Demonstration of the Plant Fuel Reload Process Optimization for an Operating Pressurized Water Reactor (PWR)

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Abstract: The United States (U.S.) Department of Energy's (DOE's) Light Water Reactor Sustainability (LWRS) Program—under the Risk-Informed Systems Analysis (RISA) Pathway Plant Reload Optimization Project—aims to develop and demonstrate an automatized generic platform that can generate optimized fuel load configurations in the reactor core of a nuclear power plant. The project targets to optimize reactor core thermal limits through the implementation of state-of-the-art computational and modeling techniques. The optimization of core thermal limits allows a smaller fuel batch size to produce the same amount of electricity, which reduces new fuel costs and saves a significant amount of money on the back-end of the fuel cycle by reducing the volume of spent fuel that needs to be processed. The cost of a typical fuel reload for a light water reactor (LWR) is about $50M. This project is leading towards a cost reduction of at least 5%, which is attainable by consolidating methods and core design procedures and practices. The project includes the development of an artificial intelligence-based 'genetic algorithm' for the platform and demonstration of plant reload optimization with selective design basis accident scenarios for licensing support during fuel reloading. This platform integrates workflow that incorporates seamlessly all the steps required for the fuel reload analysis, which traditionally is a labor-intensive and time-consuming process. This paper summarizes the recent research outcomes, which progressed from the planning and methodology development phase to the early demonstration phase—including the development of a multi-objective optimization process using genetic algorithms; development and testing of an approach for acceleration of optimization using artificial intelligence that significantly reduces the computational burden; demonstration of the fuel reload optimization framework for a generic pressurized water reactor; and demonstration of selective scenarios for evaluation of the transition from deterministic to risk-informed approach for fuel analyses. On-going activities and plans are also summarized.

1. INTRODUCTION

Given that fuel costs represent approximately 20 percent of the total generation costs per the Nuclear Energy Institute (NEI) 2020 Nuclear Costs in Context report [1], one of the top priority requests from the industry is how to reduce those costs. The cost of a typical fuel reload for a light water reactor (LWR) is about $50M. Therefore, the United States (U.S.) Department of Energy (DOE) Light Water Reactor Sustainability (LWRS) Program Risk-Informed Systems Analysis (RISA) Pathway Plant Reload Optimization Project has tried to optimize reactor core thermal limits through the implementation of state-of-the-art computational and modeling techniques. The goal of this research is to develop an integrated, comprehensive platform offering all-in-one solution for reload evaluations with a special focus on fuel optimization, which allows a reduction in the volume of new fuel [2]. The platform will provide an optimized reactor core configuration based on key safety parameters that must be considered to meet regulatory requirements. The optimization of core thermal limits allows a smaller fuel batch size to produce the same amount of electricity, which reduces new fuel costs and saves a significant amount of money on the back-end of the fuel cycle by reducing the volume of spent fuel needing to be processed. An additional benefit of this platform is an integrated workflow that seamlessly incorporates all the steps required for the fuel reload analysis, which traditionally is a labor-intensive and time-consuming process. For this purpose, we have developed a multi-objective optimization process using genetic algorithms (GAs). Moreover, the fuel reload optimization framework for a generic pressurized water reactor (PWR) using ten limiting design basis accident (DBA) scenarios has been demonstrated effectively through an evaluation of the transition from deterministic to risk-informed approach for fuel reload optimization. This paper includes demonstration of the plant reload optimization approach. It shows how optimization of fuel configuration could be conducted. A large break loss of coolant accident (LBLOCA), one of the 10 DBA scenarios, is described as the case study.

2. DEMONSTRATION OF PLANT RELOAD OPTIMIZATION FRAMEWORK

This study develops an integrated platform that includes fuel configuration decisions using Risk Analysis and Virtual Environment (RAVEN) as the main integration and computational platform. At this time, GAs are applied to find the optimized fuel configuration, including variables corresponding to a system analysis, core design, and fuel performance. In detail, the core configuration is found through the GA, and the system simulation is controlled using a physical model. Application of the risk-informed approach will be investigated. This paper describes the approach to the platform and the GA example of finding the optimized fuel configuration considering variables on a small scale. In addition, deterministic analysis for risk-informed analysis is derived through DBA analysis. Specific details are described later.

By using a loss of coolant (LOCA) scenario in a generic PWR model with an equilibrium fuel reload cycle, the plant reload optimization framework has been demonstrated successfully. HELIOS-2 lattice code was used for the cross-section and PHISICS/RELAP5-3D was used for the core performance and accident analysis [3] [4] [5]. Figure 1 shows the core design strategy. The cross-sections for an initial configuration were computed. At least eight cycles were computed until the equilibrium cycle was reached. Then, the equilibrium cycle was analyzed for both options in terms of desired cycle length, maximum assembly burnup, and radial and axial power distributions. Subsequently, the design was adjusted to meet the design goals, while the equilibrium cycle was recomputed. Once the design goals were reached, the cross-sections were recomputed with HELIOS-2 for the new designs. Then, the equilibrium cycle was recomputed with the new cross-sections. Since the equilibrium cycle characteristics may have changed due to the updated cross-sections, further optimization and changes in the core design may have become necessary. These ‘inner’ (e.g., change core design and recomputed equilibrium cycle) and ‘outer’ (e.g., the recomputed cross-sections) loops, as shown in Figure 1, were repeated until convergence was reached, thus improving the development of the RAVEN infrastructure for the performance of neutronics and thermo-hydraulic analyses.

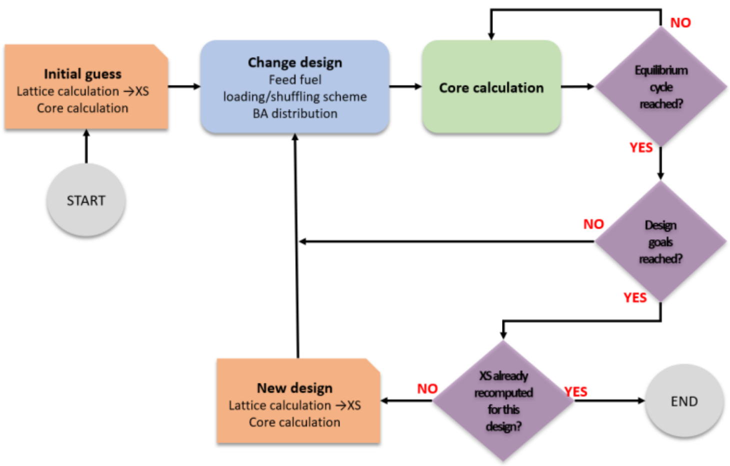

Figure 1. Flow chart for core design.

The case study considered to test the mechanics of the optimizer is a ¼ core initial loading problem. There are 56 locations needing to be loaded with five types of fuel assemblies; hence, the dimensionality of the problem is 56 (i.e., there are 56 genes in each chromosome). The objective function to be maximized is the cycle length, which for now is an unconstrained problem. However, a constraint is implied by the reactor physics code such that once the soluble boron reaches ≤ 5 ppm, the cycle ends. The inventory of the assemblies includes five different materials:

- Material – 1: Enrichment 2.2% in U-235, no burnable poisons
- Material – 2: Enrichment 2.5% in U-235, no burnable poisons
- Material – 3: Enrichment 2.5% in U-235, burnable poisons (8.0e-6 #/cm barn)
- Material – 4: Enrichment 3.5% in U-235, no burnable poisons
- Material – 5: Enrichment 3.5% in U-235, burnable poisons (8.0e-6 #/cm barn).

The GA found an optimal core configuration in the platform developed in this research. After that, the core configuration was analyzed using other codes to find the equilibrium cycle. DBA was analyzed for the risk-informed analysis. Accordingly, descriptions of the GA and those codes are described below.

2.1. Genetic Algorithm

An initial version of the optimization workflow was developed for the thermal limits and, consequentially, for the fuel pattern optimization. The development of such methods has focused on metaheuristic optimization algorithms. Specifically, GA has been selected to deal with this problem for the following reasons. The GA is preferred for non-differentiable, expensive to differentiate, or objective functions with no intrusive access (e.g., black box). Moreover, it does not get stuck in local minima; hence, it works with non-convex problems. With the GA, both constrained and unconstrained problems can be handled with discrete, continuous, or mixed design spaces, with binary, integer, real, or permutation variables due to encoding and decoding processes. The GA stems from biological evolution; it contains several evolutionary operations such as parent selection, cross-over, mutation, survivor selection, and repair/replacement. Detailed content about the GA is described in another Probabilistic Safety Assessment and Management (PSAM) conference paper: "Development of Genetic Algorithms for Plant Reload Optimization for an Operating Pressurized Water Reactor."

2.2. Core Physic Model

The goal of the demonstration is to simulate the equilibrium cycle and limiting scenarios by using an optimization framework for a simplified PWR core design. The outcomes of the limiting scenarios will then be used to inform further core optimization. The considered simplified core includes the following:

- Only one fresh fuel assembly enrichment for the whole core
- No axial lower enriched fuel zones on the assemblies
- The use of wet annular burnable absorber (WABA) rods.

For fuel design, axial fuel enrichment is constant. A variable number of WABA rods then are used to flatten the radial power distribution in the core. Furthermore, all twice-burned fuel has been placed at the periphery to have a modern low leakage loading scheme that allows for reaching high burnups, but still keeping it as simple as possible. A 17 × 17 lattice with 264 fuel rods and 25 non-fuel locations was the assembly used in these experiments. The non-fuel locations contain the guide tubes, WABA rods, and instrumental tubes.

2.3. Cross-Section Library Calculations

As mentioned, the lattice code 'HELIOS-2' was used to calculate the homogenized cross-sections. To identify the number of cross-section sets needed for this calculation, the 'proximity' approach was employed: even if the number of compositions is limited, an assembly is considered different from another when the neighboring ones are different (e.g., different composition, structural material, instrumentation tube locations, etc.). This approach led to identifying 29 different cross-section sets for the fuel region and one for the radial reflector, which was composed of the baffle, water between the baffle and the barrel, the barrel, and finally, the thermal shield. A detailed two-dimensional (2D) representation of ⅛ of the core was modeled with HELIOS-2.

The lattice calculations are generally started from pre-collapsed multi-group neutron energy structures. For the computation of the different cross-section sets, the lattice calculations were performed starting from a 44-energy group structure and then collapsed into an 8-group structure in the homogenization procedure. The 8-group structure has been used to find the equilibrium cycle. However, the calculation time is still relatively high using the 8-energy groups. For the complete (e.g., outermost) optimization loop that includes the feedback from the limiting scenario calculations, the neutron energy groups have been further collapsed into a 2-group structure.

The reactor calculation involves the simulation of the reactor during several operational cycles. To exchange feedback between the core design PHISICS tool and the thermal-hydraulic RELAP5-3D code, microscopic cross-section sets for all isotopes except the moderator are tabulated for several field parameters for each library. The cross-sections for the moderator regions have been tabulated as macroscopic cross-sections. This allows for the treatment of boron in solution in the moderator, which is a tabulation dimension, and boron in the burnable absorbers (BAs), which are tabulated microscopic cross-sections, separately. The following parameters and tabulation points have been computed for both core design options:

- Boron concentration in H_2O (ppm): 0.0, 1000, 1900
- Moderator density (kg/m^3): 640.8, 833.0, 945.2, 1000
- Fuel temperature (K): 573.2, 1073.2, 1273.2
- Burnup (GWd/tHM): 0.0, 0.152, 15, 25.

The tabulation dimensions lead to constructing a complete N-Dimensional (4-Dimensional in this case) grid characterized by 108 tabulation points in total. It is noted that the burnup points are for each cycle since the cross-section libraries are computed for fresh, once-burned, and twice-burned assemblies. Finally, the SuPer-Homogenization (SPH) method is applied for all generated cross-section libraries.

2.3. Coupled PHISICS/RELAP5-3D

To have a "fair comparison" of the behavior for different optimized cores, limiting scenarios need to be initiated from the equilibrium cycle conditions. Hence, the reactor evolution needs to be followed for several operational cycles until reaching the referenced equilibrium cycle. From a loading point of view, the equilibrium cycle can be considered the cycle from which the fuel reloading pattern is almost constant (i.e., the same composition and spatial loading of the fuel batches). The equilibrium cycle might be 'reached' after several reloads. This study assumes the equilibrium cycle is reached following the 8th reload.

The PHISICS code is coupled with the thermal-hydraulic RELAP5-3D code. For the search for the equilibrium cycle, the thermal-hydraulic model of the reactor has been set up considering the reactor core only (e.g., without a primary or secondary system). Only the core region without the primary system is considered for the base irradiation calculations (e.g., to search for the equilibrium cycle for a given core configuration) since the system does not influence the core during normal operation. For this reason, only the primary system is modeled, considering the upper and lower plenum of the core. The core is modeled using one core channel per fuel assembly to determine the initial conditions for the subsequent limiting scenario analysis as accurately as possible (i.e., 193 assemblies in total). The radial reflector is modeled as a bypass channel (i.e., 6% of the mass flow).

The PHISICS calculation of this coupled simulation is set up as the full core with 193 assemblies. The materials are assembly homogenized. One ring of assemblies containing a water/steel mixture has been placed around the active core to represent the reflector. The 3D-PHISICS calculation uses 18 axial layers.

2.4. Result

Figure 2 shows the layout and possible materials in 1/4 of the core. For this problem, a population size of 100 is used, 40 parents for the mating pool, the roulette wheel for the parent selection, the 'invLinear' fitness, one-point cross-over with 80% cross-over probability, swap mutation with 90% mutation probability, both locations of cross-over and mutation changes each iteration, a uniform discrete integer encoding with values between 1 to 5 randomly, since these are our options for the materials.

Figure 3 depicts the results of the optimization problem showing the history of the iterations, as well as the optimal value selected. It is vital to note that the problem is still non-realistic and that no thermal limits or constraints are considered yet.

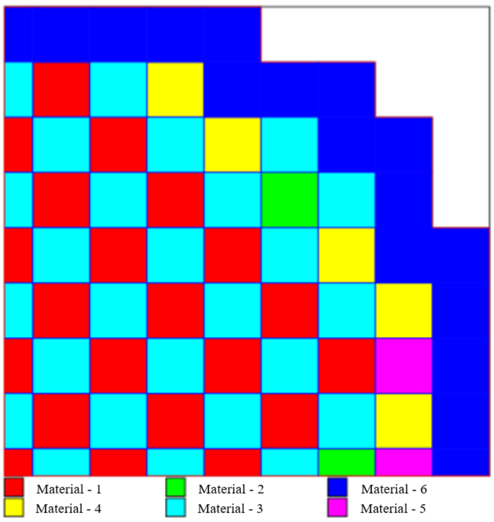

Figure 2. Layout of passible materials in 1/4 core.

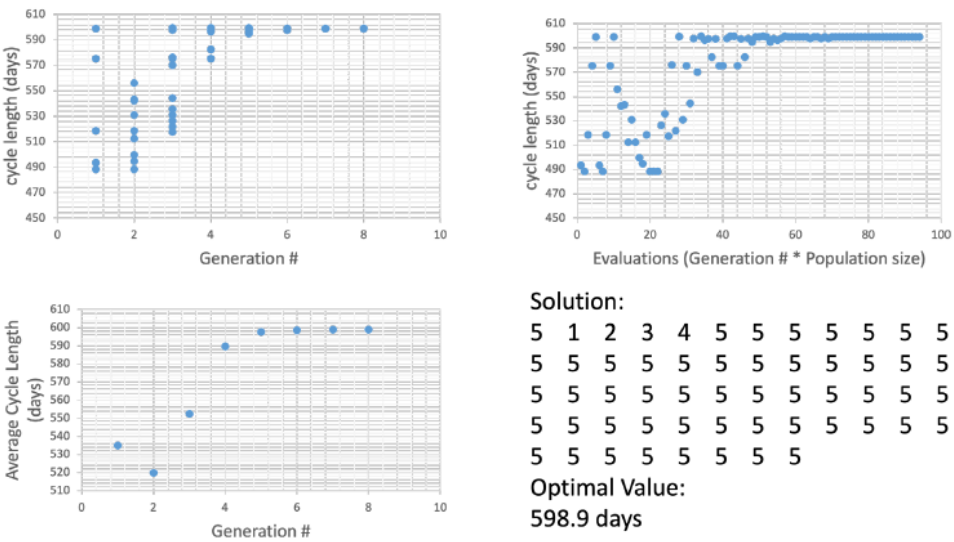

Figure 3. Demonstration result of fuel reload optimization framework.

3. ASSESSMENT OF DBA SCENARIOS FOR PLANT RELOAD OPTIMIZATION FRAMEWORK APPLICATION - LBLOCA

3.1. Overview

DBA scenarios listed in NUREG-0800 [6] need to be analyzed while applying the plant reload optimization concept. In a previous study [2], DBA scenarios in NUREG-0800 were reviewed, and ten DBA scenarios were selected. Using the Zion 4-loop Westinghouse PWR model, ten DBA scenarios were assessed in a one-dimensional (1D) deterministic method using RELAP5-3D and compared with a reference value to provide a basis for future risk-informed analyses [2]. In this paper, the Zion RELAP5-3D model has been updated to simulate other U.S. PWRs. Ten selective scenarios have been analyzed with multi-dimensional (MULTID) components, as indicated by their MULTID input cards, to simulate the complex phenomena in the reactor vessel area during DBA scenarios. Nodalization in the reactor core was changed to a finer mesh in the axial and radial directions. RAVEN code was used to control the RELAP5-3D simulation and uncertainty quantification for a LOCA scenario and test RAVEN's capabilities to support future risk-informed analyses. The MULTID component in RELAP5-3D allows the user to model more accurate multi-dimensional hydrodynamic features of reactor applications, primarily in the vessel (i.e., core, downcomer) and steam generator. The MULTID component defines a 1D, 2D, or three-dimensional (3D) array of volumes and the internal junctions connecting the volumes. The geometry can be either Cartesian or cylindrical. Mesh interval input data define an orthogonal, 3D grid in each of the three coordinate directions. It is also possible to add the full momentum flux terms and the associated input processing and checking. Out of ten limiting scenarios, only the LBLOCA scenario is described in this paper [2]. The following shows a detailed explanation of the analysis of LBLOCA and the acceptance and uncertainty analysis of the results.

3.2. Description

A LOCA results from a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analyses reported here, a major pipe break (i.e., large break [LB]) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a limiting fault—an ANS Condition IV event—in that it is not expected to occur during the plant's lifetime, but it is postulated as a conservative design basis.

The LBLOCA was the most limiting case and would not even successfully run without a huge reverse form losses applied at the broken nozzle in the transient input. The updated RELAP5-3D MULTID model shows significant improvements, but the case still failed when the form loss nominal value of the four-junction connection point, 1.783, was applied. The case failed at ~43 seconds, and it was due to the cladding temperature being too high. This is due to the core becoming uncovered around the failure time. The RELAP5-3D best-estimate plus uncertainty (BEPU) version allows the manipulation of the interfacial friction using a multiplier input. It is important to note this value varies globally, but by reducing the value, and thus reducing the interfacial drag coefficient, the downcomer liquid penetration should be easier to achieve. While the probability of liquid reaching the lower plenum and core region is increased, the ability to effectively cover the core is in question due to the multiplier being applied in the core components. Reducing the drag here may effectively hold the liquid in the lower cells and cause high cladding temperatures in the higher axial levels.

The model was rerun using the RELAP5-3D BEPU version with an interfacial friction value set to 0.5 and 0.1. Both cases failed due to high cladding temperature values. This results from an increase in reactivity. Because the fidelity of this model is not deemed sufficient to predict reflood and re-criticality, the reactor is tripped, and control rods are used at the beginning of the transient. Therefore, the reactivity is more negative, and the power is reduced immediately, whereas the original model saw a slower decrease, and in addition, no power spikes are experienced. The same two RELAP5-3D BEPU version cases were rerun, the 0.5 case fails at 102 seconds, and the 0.1 case runs successfully. Therefore, the 0.1 case will be used for the discussion in this section. All three updates (i.e., MULTID, decreased interfacial friction, attribute control rods) were required for the LBLOCA event. It is noted that through separate investigation in relevant integral effect tests, a multiplier in the order of 0.1 may be appropriate; however, this requires additional investigation.

3.3. Results

The results for the LBLOCA scenarios, as shown in Figure 4 through Figure 9, are compared to the results from the final safety analysis report (FSAR) and the acceptance criteria in Table 1. The acceptance criteria are based on an examination of the FSAR sections, in addition to the applicable sections of the standard review plan.

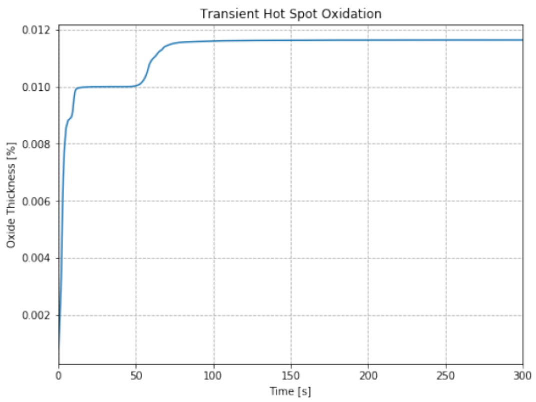

Figure 4. Transient hot spot oxidation.

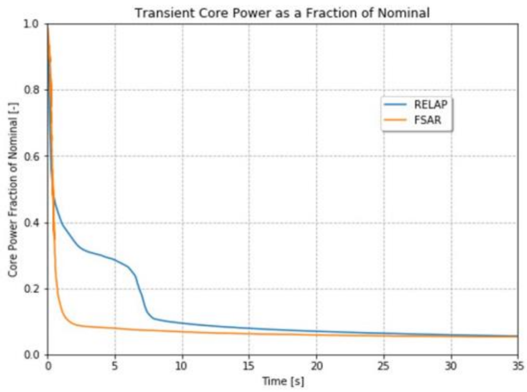

Figure 5. Transient core power as a fraction of nominal.

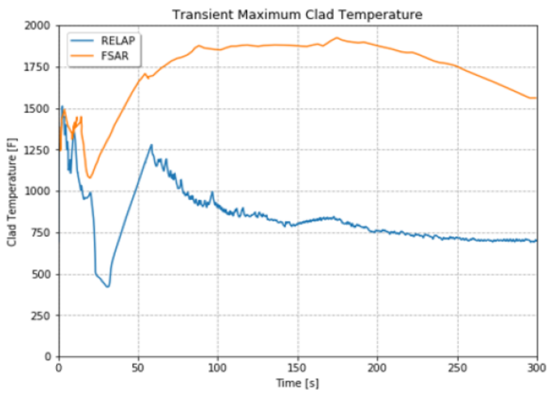

Figure 6. Transient maximum clad temperature.

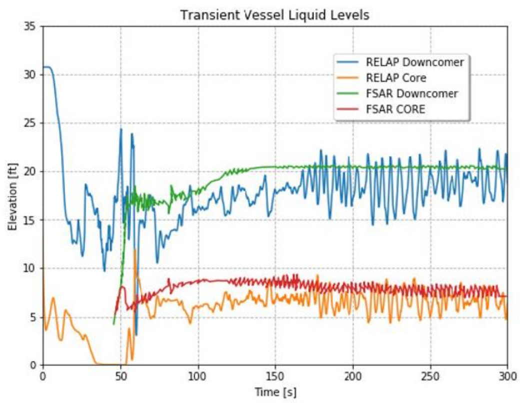

Figure 7. Transient vessel liquid levels.

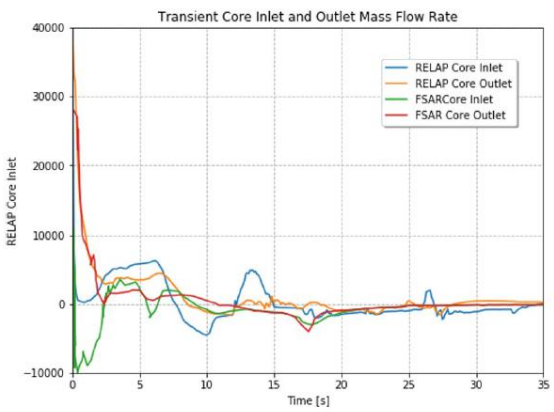

Figure 8. Transient core inlet and outlet mass flow rate.

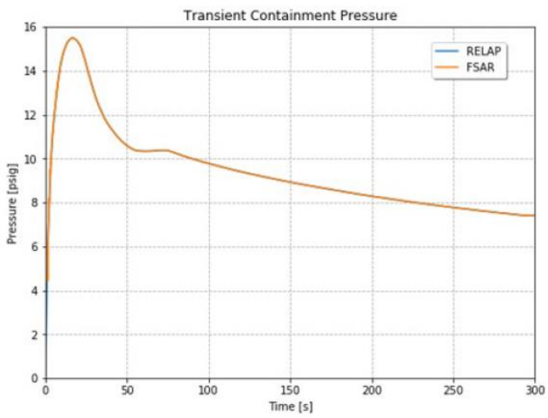

Figure 9. Transient containment pressure.

The RELAP5-3D simulations meet all acceptance criteria for the LBLOCA scenario. Some of the potential input differences between the FSAR and RELAP5-3D simulations are discussed in the following section.

Table 1. LBLOCA results.

Result	FSAR	RELAP5-3D	Acceptance Criteria
Peak Cladding Temperature [°F]	1936.3	1252	2200
Maximum Local Oxidation [%]	N/A	0.011	17

3.4. Uncertainty Analysis

The LBLOCA scenario is considered the most limiting out of events in Chapter 15 of the FSAR. Using the information presented in the LBLOCA section of this report, six parameters were chosen as essential values. Lower and upper bounds were created based on the given values. The parameters and their ranges are presented in

Table 2. All values are varied using a uniform distribution. Five of the six parameters are input into the steady-state portion of the event. The discharge coefficient is input in the transient since the break connections are applied using the trainsient input deck. A small sampling of 100 runs was executed. Of those 100 runs, 25 cases failed during the simulation. The failed cases are a combination of the cladding temperature reaching non-physical temperatures and non-convergence related to non-condensable. A failure rate of 25% is extremely high and must be reduced with future revisions. One possible issue resides in the RELAP5-3D BEPU version that was used due to the interfacial friction multiplier being global. If this multiplier can be isolated to specific components—specifically the MULTID component—the failure rate is expected to decrease. Figure 10 shows the mean value for the vessel mass at each time-step. The shaded region is in the 5th and 95th percentile of all the runs results. The initial rapid decrease in liquid inventory varied greatly between the runs. Around 50 seconds, most cases start to see the liquid inventory increase, decrease, and increase slowly to leveling off. It is important to note most failures were found to occur in the range of 45–65 seconds, which is where the rapid inventory changes are occurring. The total vessel mass in Figure 10 is likely lower than the actual data as well, since in general, the plots show similar trends.

Table 2. Monte Carlo parameter ranges for the LBLOCA uncertainty analysis.

Parameter	Default	Lower Bound	Upper Bound
Core Power [W]	3.66E+09	3.58E+09	3.73E+09
Discharge Coefficient	0.6	0.42	0.78
System Pressure [psi]	2250	2200	2300
Accumulator Temperature [°F]	120	80	120
Accumulator Volume [ft^3]	900	765	1035
Accumulator Pressure [psi]	611.3	611.3	811.3

4. CONCLUSION

This paper described the development of a multi-objective optimization process using GA. The platform developed in this study uses the GA methods for fuel optimization and considers system analysis, core design, and fuel performance. In detail, the plant reload optimization framework with a generic PWR is based on the performance of neutronics and thermo-hydraulic analyses. 10 DBAs in TR cases are assessed for the fuel reload optimization framework demonstration, and the LBLOCA scenario is described as representative. This platform will significantly simplify the necessary process of core reload evaluation performed for each fuel reload because it integrates all the required tasks into one seamless automated process. With the artificial intelligence technique combined with the GA, the platform will support flexible plant operations with increased or decreased reactor power levels following the fluctuating demand driven by the integration of renewables. For further work, we will fully connect system code such as RELAP5-3D, core design code such as SIMULATE, and fuel performance code such as TRANSURANUS. It will enable the consideration of more variables in the optimization with GA. The GA with multi-objective optimization that allows more advanced optimization will be enhanced as well. Moreover, we will perform enhancements of RELAP5-3D computer code to better support uncertainty quantification for design basis analyses (DBAs) because the proper uncertainty quantification is essential to reduce conservatism and, as a result, increase safety margins. The platform could extend the framework capabilities for other core configurations (e.g., PWR with accident tolerant fuels and a 24-month refueling cycle in a generic BWR).

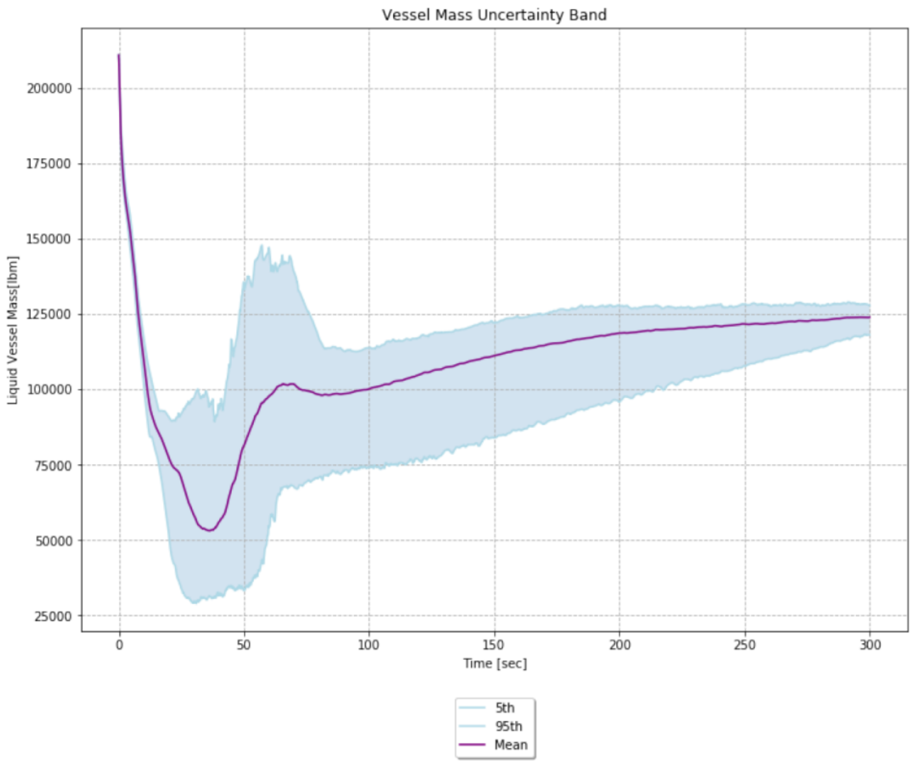

Figure 10. LBLOCA Monte Carlo sampling (Vessel liquid mass uncertainty results).

Acknowledgments

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