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A METHODOLOGY FOR THE DEVELOPMENT OF A RELIABILITY DATABASE FOR AN ADVANCED REACTOR PROBABILISTIC RISK ASSESSMENT

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ABSTRACT

GE Hitachi Nuclear Energy (GEH) and Argonne National Laboratory are currently engaged in a joint effort to modernize and develop probabilistic risk assessment (PRA) techniques for advanced non-light water reactors. At a high level the primary outcome of this project will be the development of next-generation PRA methodologies that will enable risk-informed prioritization of safety- and reliability-focused research and development, while also identifying gaps that may be resolved through additional research. A subset of this effort is the development of a reliability database (RDB) methodology to determine applicable reliability data for inclusion in the quantification of the PRA. The RDB method developed during this project seeks to satisfy the requirements of the Data Analysis element of the ASME/ANS Non-LWR PRA standard. The RDB methodology utilizes a relevancy test to examine reliability data and determine whether it is appropriate to include as part of the reliability database for the PRA. The relevancy test compares three component properties to establish the level of similarity to components examined as part of the PRA. These properties include the component function, the component failure modes, and the environment/boundary conditions of the component. The relevancy test is used to gauge the quality of data found in a variety of sources, such as advanced reactor-specific databases, non-advanced reactor nuclear databases, and non-nuclear databases. The RDB also establishes the integration of expert judgment or separate reliability analysis with past reliability data. This paper provides details on the RDB methodology, and includes an example application of the RDB methodology for determining the reliability of the intermediate heat exchanger of a sodium fast reactor. The example explores a variety of reliability data sources, and assesses their applicability for the PRA of interest through the use of the relevancy test.

INTRODUCTION

A joint effort between General Electric–Hitachi Nuclear Energy (GEH) and Argonne National Laboratory is currently underway which intends to develop and modernize probabilistic risk assessment (PRA) techniques for advanced non-light water reactors (LWRs). The primary outcome of this project will be the development of a next-generation PRA that will satisfy anticipated regulatory requirements and enable risk-informed prioritization of safety- and reliability-focused research and development, while also identifying gaps that may be resolved through additional research. Previous efforts to construct a PRA for GEH's Power Reactor Inherently Safe Module (PRISM) (ref. 1), a sodium-cooled fast reactor (SFR), are being leveraged as a foundation for this project. Additionally, this effort is being executed in accordance with guidance provided by the recently issued ASME/ANS Non-LWR PRA (2) standard, which has been approved for trial use.

A subset of this effort is the development of PRA methodologies for the creation of the reliability database (RDB). The goal of the RDB methodology is to provide a systematic procedure to review available reliability data for its applicability to the PRISM PRA, and derive reliability estimates for the quantification of initiating event frequencies and system reliability. This report provides details on the developed RDB methodology and how the methodology satisfies the requirements of the ASME/ANS PRA standard for advanced Non-LWRs.

The first section of this work describes the data analysis requirements of the new ASME/ANS Non-LWR PRA standard (2). In particular, the Data Analysis (DA) element of the standard provides the most pertinent guidance, although not all requirements of the DA element are applicable to the current work.

The following section provides details regarding the RDB methodology along with a description of how the methodology fulfills the requirements of the Non-LWR PRA standard. The RDB methodology provides a structured approach for assembling available reliability data and gauging its applicability for use in the PRISM PRA.

The penultimate section provides an application of the MST methodology using an example from the PRISM PRA, including an example RDB analysis for the intermediate heat exchanger.

ASME/ANS PRA STANDARD FOR NON-LWRS

A key focus of the current non-LWR PRA modernization and development effort is the establishment of analysis techniques that satisfy the requirements of the recently completed ASME/ANS PRA Standard for Advanced Non-LWR Nuclear Power Plants. The Non-LWR PRA standard, which was approved for a 36-month trial use period in 2013, is the only such PRA standard document specifically created for advanced non-LWRs. Unlike the ASME/ANS PRA standards for LWRs, which address only a portion of the total PRA per each standard document, the Non-LWR PRA standard is a comprehensive document that includes all plant initiators and hazards, and covers the analysis from initiating event to offsite consequence analysis.

As with the ASME/ANS standards for LWRs, the Non-LWR PRA standard is divided into PRA elements, which are composed of high level requirements (HLRs) and supporting requirements (SRs). If each SR is fulfilled, then the HLR is fulfilled. If all HLRs are met, then the PRA element is satisfied. In total, the Non-LWR PRA standard contains 18 PRA elements, with over 200 HLRs and more than 1000 SRs for all elements.

The Non-LWR PRA standard is also designed to address PRAs during the various stages of reactor design and development. Many SRs are split into multiple Capability Categories (CCs), which vary depending on the degree of realism and plant-specificity needed for the analysis. The proper CC is selected through the PRA application process, where factors such as the life-cycle stage, site characteristics, and PRA application are assessed to determine the correct CC. For the current project, the goal is to satisfy CC II of the pertinent SRs, which are described below:

- **Scope and level of detail:** Resolution and specificity sufficient to identify the relative importance of the contributors at the component level and associated human actions, as necessary, and relevant physical phenomena and release characteristics.

- **Plant-specificity:** Use of plant-specific data/models for the significant models.

- **Realism:** Departures from realism will have small impact on the conclusion and risk insights supported by good practices.

Reliability Database Requirements

The Non-LWR PRA standard contains a separate PRA element regarding Data Analysis (DA), which contains 5 HLRs and 37 SRs. While all HLRs and SRs are important for the analysis of reliability data for the PRISM PRA, only three of the HLRs are considered applicable to the development of the RDB, as they focus on the review of reliability data sources[1].

HLR-DA-C provides guidance on reliability parameter estimation, including specifying when the use of plant-specific data is necessary or when generic data sources are acceptable. However, as will be shown in the Methodology section of the current work, many of the SRs are only applicable for operating plants, not those in the design stage.

HLR-DA-D expands on the use generic versus plant-specific data and outlines the requirements of uncertainty assessments, including the necessary level of parameter characterization and the role of sensitivity studies.

Lastly, HLR-DA-E sets the documentation requirements for the DA analysis, including the reporting of assumptions, technical basis, and limitations. This HLR is included for all PRA elements, and is a necessary step for a complete PRA.

METHODOLOGY

This section provides a high-level overview of the RDB methodology that is utilized to satisfy the Non-LWR PRA standard requirements while updating and modernizing the PRISM PRA. Table 1 provides an overview of the process, which begins with the identification of systems and events requiring reliability data, followed by a multi-step approach to reviewing data sources for relevant information. The remainder of this section details each step of the RDB evaluation procedure, and reviews how the requirements of the Non-LWR PRA standard are met.

Table 1. Reliability Database Evaluation Procedure

Step	Description
1	Identify system
2	Review data source
	a) Query SFR-specific database(s)
	b) Examine SFR-specific raw data
	c) Examine non-SFR nuclear data
	d) Examine non-nuclear data
	e) Use expert judgment/analysis
3	Update as new data becomes available

- ***Step 1**:* The system or event to be considered is identified. This includes defining the system parameters that will be used as part of the relevancy test in step 2 (function, failure modes, environment/boundary conditions).

- ***Step 2**:* The next step involves a review of available data for information applicable to the system or event of interest. The determination of applicability is completed through the use of a relevancy test. This process compares

[1] DA HLRs A and B provide guidance on establishing system boundaries, event probability types, and proper system grouping. While these aspects are important, they are considered separately from the review of reliability data.

the data source with the system or event parameters shown in Table 2. A perfectly applicable data source would have the same function and failure modes, and operate in identical conditions. While a perfectly analogous system is not always available, the relevancy test permits a grading of the applicability of the data. Similar relevancy tests have been used in past advanced reactor analyses (3).

Table 2. Data Relevancy Parameters

Relevancy Property	Description
Function	The function of the system under consideration.
Failure Mode(s)	The way(s) in which the system may fail to perform its function.
Environment and Boundary Conditions	Properties of the system, including operating environment, boundary conditions, and design characteristics.

For an SFR reliability data project, data sources are reviewed using the following priority. If the first option in the list does not contain satisfactory information, then the next source on the list is examined.

a) *SFR-Specific Databases*: A summary database that focuses on nuclear SFR reliability data.

b) *SFR-Specific Raw Data*: Plant logs and operational experience from past sodium reactors and test facilities.

c) *Non-SFR Nuclear Data:* Summary databases that focus on nuclear power plant reliability data, but not SFR-specific (such as LWR databases).

d) *Non-Nuclear Data Sources*: Databases from other industries, such as offshore oil and gas, chemical industry, electronic manufacturers, etc.

e) *Expert Judgment/Analyses*: The use of fundamental system analyses or mechanistic modeling to determine component reliability, or the use of expert judgment to determine reasonable reliability estimates based on available data, even if not directly applicable.

Due to the lack of operational experience of commercial-size SFRs in the U.S., perfectly applicable reliability data are rare. Instead, a variety of sources must usually be considered, supplemented with additional analyses, and combined using expert judgment.

- ***Step 3****:* The final stage of the analysis updates the reliability information using Bayesian techniques (such as those described in NUREG/CR-6823 (4) as new data become available.

Non-LWR PRA Requirements

Tables 3 through 5 describe how the RDB methodology satisfies the requirements of the Non-LWR PRA standard. As the tables show, many requirements only pertain to operating power plants, not those in the design or preliminary design stage. Also, many of the requirements are fairly vague, with imprecise guidance on the selection of data sources. The key to satisfaction of the standard is a systematic, repeatable RDB approach that properly reviews data sources and information before inclusion in the PRA.

Table 3. Process to Satisfy Non-LWR PRA Standard Requirement DA-C.

Index No.	Process to Satisfy Requirement
DA-C1	Basic events will be identified as part of the PRA development (Step 1 of Table 1).
DA-C2	Relevant experience from past facilities will be identified and reviewed before possible use (Step 2 of Table 1).
DA-C3	N/A (operating plants only)
DA-C4	N/A (operating plants only)
DA-C5	N/A (operating plants only)
DA-C6	N/A (operating plants only)
DA-C7	N/A (operating plants only)
DA-C8	The use of generic data for test and maintenance unavailability will be justified in Step 2.e of Table 1.
DA-C9	N/A (operating plants only)
DA-C10	N/A (operating plants only)
DA-C11	N/A (operating plants only)
DA-C12	N/A (operating plants only)
DA-C13	Consistency of unavailability for front-line and support systems will be maintained.
DA-C14	The assumptions and bases for unavailability of equipment for maintenance will be documented.
DA-C15	See DA-C14.
DA-C16	The assumptions and bases for repair times will be documented.
DA-C17	Data on the recovery of loss of off-site power, loss of service water, etc., will be collected, if available.
DA-C18	N/A (operating plants only)
DA-C19	Generic parameter estimates will not be used across multiple plant operating states unless it can be established that they are applicable.
DA-C20	N/A (operating plants only)

Table 4. Process to Satisfy Non-LWR PRA Standard Requirement DA-D.

Index No.	Process to Satisfy Requirement
DA-D1	Realistic parameter estimation will be established using relevant evidence, in conjunction with a Bayesian updating process.
DA-D2	Expert judgment will be used (and documented in Step 2.e of Table 1) when plant-specific or generic data are not available.
DA-D3	A mean value and probabilistic representation of uncertainty will be provided for significant basic events. Non-significant basic events will include a point estimate and a characterized uncertainty range.
DA-D4	N/A (operating plants only)
DA-D5	A common cause failure (CCF) approach is selected separately from the RDB methodology, but is informed by the available CCF data.
DA-D6	Generic CCF probabilities will be utilized, but supplemented with plant experience when possible.
DA-D7	Any available CCF probabilities will undergo the same screening and review process as the reliability data (Table 1).
DA-D8	N/A (operating plants only)

Table 5. Process to Satisfy Non-LWR PRA Standard Requirement DA-E.

Index No.	Process to Satisfy Requirement
DA-E1 DA-E2 DA-E3	All documentation procedures will be maintained in accordance with the requirements of the standard and GEH protocol.

APPLICATION

The following section describes the application of the RDB methodologies reviewed in preceding section for the development and modernization of the PRISM PRA. Graphically, the process follows the procedure outlined in Figure 1. The remainder of this section describes each step of the process, and includes a simplified example.[2]

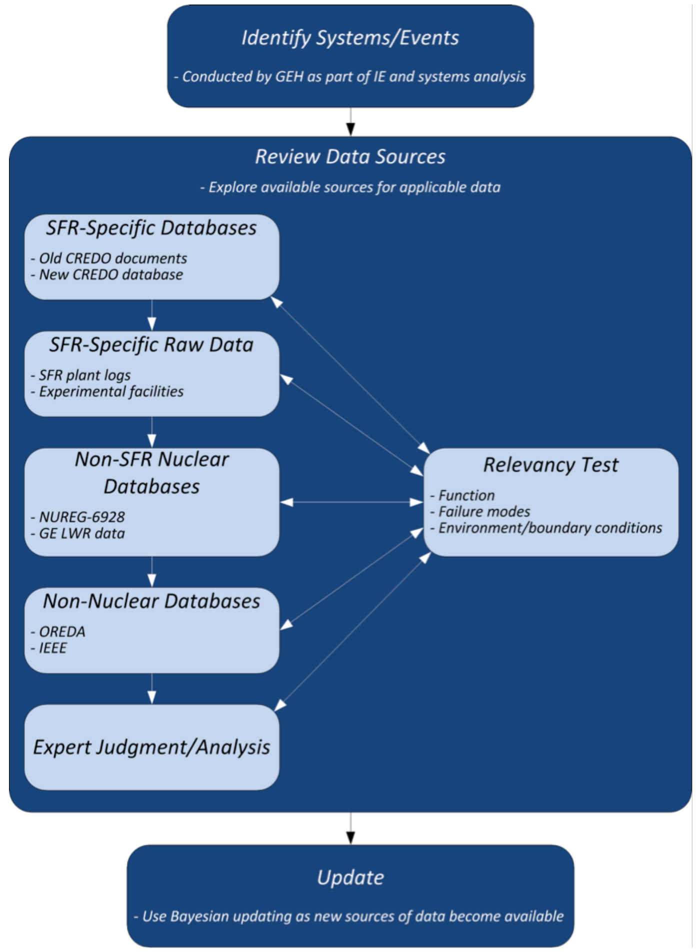

Figure 1. PRISM PRA RDB Procedure

Step 1: System Identification

The first step of the process is to identify the system or event, along with the properties to be used as part of the relevancy test in step 2. For the PRISM PRA update and modernization, an example is presented for the sodium-to-sodium intermediate heat exchanger (IHX) to demonstrate the RDB methodology.

First, the system boundaries and system function of the IHX are defined and identified, with the intent of use during the relevancy test. All system boundaries and functions are defined consistently with the systems analysis of the PRA. Table 6 contains the complete list of IHX properties.

[2] Some details of the following example have been omitted due to proprietary restrictions or access limitations, such as data from documents deemed Applied technology.

Table 6. Process to Satisfy Non-LWR PRA Standard Requirement DA-D.

Parameter	Description
Function	Exchange heat from the primary sodium coolant to the sodium in the intermediate loop
Failure Modes	IHX Shell External Leakage External Rupture IHX Tubes Leakage Rupture Plugging
Environment/Boundary Conditions	Environment: 2 Vertical IHXs within reactor vessel 420 MWt Boundary Conditions: Shell Side Fluid – Sodium Shell Side Inlet – 500°C Shell Side Outlet – 360°C Shell Side Pressure ~ 1 atm Shell Side Flowrate – 8.57E6 kg/hr Tube Side Fluid – Sodium Tube Side Inlet – 325°C Tube Side Outlet – 475°C Tube Side Pressure ~ 1 atm Tube Side Flowrate – 7.94E6 kg/hr

Step 2: Data Source Review

Next, information sources are reviewed for applicable data (through use of the relevancy test). The IHX example presented in step 1 is continued as a demonstration of the RDB methodology for the PRISM PRA update and modernization:

SFR-Specific Databases

SFR-specific databases are the first information sources reviewed. While there are U.S. Department of Energy (DOE) efforts currently underway to create a new, centralized SFR-specific reliability database, a complete, searchable database does not yet exist. However, partial summary documentation from the historical Centralized Reliability Data Organization (CREDO) database, which was operated by the DOE from the late 1970s until 1992, is still available. A search of these documents reveals a sodium-to-sodium IHX failure rate of approximately 4.4E-6 per hour, with three failure events. The CREDO documents only state that two of the events involved plugging (due to seal gas), while the third event fell in the "other" category. While these data are informative, there is no supporting documentation to assist in determining the relevancy of the results. Additional IHX system and failure event information is needed for a proper comparison through the relevancy test.

Another SFR-specific summary data source is a 1990 report by the Idaho National Engineering Laboratory (INEL), which not only reviewed data from CREDO, but other sources (5). The summary data from this source is presented in Table 7, where this data has been derived from CREDO (6), the Liquid Metal Engineering Center (LMEC) (7), a French study (LYON) (8), the Clinch River Breeder Reactor PRA (CRBRP-4), and a CRBR reliability study (GEFR-00554). Ref (4) recommends the value from CREDO, if available. For IHX shell external leakage, data from available sources were approximately within an order of magnitude. There is slightly greater variance in the values for IHX tube leakage.

In its review of CREDO IHX data, ref (5) also finds three failure events (two plugging and one "other" event), but does not consider these events applicable, and excludes them for an unstated reason. Therefore, ref (5) uses the reliability estimates from the CREDO data results seen in Table 8 (which are the basis of the "recommended" values in Table 7). As with the primary CREDO documents, this data is informative, but without additional details on the IHX population or failure events, it is difficult to perform a complete comparison as part of the relevancy test. Therefore, it is necessary to move to the next class of data sources in an attempt to gather additional data and further insight

Table 8. Ref (5) IHX CREDO Data Review

Failure Mode	Failures	Hours	Mean Failure Rate
IHX Shell			
External Leakage	0	4.97E+5	1.0E-6 /hr
External Rupture	0	4.97E+5	1.0E-6 /hr
IHX Tube			
Leakage	0	4.97E+5	1.0E-6 /hr
Plugged	0	4.97E+5	1.0E-6 /hr

Table 7. Summary IHX Reliability Data from Ref (5)

Failure Modes	Recommended (A)		LMEC		LYON		CRBRP-4		GEFR-00554	
	Mean	EF	Mean	EF	Mean	EF	Mean	EF	Mean	EF
IHX Shell										
External Leakage	1.0E-6 /hr	10			2.0E-6 /hr (B)		3.8E-7 /hr	10	3.8E-7 /hr	
External Rupture	1.0E-6 /hr	10			(C)				2.5E-13 /hr	
IHX Tubes										
Leakage	1.0E-6 /hr	10	1.6E-4 /hr (B)		(C)		4.0E-6 /hr (B)		4.0E-6 /hr (B)	
Rupture	1.0E-6 /hr	10	(C)		(C)		(C)		(C)	
Plug	1.0E-6 /hr	10			(C)					

(A) Recommended value from CREDO, if information existed. Otherwise the value was obtained from GEFR-00554. The mean and EF were rounded.
(B) This failure rate includes other failure modes.
(C) Covered by failure rate listed above.

SFR-Specific Raw Data

Without a satisfactory SFR-specific reliability database, raw data from past sodium reactors and experimental facilities must be reviewed. There are several past U.S.[3] sodium reactors and experimental facilities that utilized sodium-to-sodium heat exchangers, as shown in Table 9. To compare the IHXs located at these facilities to the PRISM IHX design, additional operational details are necessary and are provided in Table 10. The operational properties allow the first and third metrics of the relevancy test (function and environment/boundary conditions) to be evaluated. As can be seen, the PRISM IHX power rating (420 MW_t) is much higher than past U.S. experience.

Table 9. U.S. Sodium Reactor IHX Experience

Abbreviation	Facility
SCTI[1]	Sodium Component Test Installation
HNPF	Hallam Nuclear Power Facility
SRE	Sodium Reactor Experiment
Fermi 1	Fermi 1 Nuclear Power Plant
SEFOR	Southwest Experimental Fast Oxide Reactor
EBR-II	Experimental Breeder Reactor II
FFTF	Fast Flux Test Facility

[1]Test Facility

Table 10. U.S. Sodium Reactor IHX Properties

Facility	IHX Description
HNPF	Environment: 6 Vertical IHXs outside reactor vessel 42.6 MW_t Boundary Conditions (9), (10): Tube Side Fluid – Primary Sodium Tube Side Inlet Temperature – 507°C Tube Side Outlet Temperature – 321°C Tube Side Pressure – 4.4 atm Tube Side Flowrate – 6.4E5 kg/hr Shell Side Fluid – Secondary Sodium Shell Side Inlet Temperature – 290°C Shell Side Outlet Temperature – 480°C Shell Side Pressure – 5 atm Shell Side Flowrate – 5.9E5 kg/hr
SRE	Environment: 1 Horizontal IHX outside reactor vessel 20 MW_t Boundary Conditions (10): Tube Side Fluid – Primary Sodium Tube Side Inlet Temperature – 515°C Tube Side Outlet Temperature – 260°C Tube Side Flowrate – 2.2E+5 kg/hr Shell Side Fluid – Secondary Sodium Shell Side Inlet Temperature – 217°C Shell Side Outlet Temperature – 480°C Shell Side Flowrate – 2.2E+5 kg/hr
Fermi 1	Environment: 3 Vertical IHXs inside reactor vessel 143 MW_t Boundary Conditions (10): Tube Side Fluid – Secondary Sodium Tube Side Inlet Temperature – 270°C Tube Side Outlet Temperature – 437°C Tube Side Flowrate – 2.4E+6 kg/hr Shell Side Fluid – Primary Sodium Shell Side Inlet Temperature – 480°C Shell Side Outlet Temperature – 315°C Shell Side Flowrate – 2.4E+6 kg/hr
SEFOR	Environment: 2 Vertical IHXs outside reactor vessel 20 MW_t Boundary Conditions: Shell Side Fluid – Secondary Sodium Tube Side Fluid – Primary Sodium Tube Side Inlet Temperature – 438°C Tube Side Outlet Temperature – 371°C Tube Side Flowrate – 2150 gpm (510 m^3/hr)
EBR-II	Environment: 1 Vertical IHX inside reactor vessel 62.5 MW_t Boundary Conditions (11): Shell Side Fluid – Primary Sodium Shell Side Inlet Temperature – 475°C Shell Side Outlet Temperature – 370°C Shell Side Pressure ~ 1 atm Shell Side Flowrate – 1.7E6 kg/hr Tube Side Fluid – Secondary Sodium Tube Side Inlet Temperature – 308°C Tube Side Outlet Temperature – 463°C Tube Side Pressure ~1.5 atm Tube Side Flowrate – 1.1E6 kg/hr
FFTF	Environment: 3 Vertical IHXs outside reactor vessel 133 MW_t Boundary Conditions (10): Tube Side Fluid – Secondary Sodium Tube Side Inlet Temperature – 270°C Tube Side Outlet Temperature – 410°C Tube Side Flowrate – 2.6E+6 kg/hr Shell Side Fluid – Primary Sodium Shell Side Inlet Temperature – 450°C Shell Side Outlet Temperature – 315°C Shell Side Flowrate – 2.6E+6 kg/hr

Evaluation of the second relevancy test metric, failure modes, requires additional data on historic U.S. IHX system failures. Table 11 has details on four IHX failure events that occurred at the U.S. facilities listed in Table 9. The first failure event, at the SCTI, an Atomics International-constructed facility that operated from 1964 to 1995 in California, fails the relevancy test since the disruption was the fault of trapped gas, which could not be vented. This occurrence is now well understood and the PRISM will have appropriate venting

[3] U.S. facilities are the focus of the review presented here, as access to international sodium reactor reliability data is limited.

pathways, as the PRISM design process incorporates available operating experience. The second failure event at HNPF, a sodium-graphite reactor that operated from 1962 to 1964 in Nebraska, may be relevant, as the function coincides with that of the PRISM IHX, and the failure mode of tube cracking due to vibration is a possible cause. The only questionable aspects are the boundary conditions and environment, shown in Table 10. The HNPF IHX was much smaller than the designed PRISM IHX, and was located outside of the reactor vessel. The operating temperature and pressures are not grossly different from the PRISM IHX though. Therefore, the HNPF failure is considered *partially relevant*. The third fault, at EBR-II, is again *not relevant*, as it was the result of vibration in an access port, which was a fault induced by the design. Lastly, a seal gas plugging issue at FFTF is also considered *not relevant* due to the failure mode.

In essence, the review of SFR raw data returns reliability estimates very similar to the summary data from past CREDO documents (which holds since CREDO likely had much of the same information). While there is a considerable amount of experience with sodium-to-sodium IHXs, the sizes of the IHXs were much smaller than what is designed for the PRISM reactor. However, many of the IHXs operated for a considerable number of hours with few to no failures, such as at EBR-II. Therefore, the SFR raw data appear to support the reliability estimates made by CREDO (and ref (5))

Non-SFR Nuclear Databases

There are many non-SFR nuclear data sources available. One of the most widely cited databases is NUREG/CR-6928 (12), which summarizes data from the U.S. nuclear industry. A search of NUREG/CR-6928 for heat exchanger (HX) related events (since intermediate heat exchangers are not used at the currently operating LWRs) reveals the results Table 12. The large LWR reactor population and decades of operational experience result in a fairly well developed database.

Table 12. Non-SFR Nuclear IHX Data from Ref (12)

Failure Mode	Failures	Hours	Mean Failure Rate
HX Shell			
External Leakage Small	60	222547790	3.34E-7 /hr
External Leakage Large	-	-	2.34E-8 /hr
HX Tube			
External Leakage Small	78	222547790	3.79E-7 /hr
External Leakage Large	-	-	7.58E-9 /hr
HX Loss of Heat Transfer	82	222547790	4.57E-7 /hr

However, the data in NUREG/CR-6928 do not satisfy several items in the relevancy test. First, the HXs used in LWRs do not contain sodium on either the shell or tube side. Second, the HXs in LWRs are not used for power production heat transfer, but for secondary or auxiliary functions, such as residual heat removal or component cooling water. Third, the operating temperatures and pressures of the LWR HXs are usually quite different than the sodium-to-sodium IHX in PRISM. In LWR HXs, the temperatures are usually much lower than what is seen in a pool type SFR, and pressures are typically much higher than SFRs, which operate at approximately atmospheric pressure.

Although the non-SFR nuclear data in NUREG/CR-6928 are not directly applicable to the PRISM IHX, they do provide insight into failure rates. As Table 12 shows, the LWR HX failure mode rates are approximately 1E-7 and 1E-8 per hour, which is at least an order of magnitude lower than the sodium IHX failure rates suggested by CREDO, despite operating pressures that are typically higher than sodium IHXs. Much greater operational experience and the use of water may account for some of this improved performance, but the difference is note-worthy.

Table 11. U.S. SFR IHX Events

Facility	Event	Operating Hours	Source	Relevancy Test
SCTI	Minor Malfunction - Original piping did not include a cover gas vent from the top of the IHX shell side. Gas was trapped between the sodium inlet nozzle and the upper tubesheet, reducing heat transfer.	611	SCTI, incident report No. 46 (described in ref (7))	*Not relevant due to failure mode*: cover gas venting during sodium fill is now well known and understood.
HNPF	Major Malfunction - Tubes cracked and leaked as a result of flow induced vibration.	5,640	NAA-SR-10743 (11-18-62) (described in ref (7))	*Partially relevant*: Function and failure mode applicable, environment is different, boundary conditions slightly different
EBR-II	Access port vibration and wear. The section was cut out of the inlet elbow and was re-welded in place and the secondary system was restored to operational status. Quiet operation of the IHX verified that the repair was successful.	44,000	ANL-EBR-R47, ANL-7834	*Not relevant due to failure mode*: vibration at access port
FFTF	Seal gas plugging issue (access limitation)		Access limitation	*Not relevant due to failure mode*: Access limitation

Non-Nuclear Databases

The next step on the search for reliability information is non-nuclear data sources, such as the OREDA offshore reliability database (13). However, for the IHX, it is very unlikely that a similar sodium-to-sodium heat exchanger is present in any industry outside of the nuclear sector. Therefore, there are no applicable reliability data from non-nuclear data sources.

Expert Judgment/Analysis

After exhausting all data sources, expert judgment and/or additional analyses are used to determine preliminary component reliability estimates. This process involves consideration of all available data, even if not completely relevant. For the IHX design presented here, the CREDO database (and ref (5)), along with the SFR raw data search, provide a good starting point for preliminary reliability estimates. These data are based on sodium-to-sodium IHXs that were operated at SFRs, or test facilities. While the IHX designs and failure modes incorporated in this data might not be directly equivalent to the PRISM IHX, they are at least representative of the same class of component.

Additionally, there have been attempts in the past to determine sodium-to-sodium IHX failure rates for sodium reactors. As part of the Clinch River Breeder Reactor (CRBR) project, a PRA was performed. Tube (secondary sodium) leakage was assumed to occur once in the plant's lifetime (3.8E-6 /hr-IHX). This was later revised to 4.0E-6 per hour per IHX based on study of available reliability data at the time[4]. It was assumed that primary (shell) leakage occurred with 1/10th frequency of tube leakage (3.8E7 /hr-IHX). Primary side (shell) rupture was assumed to occur 2.5E-13 per hour per IHX, based on a structural analysis.

Taken together, the historical data from CREDO/sodium reactor experience and the CRBR analyses are within an order of magnitude (~10^{-6} per hour) regarding IHX shell and tube leakage. These values are one to two orders of magnitude higher than what is seen in LWR heat exchangers. Arguments can be made that the PRISM IHX should be more reliable than LWR experience, due to lower operating pressures, but there is also reduced operating experience with a sodium-to-sodium IHXs of comparable size of PRISM, in addition to higher operating temperatures.

Therefore, based on the available data, the failure rate in the INEL CREDO study, of approximately 1.0E-6 per hour for tube leakage, appears reasonable. However, shell leakage is assumed to occur with 1/10th the frequency of tube leakage (as with the CRBR assumptions). Rupture, of the tube or shell, is assumed to occur with 1/100th the frequency of leakage, due to the very low pressure differential across the IHX tubes and shell. Although no tube plugging events were found during the review of past data, the tube plugging rate was conservatively assumed to occur at the same rate as tube leakage[5].

Table 13. Recommended IHX Reliability Values

Component Type	Failure Modes	Recommended Mean	Recommended EF
IHX Shell			
	External Leakage	1.0E-7 /hr	10
	External Rupture	1.0E-9 /hr	10
IHX Tube			
	Leakage	1.0E-6 /hr	10
	Plugged	1.0E-6 /hr	10
	Rupture	1.0E-8 /hr	10

Step 3: Estimate Updating

In the final step of the RDB process, the reliability estimates are updated, using Bayesian methods, if new, applicable data are found. This process is common in the nuclear industry, and will the follow the guidelines set by the NRC in NUREG/CR-6823 (4). This step has not yet been completed for the IHX example presented here. Please see ref (4) for additional detail on the updating process.

SUMMARY AND CONCLUDING REMARKS

As part of an effort to modernize and develop PRA techniques for advanced on-LWRs, a methodology was developed for the creation of a reliability database. The reliability database method utilizes a relevancy test to determine the applicability of existing reliability data for the system under examination. The relevancy test examines the function, failure modes, and environment/boundary conditions of the system to determine applicability. The methodology described throughout this paper satisfies the data analysis (DA) element of the recently issued ASME/ANS Non-LWR PRA Standard.

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4 This is the number shown in Table 7 under "GEFR-00554."

5 The most likely cause of IHX tube plugging is sodium impurities or radionuclides in the sodium from failed fuel pins plating on the colder IHX surface. The "plugging" failure rate could likely be reduced with additional analyses investigating the impurity level of the sodium and the likely radionuclide release from stochastically failed fuel pins.

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