CIAE-ANL CRADA for SFR SAFETY ANALYSIS

Final CRADA Report

Nuclear Science and Engineering

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Non Proprietary Final CRADA Report

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Nature of Work

Describe the research (summary of Scope of Work and principal objectives of the CRADA):

The purpose of this project is joint verification and validation of system analysis code capabilities for Sodium-cooled Fast Reactor (SFR) design and licensing. As the researchers and code developers at the Argonne National Laboratory (ANL) and China Institute of Atomic Energy (CIAE) independently build their own SFR system analysis capabilities, this collaboration facilitated exchange of information on, and evaluation of alternative approaches for, modeling the key phenomena (such as decay heat removal and natural circulation) and the safety-related components (such as pumps, heat exchangers, and outlet plenum) common in SFR systems. The analysis at ANL and CIAE also applied their respective computational capabilities and methods (in SAS4A/SASSYS-1 computer code at ANL and FR-Sdaso computer code at CIAE) on jointly identified benchmark problems for verification, and publicly available integral tests for validation, of their results. ANL team also shared the expertise on the established software quality assurance practices and the techniques for component reliability estimates to support probabilistic risk assessments.

DOE mission area:

Energy and Environmental Science and Technology

Conclusions drawn from this CRADA; include any major accomplishments:

The scope of this project consisted of seven technical tasks:

Software Quality Assurance and V&V Methodology

The objectives of this were to provide guidance on software quality assurance (SQA) best practices and verification and validation (V&V) methodologies through training of the CIAE personnel. Argonne's experience in this area was used to provide guidance to CIAE as they proceed with development and qualification of their FR-Sdaso software as they continue to engage with their regulatory agency. This task consisted of two subtasks, as well as an SQA/V&V workshop.

Argonne provided a one-day workshop to approximately 30 CIAE personnel during the first Joint Project Meeting held in Beijing. Four topics were covered during the all-day workshop: SQA Best Practices, V&V Best Practices, SAS4A/SASSYS-1 V&V Test Suite, and Historical Validation of SAS4A/SASSYS-1. The first two sessions on SQA and V&V best practices targeted the development of SQA and qualification frameworks that fit within the modern regulatory environment. Key takeaways regarding SQA best practices included an improved understanding of the following concepts: software requirements, software design, configuration management (CM), testing and acceptance, problem reporting and corrective actions, and documentation and traceability. As a key topic in SQA, additional focus was given to CM practices, including configuration identification, configuration control, configuration status accounting, and configuration evaluation and reviews. These elements are necessary to enable and support important processes during V&V. Key takeaways regarding V&V best practices included an in-depth distinction between the process of verification versus validation, discussions on grading of V&V activities as per the intended end-use (e.g. integrity level), development of a V&V Plan of suitable rigor, and associated processes such as testing and acceptance in accordance with the V&V Plan. The final two workshop sessions focused on Argonne's historical and present-day experience with software qualification as it relates to SAS4A/SASSYS-1. An overview of the legacy testing and documentation activities was provided, where the intention was to highlight the differences between historical and current SQA/V&V requirements. The current software qualification strategy for the code was also discussed (as shown in Figure 1), where it was expected that CIAE could adopt a similar strategy.

Figure 1: Test Problem Development Strategy for SAS4A/SASSYS-1

With the aim of supporting CIAE's objective to develop a pedigreed safety analysis tool, and as a follow up to the SQA/V&V Workshop, remaining work focused on identification of U.S. SQA and validation requirements and best practices and providing an overview of SAS4A/SASSYS-1 V&V experience. Recommendations on minimum V&V activities that should be pursued in the near-term were also provided with the objective of improving the acceptability of CIAE's SFR analysis tool, FR-Sdaso, for use in licensing safety analyses. While nine U.S.-developed V&V standards, requirements, and guidance documents were surveyed, particular focus was given to the NQA-1 Standard, DOE Order 414.1D and Guide 414.1-4, and Regulatory Guide 1.203 due to their broad acceptability for V&V and software qualification and direct relevance to HAD102, the guiding regulatory document with which CIAE must comply. Also, historical and recent experience with SAS4A/SASSYS-1 SQA and V&V was formally outlined in detail as an efficient SQA framework that satisfies a broad range of requirements with minimal detail, all of which are characteristics that align with the resources available to CIAE. The range of existing V&V tests supporting SAS4A/SASSYS-1 was also described to enable CIAE to rapidly build a set of acceptance tests to support FR-Sdaso qualification. Lastly, a series of near-term activities were recommended for CIAE to help bootstrap and rapidly build an SQA/V&V basis. These recommendations include: Develop and implement an SQA Program similar to the SAS4A/SASSYS-1 Program; and, prepare the documentation needed to support the Evaluation Model Development and Assessment Process (EMDAP) outlined in RG 1.203. The latter recommendation directly aligns with CIAE's requirements as HAD102 is largely derived from EMDAP.

This task also focused on documenting the current status of FR-Sdaso V&V and identifying a feasible path forward for compliance with the existing regulatory requirements outlined in HAD102. CIAE's V&V methodology (shown in Figure 2) largely aligns with the EMDAP strategy of RG 1.203, although some differences (such as postponement of the PIRT evaluation) were implemented to better accommodate CIAE's workflow and licensing plan. An existing PIRT derived from CFR-600 operating characteristics has been utilized for FR-Sdaso. The PIRT scopes approximately forty components/areas and was considered comprehensive enough to support full V&V of FR-

Sdaso. Existing V&V benchmark cases and their results were also discussed with existing gaps being noted for future resolution that is beyond the scope of this CRADA.

Figure 2. FR-Sdaso V&V Methodology

Beyond assessment of the current status of FR-Sdaso V&V, the recommendations provided were revised and refined as per feedback from CIAE to identify a path forward that will achieve full compliance with their regulatory requirements. At the center of this recommendation was development of an SQA Program that contained the following core work activities: Project Management and Quality Planning; Configuration Management; Software Requirements Identification and Management; Software Design and Implementation; Problem Reporting and Corrective Action; Verification and Validation; and Training. A simplistic structure for SQA Program documentation was recommended that entails the following: SQA Plan, CM Plan, Coding Standards, and procedures for basic activities such as problem reporting and corrective actions, software testing, and software development.

The last activity entailed the identification of the documentation necessary to support EMDAP and therefore HAD102. RG 1.203 was utilized as a framework for development of related documentation as HAD102 was in draft format and under revision during the CRADA project. Therefore, a series of documents with bases from RG 1.203 were identified and outlined at a high level as meeting the requirements of HAD102: Requirements Identification Report; EM Methodology Report; Code Manual; Models and Correlations Quality Evaluation Report; Scaling Reports; Assessment Reports; and Uncertainty Reports.

EBR-II SHRT-45R Benchmark Analysis

SHRT-45R was the most severe unprotected loss of flow test performed during the EBR-II Shutdown Heat Removal Test (SHRT) program. During this test, the plant protection system (PPS) was deliberately prevented from initiating a scram to demonstrate the effectiveness of EBR-II's passive reactivity feedbacks. This test provided critical data for validation of SFR computer codes for analysis and design. An EBR-II benchmark specification was originally

created for an International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP), for which ANL was the lead technical organization. Although CIAE was also part of the IAEA-CRP, CIAE and ANL performed joint simulation and analysis of the EBR-II SHRT-45R test under this project as well focusing more on emerging modeling capabilities at each institute.

The first part the effort focused on reactivity feedback calculations. ANL documented the neutronics calculation methods and the resulting reactivity feedback coefficients in a report and CIAE subsequently began calculating their own neutronics calculations, which were refined throughout the duration of the collaboration. The second part of the effort focused on simulating the SHRT-45R test. ANL documented its initial results in another report, discussing the SAS4A/SASSYS-1 model developed for simulations and analysis of the subject EBR-II test, the initial simulation results from the blind phase of the IAEA CRP, and subsequent analysis that was performed to improve and refine the results. The modeling updates focused on understanding the differences between model predictions and measured test data and reducing the discrepancies between them as appropriate. CIAE team shared their initial SHRT-45R predictions using measured power data as a boundary condition with ANL for comparison.

The parametric and sensitivity analyses that were performed after the conclusion of the blind phase, focusing on what changes to key input parameters might be necessary to improve test predictions. These analyses focused principally on improving agreement of both simulations with recorded temperatures and flows, and also investigating the power predictions for SHRT-45R. A direct comparison of ANL and CIAE's SHRT-45R predictions was performed to illustrate similarities between the two sets of predictions and to understand the source of differences, prompting several updates to the models used to achieve consistent assumptions. Additional model refinements were also considered to activate the transient power and reactivity feedback models in the codes.

For most aspects of the transient, CIAE's FR-Sdaso and Argonne's SAS4A/SASSYS-1 models predicted very similar results. The biggest discrepancy was observed for the power prediction in the second half of the test, with both models slightly overpredicting the measured power. The absolute difference for both codes is very small, while the relative difference is approximately 50%. This power discrepancy in the second half of the test affects the Z-Pipe inlet temperature prediction, which is also the parameter with the largest prediction discrepancy between the two models. This difference is speculated to be due to different heat transfer rates predicted by the two models. Further analysis focused on improving the power prediction would likely reduce the difference between the measured and predicted temperatures, but additional analysis would be required to resolve any differences related to heat transfer rates in the top of the core channels and the upper plenum.

Additional differences were observed for several of the reactivity feedback effects, although the sum of these differences resulted in very similar predictions for net reactivity due the individual feedback differences canceling out. Further investigation of the individual feedbacks, particularly the reactivity feedback coefficients that were input in the models, could illuminate the cause of these differences. It is believed that the overpredicted power levels during the second half of the test were due to the simple radial core expansion models in both codes being unable to capture subassembly bowing due to EBR-II's unique core restraint system design.

Flow rates predicted by the two models are found to be in good alignment. One reason was that CIAE updated their pump model to account for the hydraulic resistance of a locked pump rotor, which may occur around 90 seconds when the pumps stop actively turning. The Argonne model does not predict pump locking for SHRT-45R, but previous analysis of the protected loss of flow test SHRT-17 predicted pump locking. It is uncertain if the pump speeds and flow rates were low enough in SHRT-45R to cause pump locking. Even though the CIAE's model does predict pump locking and Argonne's does not, the differences between the measured and predicted flow rates and between FR-Sdaso and SAS4A/SASSYS-1 flow predictions are well within the expected range of uncertainty.

Phénix Natural Circulation Test Benchmark Analysis

This task centered on analysis of the Phénix Natural Circulation (NC) Test conducted by CEA in 2009, where Argonne and CIAE utilized their respective SFR safety codes to create models of the system, compare results, and improve their models as necessary. This activity proceeded as follows: both Argonne and CIAE individually developed preliminary system models for the NC Test based on specifications provided in the open-literature IAEA report.[1] Comparisons of steady-state and transient results obtained with SAS4A/SASSYS-1 and FR-Sdaso were completed in the early stages of the project. The system models (including input and/or software implementation, as necessary) were then updated to improve the results. These final models were compared with each other and with the experimental data, where any remaining discrepancies between codes or experimental data were fully explored. The summary provided here focuses only on the final findings of Task 4. Only limited details regarding system models and quantitative comparisons are provided here.

Phénix is a pool-type MOX-fueled prototype SFR that operated from 1974 to 2009. As part of the end-of-life test campaign for Phénix, the 2009 Natural Circulation Test was performed to simulate a loss-of-heat sink event from reduced power and flow. Prior to the Natural Circulation Test, the reactor power and flow were reduced. After reaching equilibrium at this state, the steam generators were dried out followed by manual scram and primary pump trip. At a later stage, the steam generator containments are opened to promote cooling by air natural convection for long-term decay heat removal. Thus, from a reduced power and flow condition, the Phénix Natural Circulation Test mimics a protected loss of heat sink event. The cooling system of Phénix includes three primary pumps and three secondary loops, two of which were in service during the test. Each secondary loop includes two intermediate heat exchangers (IHX); therefore, only four of six IHXs were in service during the test. The two out-of-service IHXs, which were plugged, have some instrumentation inserted nearby. Approximately 1500 subassemblies comprise the core and are partitioned into six core power and flow rate zones: inner core, outer core, fertile (blanket) subassemblies, steel reflectors, control rods, and in-containment fuel storage zones. During normal operation, approximately 10% of the flow provided by the pumps is diverted from the core through the diagrid in order to cool the vessel wall.

Both Argonne and CIAE utilized similar system nodalization approaches in their respective system models, as shown in Figure 3. Prior to transient simulations, steady-state models were created in both codes to confirm the use of correct initial conditions for the test. Measured data provided in the IAEA specification[1] was utilized as boundary conditions for the transient, where these parameters included core power, steam generator outlet temperature, secondary circuit flow rate, and primary pump impeller speed. Due to modeling constraints, several boundary conditions were imposed using surrogate parameters. Reactor power boundary condition is entered in both codes as a normalized power. The IHX secondary side inlet temperature boundary condition is entered in SAS4A/SASSYS-1 as a steam generator outlet temperature, whereas FR-Sdaso could directly apply this boundary condition to the IHX. The secondary side flow rate boundary condition is entered in SAS4A/SASSYS-1 as a normalized secondary pump motor torque, whereas in FR-Sdaso it is entered as a normalized pump speed. The primary pump speed boundary condition is entered in both codes as a normalized primary pump speed.

General comparisons are provided here regarding the capabilities of FR-Sdaso and SAS4A/SASSYS-1 in modeling of the Natural Circulation Test. In this case, it is helpful to divide the test into three phases: Phase 1, a protected loss of heat sink from 0 to ~470 s; Phase 2, a natural circulation development transient without a significant heat sink from ~470 s to ~1,000 s; and Phase 3, a natural circulation transient without and then with efficient heat rejection via secondary cooling from ~1,000 s to ~10,000 s. Predicted temperatures of interest include the core inlet, core outlet, IHX primary inlet, IHX primary outlet, IHX secondary outlet. Predicted flow rates of interest include core, IHX primary, and main vessel cooling (MVC).

Figure 3. Phenix Reactor Primary System Nodalizations of FR-Sdaso (left) and SAS4A/SASSYS-1 (right)

In general, both codes agree well with each other and the measured data in Phases 1 and 3 for all liquid temperature metrics of interest. The only exception to this is prediction of core outlet temperature, which both FR-Sdaso and SAS4A/SASSYS-1 significantly overpredict during peaking in Phase 1 in the short term subsequent to reactor scram. It is not clear that the measured data provided for core outlet temperature can be used for direct comparison with the system code results, as the averaging scheme for core outlet thermocouples and flow fields near the measurement locations are not well described. During Phase 2, there is increased disagreement between code results and with experimental data. Both FR-Sdaso and SAS4A/SASSYS-1 agree well with the prediction of core outlet temperature and IHX primary outlet temperature, but they disagree in prediction of the core inlet temperature and IHX primary inlet temperature. It is likely that alternative treatments of the hot and cold pools in the codes contribute to this difference, with the most significant issue being the use of the thermal stratification model in SAS4A/SASSYS-1.

Relative to experimental data for Phase 2, SAS4A/SASSYS-1 aligns with the measured data for core inlet temperature, whereas FR-Sdaso predicts a different, but similar, trend. Both codes predict a similar, increased cooling trend rate for core outlet temperature in which the rate differs slightly from measured data. Both codes overpredict the primary IHX inlet coolant temperature, which as discussed above, is likely due a combination of modeling choices and inability of the codes to capture the 3D effects that may be affecting the thermocouple measurement. Regarding IHX primary outlet temperature in Phase 2, both codes predict a gradual temperature decrease, whereas the measured data is capturing sharper 3D effects. Regarding total core, IHX primary, and main vessel cooling (MVC) flow rate predictions, both codes show similar trends for all phases, and agree well in Phase 1 of the transient. Upon establishment of natural circulation flow, core flow rates show a non-negligible difference between the codes, likely a result of differences in model parameters and differences in core inlet temperature during Phase 2, however trends are generally similar in Phases 2 and 3. The long-term trends of the vessel cooling flow rate agree between codes, but SAS4A/SASSYS-1 predicts a slightly higher flow rate. A general comparison of code-to-code and code-to-experiment data indicate that both FR-Sdaso and SAS4A/SASSYS-1 provide reasonable results that were within expected ranges.

Decay Heat Modeling

Under this task, ANL and CIAE jointly developed detailed benchmarks studies to evaluate the accuracy and completeness of the point kinetics and decay heat models implemented in their respective codes. Argonne develops and maintains the SAS4A/SASSYS-1 safety analysis code for fast reactor safety simulations, and CIAE is developing the FR-Sdaso code for similar purposes. In fast reactor safety simulations, point kinetics models are central to predicting the power evolution during postulated accidents as a result of changing reactor conditions. Verification of the point kinetics implementation is vital to ensure confidence in the simulations. Likewise, decay heat power is produced during normal operations, accident conditions, and following shutdown. Verification of the decay heat model implementation is vital to ensure that residual power is properly quantified so that heat rejection performance and resulting safety margins can be evaluated.

As part of the collaboration, both institutions exchanged high-level documentation of the theory and capabilities of the point kinetics and decay heat models available in each code. This documentation was used to develop common benchmark studies for code verification. In most cases, benchmark studies were derived from published standards or open literature references. One case was developed by CIAE to be representative of conditions expected in the CFR-600 design concept. Benchmarks included the following:

- Decay heat benchmark based on the ANS/ANSI decay heat power standard
- Decay heat benchmark with multiple pre-transient irradiation intervals (derived from ANS/ANSI standard)
- Decay heat benchmark based on CIAE-provided data
- Point kinetics benchmark based on reference cases published by Ganapol for step, ramp, and sinusoidal reactivity insertion
- Coupled point kinetics and decay heat benchmark based on the preceding cases

Overall, the outcome of these verification benchmarks showed that both codes had implemented highly-accurate computational methods that matched reference results to at least one part per million and often better by several orders of magnitude. In addition to the value of code verification, the benchmark cases did expose some minor deficiencies in each code:

- A very minor error was found in the decay heat data used in SAS4A/SASSYS-1. Although the bug introduced negligible error in the results, it has been corrected.
- FR-Sdaso was found to have a small normalization error when predicting decay heat power based on the ANS/ANSI standards. This has been corrected.
- FR-Sdaso did not originally have the capability to predict decay heat with multiple pre-transient irradiation intervals. CIAE added this capability to their code to facilitate the full spectrum of the benchmark cases.

It is worth noting that the error in SAS4A/SASSYS-1 was found when computing the results for the cases with multiple pre-transient irradiation intervals, and the error would not have been detected without the highly accurate results available from FR-Sdaso, which was updated for that specific benchmark. Based on the outcomes from this task, verification of the point kinetics and decay heat models in SAS4A/SASSYS-1 were verified with a high degree of confidence.

Multi-dimensional Flow and Heat Transfer in Sodium Pool

Under this task, ANL and CIAE formulated solutions for modeling the 3-D effects of flow and heat transfer in a sodium pool. Based on the initial discussion between ANL and CIAE, it was decided to focus on examining different modeling approaches to capture the thermal stratification effects in SFR transient. Mixing, thermal-stratification, and mass transport phenomena in large pools or enclosures play major roles for the safety of reactor systems. It is

important to accurately predict pool temperature and density distributions for both design optimizations and safety analyses of pool-type SFRs. Suggestions for consideration of 3-D effects in system analysis models were proposed in this work in order to build confidence in CIAE capability to consider 3-D effects for analysis of thermal stratification.

The first deliverable was to develop or define benchmark problems that both CIAE and Argonne will analyze and to perform code-to-code comparisons of simulation results, and code-to-measurement comparisons where available. Based on the initial discussions, it was decided to develop a simple problem that ANL and CIAE will work together and compare simulation results using their respective tools and modeling approaches. This exercise focused on examining the modeling approaches in the tools for thermal stratification and mixing in the pool and only performing code-to-code comparisons. Based on the Advanced Burner Test Reactor (ABTR) primary loop design, the Argonne team developed a simple test problem (see Figure 4), which represents a simplified SFR primary loop during a protected loss-of-flow transient.

Figure 4. The Simplified SFR Primary System Model: Loop Model (left) and 3-D rendering (right)

Both CIAE and ANL performed simulations of the simple benchmark problem and compared simulation results using their respective tools and modeling approaches. The Argonne team performed both system-level simulations using the SAM code and CFD simulations using STAR-CCM+. The simulation results of both steady state and PLOF transient were summarized and submitted to the CIAE team, including: 1) the stand-alone heat exchanger simulations with different boundary conditions; 2) the system-level simulations of the loop problem; 3) the CFD simulations of the stand-alone tank case; and 4) the CFD simulations of the loop problem. The tank velocity and temperature distributions in the CFD simulation at one timestep during the transient are shown in Figure 5.

A detailed literature review was also performed on various modeling approaches in system codes for thermal stratification and mixing modeling. Three different modeling approaches were included. The first one is commonly used in existing system codes which models a large enclosure with multiple zero-dimensional volumes. The second approach is based on one-dimensional fluid models with additional models accounting for jet/plume behavior and the thermal mixing from macroscopic flow circulations and microscopic turbulent mixing. The third approach is based on three-dimensional coarse-grid CFD approach, in which the three-dimensional fluid conservation equations are solved with embedded closure models for the viscous stress and turbulence effects.

Figure 2. Velocity (left) and Temperature (right) Distributions in the Tank at $t = 50$ s

The technical basis, pros and cons of the three modeling approaches were summarized in a report along with some example simulations. Suggestions for consideration of 3-D effects in system analysis codes were provided for modeling thermal mixing and stratification in SFR pools. Among others, a key suggestion was, for a system code that is under development and does not have any models for thermal stratification modeling in pools, first the 0-D models should be developed because of its simplicity and ease of implementation, and the multi-dimensional modeling capabilities should be next focus based on the evolving available resources and developmental priorities.

Primary Pump Coastdown Modeling

The centrifugal pump models in SAS4A/SASSYS-1 and FR-Sdaso are jointly evaluated. The SAS4A/SASSYS-1 pump model is predicated upon the dimensionless homologous pump theory of Wylie and Streeter and later modified by Marchal, Flesch and Suter. The homologous pump head and torque curves are represented by sixth degree polynomials based on data obtained from a medium size centrifugal pump with specific speed $N_s=1800$. The data are available for all four quadrants of pump operation: normal pump, turbine, energy dissipation, and reverse pump. The frictional torque representation is based on measurements for CRBR prototype pumps. The implementation of the homologous pump curves and the frictional torque representation allows direct integration of the equation of motion for the pump rotor to predict pump coastdown behavior. The pump model has been validated against coastdown data from the FFTF primary pump ($N_s=1264$), the EBR-II primary pump ($N_s=1602$) and the CRBR prototype primary pump ($N_s=2069$). The good agreement with data supports the use of the homologous curves for pumps with specific speed in the range of 1200 – 2100. It is expected that the SAS4A/SASSYS-1 pump model should be applicable for modeling the CEFR primary pump ($N_s=1628$) and secondary pump ($N_s=1414$). In addition, the validity of the model is demonstrated using the homologous curves fitted to 11th degree polynomials. The results are practically identical to previous work.

The FR-Sdaso pump model for predicting steady state pump characteristics is derived based on data published by Ebara, a major Japanese pump manufacturer. Using affinity laws for pump, the pump head can be represented by a quadratic function of flow discharge and pump rotor speed. There is good agreement between the model predictions and CEFR secondary pump data for flow rate versus pump speed. For predicting transient pump coastdown behavior, the FR-Sdaso model uses an analytical method that does not require information on pump characteristics such as moment of inertia of pump impeller, hydraulic and frictional torques. This analytical method only requires two values that can be provided by the pump manufacturer: the rotor lock time and the half-time (when

the pump speed reaches one half of its steady state speed). The initial transient model, known as Model A, treats the coastdown behavior in two periods. In the first period, the rate of change of pump rotor speed is proportional to the square of speed and the solution is represented by a hyperbola function. In the second period the rate of change of pump rotor speed is assumed constant so the pump speed decreases linearly with time. The model results agree well with the CEFR pump coastdown data in the first period but reveal significant deviation in the second period. An improvement to the model is formulated to address the discrepancy seen in Model A. The improved model, known as Model B, combines the two separate formulations in Model A and its analytical solution covers the entire duration of pump coastdown. Model B also utilizes two data values, the rotor lock time and an intermediate time which is not restricted to the half-time. The Model B predictions are in excellent agreement with coastdown data from many LMR centrifugal pumps including the CEFR primary and secondary pumps, the FFTF primary pump, the EBR-II primary pump, and the CRBR prototype primary pump.

Probabilistic Risk Assessments and Component Reliability Estimates

This task centered on the assessment of component reliability and consisted of three subtasks, as outlined below:

- Development of sodium fast reactor component reliability estimates, including the methodologies to derive reliability estimates from operational data.
- A detailed reliability assessment of the CFR-600 design primary sodium pumps and steam generator modules.
- A reliability assessment of the CFR-600 passive control rod design and a description of the methodology utilized to assess the potential for common-cause-failure of reactor protection system components.

First, training materials were prepared for CIAE staff regarding the development of component reliability estimates based on available operational data. The training was delivered to CIAE staff during a visit to Beijing, China on in August 2017. The training consisted of five lectures, described in Table 1.

Table 1: Component Reliability Training Lectures Provided by Argonne

Number	Title	Description
1	<i>Use of Data in PSA and Equipment Reliability Analysis</i>	<i>An overview of the use of reliability data within different parts of a probabilistic safety assessment (PSA) and in general equipment reliability analyses performed outside of PSA.</i>
2	<i>Reliability Fundamentals</i>	<i>An overview of the basics of component reliability, such as the different types of components (operating, standby, etc.), the reliability models utilized for each type of component, and the data necessary to calculate reliability estimates.</i>
3	<i>Calculating Reliability Estimates from Data</i>	<i>An overview of how to calculate reliability estimates given data, including Bayesian versus Frequentist analysis and available tools to assist with calculations.</i>
4	<i>Methods of Data Collection</i>	<i>An overview of how to establish a data collection system for an operating facility (or fleet of operating facilities), including developing a database structure and organizing data collection.</i>
5	<i>PSA Standards and Data Analysis</i>	<i>An overview of available PSA standards, such as the ASME/ANS Non-LWR PRA standard, and the requirements associated with data collection and reliability analysis.</i>

The training lectures were well received by the CIAE audience, which included staff members from the PSA area and staff from operating facilities. Due in part to this training, a data collection and analysis system was developed by CIAE, which is being utilized to collect data from the operating CEFR facility and planned to be utilized for CFR-600. Based on analysis performed by ANL, generic component reliability estimates were developed and shared with CIAE as part of a long-term reliability sharing program (with anticipated future contributions from CIAE). In fact, CIAE initiated data sharing based on experience from CEFR before project termination. Details regarding U.S. sodium reactor component experience (such as information on individual component failures, component design, or facilities where failures occurred) were not transmitted.

Regarding the “Sodium Pump and Steam Generator Reliability” subtask, a reliability analysis was performed of the CFR-600 primary sodium pump design and steam generator module design. Design information on these systems was provided by CFR-600 for these analyses. The analysis of the primary sodium pumps focused on three PSA initiating events (loss of flow from a single pump, loss of flow from two pumps, and loss of flow from a single pump with pump seizure). Failure modes and effects analyses (FMEAs) were performed, which were utilized to develop detailed fault trees for the systems. Based on the fault trees, frequencies were developed for each of the three initiating events. A similar approach was utilized for the assessment of the steam generator modules, including an FMEA and fault tree analysis. Due to the complexity of the CFR-600 steam generator system, significant effort was necessary to assess the different failure pathways of the system and its response to different event sequences. The reliability analysis focused on PSA initiating events, including loss of heat sink and a large sodium-water interaction, which were both quantified using the fault tree approach.

Regarding the “Reactor Protection System Reliability” subtask, first a reliability analysis was performed of the CFR-600 passive control rod design. As with the previous subtask, design information was provided by CIAE regarding the component. An FMEA was performed and a fault tree constructed. In this analysis, the event of interest was the failure of the passive control rods to insert given a reactor state that would initiate their action. Given the innovative design of the system, this reliability analysis was more mechanistic in nature than those conducted in previous subtask (which were generally focused on component failures). There was also additional uncertainty in the evolving component design. The second part of this subtask was the methodology for the analysis of common-cause-failure (CCF) for components related to the reactor protection system. This included the possibility of both intra-system CCF and inter-system CCF, the latter of which has the potential to disable multiple systems simultaneously. The methodology outlined an approach to gauge the degree of vulnerability to such CCF mechanisms, including how to quantify impact of CCF within the PSA.

Technology Transfer-Intellectual Property

Argonne National Laboratory background IP:

SAS4A/SASSYS-1 liquid metal reactor accident analysis code

Participant background IP:

FR-Sdaso Sodium-cooled Fast Reactor (SFR) system and safety analysis code

Identify any new Subject Inventions as a result of this CRADA:

N/A

Summary of technology transfer benefits to industry and, if applicable, path forward/anticipated next steps towards commercialization:

The collaboration between ANL and CIAE under this CRADA complemented ongoing activities under the “Fast Reactor Methods, Modeling and Validation” technical area of DOE Office of Nuclear Energy Advanced Reactor

Technologies (ART) program by facilitating its science-based R&D goals toward verification and validation of the state-of-the-art SFR systems and safety analysis capabilities and methods.

Other information/results (papers, inventions, software, etc.):

1. IAEA-TECDOC 1703 "Benchmark Analyses on the Natural Circulation Test Performed during the Phenix End-of-Life Experiments."

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Nuclear Science and Engineering

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