Westinghouse products. Most of the manufacturing processes qualified for TRITON11 fuel are the same or very similar to those used for previous Westinghouse BWR products, but some new design features have required development of new or adapted manufacturing processes.

Even though the TRITON11 fuel channel differs significantly from the SVEA channel, most of the manufacturing processes are the same. The longitudinal bending of the channel sheets into U-halves has required process development and innovative solutions, since this operation is done on pre-machined sheets with varying thickness, see Fig 10. Before the longitudinal bending operation of the flat sheets, a transverse pre-bending is performed to create the expansion of the inside flow area in the top of the channel, shown in Fig 10. The longitudinal bending process for the TRITON11 fuel channel has been made successful by optimizing the dimensions of the bending tools, introducing high precision shims in the transition areas of the sheet and electrically heating the sheet prior to bending.

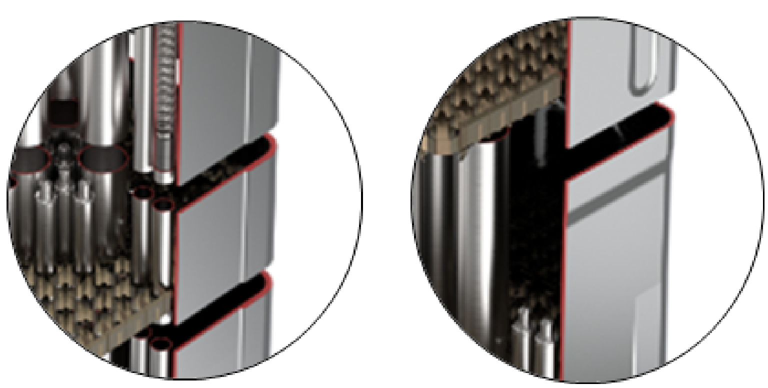

Fig 10. ***Left***: Lower 2/3 section of channel with thicker corner regions. ***Right***: Transition to upper 1/3 section of channel with uniformly thin sheet thickness and expansion of the inner dimension.

Bending of the U-halves is the first step in the shaping of the channel. In order to manufacture a channel with correct square dimensions, the shank height of the U-halves (planed), the longitudinal welding process (TIG) and the forming heat treatment (where a mandrel is used for sizing) are equally important.

The bundle assembly and spacer grid manufacturing processes are based on the processes used for the Optima 3 product, but scaled to fit the TRITON11 configuration where four sub-bundle grids have been replaced by one large grid. The larger grid adds complexity to the grid manufacturing and assembly operation. The development of the TRITON11 assembly process has been successfully completed without jeopardizing the favorable process features of the Optima 3 manufacturing, where rod lubrication is used to enable pushing all rods through the spacer grids simultaneously and protecting the fuel cladding from scratches and burrs. A final cleaning operation removes the rod lubrication and ensures cleanliness.

The load bearing WRs are new components in the TRITON11 product, compared to previous Westinghouse BWR fuel products, and the manufacturing processes developed for these rely on common processes for which Westinghouse has years of experience: Orbital TIG welding is used for the top and bottom end plugs, whereas the spacer capture heads, used to secure the axial locations of the spacer grids, are resistance welded. The heads are the same as in previous designs where they have been resistance welded to the spacer capture fuel rods for the same purpose.

The fuel rods are similar to previous Westinghouse BWR designs. The tube and pellet diameters and the end plug designs differ, but are well within the Westinghouse experience from PWR and VVER fuel rods. All TRITON11 fuel rods have the same top end plugs and the same bottom end plugs since tie fuel rods have been eliminated and the spacer capture function has been moved to the load bearing WRs. Hence, no fuel rods are exposed to handling loads. The end plugs are further optimized to facilitate manufacturing and fuel services. The top end plugs all have axial fill holes, in a similar configuration as Westinghouse PWR fuel rods, to facilitate manufacturing.

The TRITON11 fuel uses materials with operating experience and known benefits from previous Westinghouse BWR products [1]. Although all materials have been verified in previous products some applications or processes are specific to the TRITON11 design and these have all been successfully tested and qualified. The tests have included dimensions, corrosion performance as well as other material properties. The manufacturing process for the WR tube was optimized for corrosion resistance prior to the final qualification; a series of test tubes made with different chemical compositions and heat treatment parameters were subjected to a long term corrosion test.

5. Fuel Inspection and Repair Tooling Development and Testing

The basic concepts of fuel inspection and repair have been developed from the many years of experience with previous fuel products. When adapting the fuel service concepts to the TRITON11 design, special emphasis has been on enabling safe and efficient extraction and reinsertion of any fuel rod in the assembly. This is an important aspect both for fuel inspection and any potential fuel repair.

The bottom and top end plugs for TRITON11 fuel rods are shown in Fig 11. The bottom end plug has been designed so that it always guides the rod into the spacer grid, when a single rod is inserted, without risk for harmful interference with the spacer grid cells. This has been verified in analyses as well as by full scale tests, where the inserted rod has been provoked to simulate transverse forces from geometrical changes in the bundle after operation or potentially a slightly bent fuel rod during insertion.

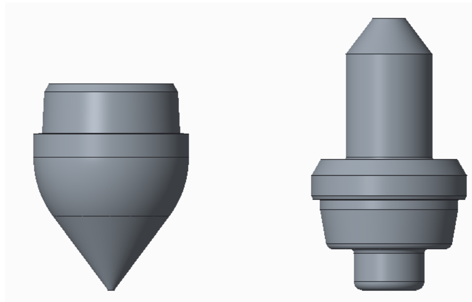

Fig 11. ***Left***: TRITON11 fuel rod bottom end plug. ***Right***: TRITON11 fuel rod top end plug.

The top end plug has a cylindrical pin for handling. The chuck of the rod handling tool grips the fuel rod by the end plug pin, creating a tight friction grip that allows for lifting and also rotating a rod while it is still inside the fuel assembly. The friction grip has been verified in Westinghouse PWR fuel. The ability to rotate the rod allows efficient inspection of all sides of the rod without (completely) removing it from the fuel assembly. The rod handling tool has been verified in analyses and tests.

In parallel to the TRITON11 specific inspection and repair processes, more general processes for inspection and detection are being developed and improved. Among those tools are: *F-SECT* (Frequency Scanning Eddy Current Technique) for Lift off, wall thickness and hydrogen examinations [6], *Rod Finder* using an ultrasonic technique to identify leaking rods inside a fuel bundle, vacuum sipping methods with increased sensitivity for identifying leaking fuel assemblies / fuel rods and *DECOSI*, a suctioning device that maintains a constant upward flow through the fuel assembly during fuel shuffling to prevent potential debris in the bottom of the assembly from being dropped onto other fuel assemblies (Westinghouse patent, EP2062266).

6. Summary and Conclusions

The TRITON11 fuel assembly design has been verified by out-out-pile testing and analysis to fulfill all requirements for insertion of LTAs in a nuclear reactor. This includes all mechanical, thermal-hydraulic and nuclear design features as well as the ability to safely transport the fuel in channeled condition and the ability to perform reliable fuel service, inspection and repair. The verification scope has been significantly extended by use of more advanced testing techniques and analysis methods to gain further insight into the behavior of the new design. All manufacturing processes for the first LTA deliveries to two Nordic utility customers in 2019 have been tested, verified and qualified to meet the high quality standards and efficiency recognized in Westinghouse products.

7. Nomenclature

The following abbreviations are used throughout this paper:

AOO	Anticipated Operational Occurrence
BTP	Bottom Tie Plate
CPR	Critical Power Ratio
FE	Finite Element
LTA	Lead Test Assembly
PLR	Part-Length Rod
WR	Water Rod

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