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Software Requirements Specification for RELAP-7

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CONTENTS

1. INTRODUCTION

1.1. Purpose of this Plan

The purpose of this document is to specify the software requirements necessary to achieve the function and capabilities presented in the RELAP-7 Theory Manual. This specification is intended for use by those performing nuclear power plant (NPP) reactor systems safety analysis.

1.2. System Scope

The RELAP-7 (Reactor Excursion and Leak Analysis Program) will be the next generation nuclear reactor system safety analysis code developed at Idaho National Laboratory (INL). The code will be based on the INL's modern scientific software development framework MOOSE (Multi-Physics Object Oriented Simulation Environment). The overall design goal of RELAP-7 is to take advantage of the previous thirty years of advancements in computer architecture, software design, numerical integration methods, and physical models.

RELAP-7 will become the main reactor systems simulation toolkit for LWRS (Light Water Reactor Sustainability) programs RISMC (Risk Informed Safety Margin Characterization) effort and the next generation tool in the RELAP reactor systems analysis application series. The key to the success of RELAP-7 is the simultaneous advancement of physical models, numerical methods, and software design while maintaining a solid user perspective. Physical models include both PDEs (Partial Differential Equations) and ODEs (Ordinary Differential Equations) and experimental based closure models. RELAP-7 will utilize well-posed governing equations for two-phase flow, which can be strictly verified in a modern verification and validation effort. RELAP-7 will use modern numerical methods, which allow implicit time integration, second-order schemes in both time and space, and strongly coupled multiphysics.

1.3. Constraints

1.3.1. Policies and Regulations

The system shall adhere to strict software quality requirements meeting For ASME NQA-1-2008 with the NQA1a2009 addenda standards.

The system shall comply with the requirements of LWP-13620, Managing Information Technology Assets.

1.3.2. Environmental, Health, Safety, and Security Considerations

There are no environmental, health, safety, nor security considerations that specifically apply to the installation and use of RELAP-7.

1.3.3. System Criticality

The system quality level has been determined to be QL-1 as documented in Quality Level Determination No. ALL-000833 and Safety Software Determination No. SSD-000635.

1.3.4. Control of Information

For many of the potential users, there will be a need to insert proprietary data (e.g. fuel material properties) and correlations (e.g. CHF correlations) into the code, otherwise the code will have limited utility to these users.

1.4. Assumptions and Dependencies

The core dependency for RELAP-7 to function is the MOOSE framework. MOOSE provides a general platform for solving the physics problems of interest for the simulation of NPP systems behavior. MOOSE provides capabilities for solving a general set of coupled partial differential equations using the finite element method. This includes support for equation solvers, file input/output, infrastructure for communication between processors in a parallel high performance computing environment, finite element data structures and functions, and a modular architecture that permits incorporation of physics models within an application derived from MOOSE.

The physical models provided by RELAP-7 are implemented as classes that are derived from MOOSE classes that provide pluggable interfaces for basic functionality needed by any physics code based on MOOSE. These include constructs such as Kernels, Boundary Conditions, etc.

RELAP-7 adds physics models to the base capabilities provided by MOOSE to provide a powerful, flexible tool for modeling NPP systems behavior during normal operation and a variety of accident conditions.

Minimum system requirements are:

- A POSIX compliant UNIX including the two most recent versions of OS X and most current versions of Linux.
- RAM: 4GB for optimized compilation (8GB for debug compilation), 2GB per core execution
- Disk: 100 GB
- Compilers: gcc, clang, or Intel
- Python 2.6+
- Git

1.5. Applicable Standards

- PLN-4005, SQAP for MOOSE and MOOSE-Based Applications
- INL/EXT-14-31366 (Revision 2), RELAP-7 Theory Manual

1.6. Apportioning of Requirements

In some instances, the models reported in this initial version of the Software Requirements Specification cover phenomena which are not yet implemented, for example, the species balance equation for two phase flows. But when it made sense to include derivations, which we have already developed, or descriptions of models which are currently ongoing, such as the entropy viscosity method, we have included such.

2. STAKEHOLDERS NEEDS

2.1. Stakeholders

The stakeholders for RELAP-7 include DOE's LWRS Program. Once the code is developed to a state of maturity, other stakeholders will include DOE's Consortium of Advanced Simulation of Light Water Reactors (CASL) Program, DOE's Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program, and the Electric Power Research Institute (EPRI).

2.2. Stakeholder Needs

The risk-informed safety margin characterization (RISMC) methodology of the LWRS Program aims at developing and demonstrating a risk-assessment method coupled to safety margin quantification that can be used by decision makers as part of their margin recovery strategies. Toward that end, an advanced RISMC toolkit is being created, which enables more accurate representation of NPP safety margins. The RELAP-7 code is an important component of the advanced RISMC toolkit to perform NPP systems safety analyses.

3. SYSTEM OVERVIEW

3.1. System Description

The purpose of RELAP-7 is to be the next-generation NPP systems safety analysis code. NPP systems safety analysis codes are constructed to reduce the extreme complexity and physical multi-dimensionality of NPPs into a tractable numerical computational problem. They are designed to simulate licensing basis transients and both postulated and actual NPP accidents. RELAP-7 will be a new thermal-hydraulics code that employs one-dimensional (1-D) flow networks of pipes, combinations of simple zero-dimensional (0-D), 1-D, and two-dimensional (2-D) structures to represent complicated reactor components, and a 0-D point-kinetics representation to model the design features of reactor systems.

RELAP-7 will incorporate a unique software design, will use advanced numerical methods, and will be flexible in fidelity of physics representation through a sophisticated multiphysics coupling capability. Furthermore, the numerical algorithm to be implemented in RELAP-7 is capable of generating significantly reduced numerical error through the use of high-order spatial and temporal integration. RELAP-7 NPP simulation capabilities will focus on the current fleet of boiling water reactor (BWR) and pressurized water reactor (PWR) NPPs. Additional components including PWR-specific and BWR- specific components and additional closures are anticipated in future years for advanced light water reactor (LWR) concepts, such as small modular reactors (SMRs).

3.2. Components

The RELAP-7 code is intended to be an advanced system analysis tool based on components to represent the major physical processes in the reactor system. A real reactor system is very complex and contains hundreds of different physical components. It is impractical to resolve the real geometry of the entire system. Instead simplified thermal hydraulic models are used to represent (via "nodalization") the major physical components and describe the major physical processes (such as fluids flow and heat transfer). There are four main types of components: (1) 1-D components describing the piping network of the reactor system, (2) 2-D heat structures that simulate heat conduction, (3) zero-dimensional (0-D) components for setting boundary conditions, and (4) 0-D components for connecting 1-D components.

3.2.1. Pipe

Pipe is the most basic component. It is a 1-D component which simulates thermal fluids flow in a pipe. It can be either single or two phase flow.

3.2.2. Boundaries

Pipe or duct inlets and outlets, as well as pipe or duct closed ends or free boundary are treated as zero dimensional (0-D) components for setting boundary conditions.

3.2.3. Pipe with Heat Structure

The pipe with heat structure component will simulate fluid flow in a 1-D pipe coupled with 2-D heat conduction through the pipe wall. The adiabatic, Dirichlet, or convective boundary conditions could be prescribed at the outer surface of the pipe wall. The heat structure geometry could be either of a plate or cylindrical type.

3.2.4. Core Channel

The core channel component simulates the coolant flow and heat conduction inside a fuel rod as well as the conjugate heat transfer (CHT) between the coolant and the fuel rod. The fuel rod will be modeled as 2-D heat conduction structure adjacent to the 1-D fluid flow channel model. Both plate type fuel rod and cylindrical fuel rod type could be simulated. The 2-D heat structure representing solid fuel will be able to deal with typical LWR fuel rod geometry with clad/gap/fuel pellet sub-geometries. The single and two-phase fluid flow models and CHT model will be strongly coupled.

3.2.5. Branch

The pipe network branch model will be a 0-D component representing a joint/junction model for arbitrary number of conjoined pipes. The branch component will have loss coefficients that will be provided by industry manufacturers.

3.2.6. Pump

The pump is a 0-D junction component where the fluid enters the pump near the axis and the rotor accelerates the fluid to high speed. The fluid then passes through a diffuser which is a progressively enlarging pipe, which permits recovery of the dynamic NPP operating pressure.

3.2.7. Jet Pump

Jet pumps are considered as part of the reactor cooling system. The pumps are installed in the annular region between the core barrel and the outside reactor vessel. The driving head for these jet pumps is provided by the pressure head provided by the discharge of the recirculation pumps.

3.2.8. Turbine

A turbine is a 0-D component that extracts thermal energy from pressurized steam and uses it to do mechanical work on a rotating output shaft to generate large quantities of electrical power.

3.2.9. Separator/Dryer

The separator dryer is a 0-D component that will model both the steam separator and moisture dryer in BWRs and the steam generator system components in PWRs. The separator removes water from the two-phase flow and the dryer further increases the quality of the steam and provides high quality dry steam to the turbine.

3.2.10. Downcomer

The downcomer is a reactor component with a large volume that connects the feedwater pipe and the downcomer outlet. The volume is filled with vapor at the top and liquid at the bottom. During transients, the liquid level will increase or decrease (depending on the nature of the transient), which affects the mass flow rate through the reactor core; therefore, it is important to track the liquid level for transient analysis.

3.2.11. Valves

The valve component connects one pipe on each side. The valve is initiated with a given state, i.e. fully open or fully closed. It then starts to react (i.e., close or open) and is triggered either by a preset user given trigger time or by a trigger event. In its opening status, either fully open or partially open, it serves as a regular flow junction with form losses. In its fully closed status, the connected two pipes are physically isolated. The current valve model also includes the gradually open/close capability similar to a motor driven valve to simulate the physical behavior of a valve open/close procedure. It also has the benefit of avoiding spurious numerical oscillations that are caused by an instantaneous open/close procedure.

3.2.12. Wet Well

The wet well component is designed to simulate the suppression chamber of a BWR reactor, which is composed of water space and gas space. In the event of station blackout, the suppression pool of the wet well condenses the steam from the reactor and provides the cooling water to the reactor.

3.2.13. Subchannel

The subchannel component models the local conditions of fluid flow and heat transfer within fuel assemblies so that the margin to the established thermal limits, such as the departure from nucleate boiling ratio (DNBR) for PWRs, or the minimum critical power ratio for BWRs can be quantified.

3.2.14. Point Kinetics

The reactor point kinetics model is the simplest model that can be used to compute the transient behavior of the neutron fission power in a nuclear reactor. The power is computed using the space-independent, or point kinetics, approximation which assumes that power can be separated into space and time functions. The point kinetics model will compute both the immediate (prompt and delayed neutrons) fission power and the power from decay of fission products. The immediate power is that released at the time of fission and includes power from kinetic energy of the fission products and neutron moderation. Decay power is generated as the fission products undergo radioactive decay. The decay power model will be based on the 1979 ANSI/ANS Standard, the 1994 ANSI/ANS Standard, or the 2005 ANSI/ANS Standard.

3.2.15. Accumulator

An accumulator is a tank connected a cold leg of a PWR. It contains large amounts of borated water with a pressurized nitrogen gas bubble in the top. If the pressure of the primary system drops below the low pressure set point, the nitrogen will force the borated water out of the tank and into the reactor coolant system. These tanks are designed to provide water to the reactor coolant system during emergencies in which the pressure of the primary drops very rapidly, such as large primary breaks. The cold leg accumulators do not require electrical power to operate.

3.2.16. Pressurizer

The pressurizer is the component in the reactor coolant system which provides a means of controlling the system pressure. Pressure is controlled by the use of electrical heaters, pressurizer spray, power operated relief valves, and safety valves. If the pressure increases and exceeds the desired setpoint, the spray line will spray cold water into the steam space. The cold water will condense the steam into water, which will reduce pressure. If pressure continues to increase, the pressurizer relief valves will open and dump steam to the pressurizer relief tank. If this does not relieve pressure, the safety valves will lift, also discharging to the pressurizer relief tank. If pressure starts to decrease, the electrical heaters will be turned on to boil more water into steam, and therefore increase pressure.

3.3. Problem Scenarios

RELAP-7 is intended to be used for the system thermal-hydraulics simulation for a wide range of phenomena and events including licensing basis transients and accidents. The problem scenarios are briefly described here.

3.3.1. Natural Circulation

Natural circulation refers to the ability of a fluid in a system to circulate continuously due to gravitational effects upon differences of fluid density. The difference of density is the only driving force in natural circulation. A fluid system designed for natural circulation will have a heat source and a heat sink. Each of these is in contact with some of the fluid in the system, but not all of it. The heat source is positioned lower than the heat sink.

3.3.2. Water Hammer

Water hammer is a pressure wave caused when a fluid in motion is forced to stop or change direction suddenly. A water hammer commonly occurs when a valve closes suddenly at an end of a pipeline system, and a pressure wave propagates in the pipe. It is also called hydraulic shock. This pressure wave can cause major problems, ranging from noise and vibration to pipe collapse.

3.3.3. Large Break Loss of Coolant Accidents

Large break loss of coolant accident (LBLOCA) is defined as the loss of reactor coolant caused by a double-ended break of the largest coolant pipe. There are four phases involved in an LBLOCA: (1) blowdown, (2) refill, (3) reflood and (4) long term cooling. The blowdown period (0-30 s) occurs as a result of a break in the coolant system through which the primary coolant is rapidly expelled. Within a fraction of a second after the break, the core voids and goes through departure from nucleate boiling. The negative void reactivity rapidly shuts down the core.

The refill period occurs between 30 and 40 s following the start of the LOCA. The primary pressure has decreased to a level at which the low-pressure injection system activates and begins to inject water into the system. The lower plenum begins to fill with accumulator water as coolant bypass diminishes. The reflood period occurs between 40 and 200 s; it begins at the time when the lower plenum has filled and the core begins to refill.

3.3.4. Small Break Loss of Coolant Accidents

The small break loss of coolant accident (SBLOCA) is defined for piping breaks up to sizes where the reactor remains pressurized despite the occurrence of the break. This encompasses up to 3 inches diameter holes in the primary circuit piping. In the SBLOCA, the reactor depressurizes more slowly than in the LBLOCA, following a different set of physical phenomena. Since the core remains at high pressure for a long time in a SBLOCA, it is not possible to activate the Low Pressure Safety Injection (LPSI) system, with its relatively large rate of coolant flow, until a late stage into the accident.

3.3.5. Increase in Heat Removal Events

For PWRs the increase in heat removal events includes:

- Increase in feedwater flow
- Decrease in feedwater temperature
- Inadvertent opening of a secondary relief or safety valve
- Steam line break

For BWRs the increase in heat removal events include:

- Loss of a feedwater heater
- Steam pressure regulator malfunction or failure that results in an increase in steam flow

The safety concern for the increase in heat removal events is that the resulting moderator overcooling will cause a positive reactivity insertion and challenge the specified acceptable fuel design limits. Therefore the phenomena and processes of interest are related to the excessive heat transfer and the reactivity addition phenomena and processes.

3.3.6. Decrease in Heat Removal Events

The decrease in heat removal events includes:

- Turbine trip
- Loss of load
- MSIV closure
- Steam pressure regulator failure (BWR)
- Loss of main feedwater
- Loss of condenser vacuum

• Loss of non-emergency AC power
• Feedwater line break (PWR)

The safety concerns for the decrease in heat removal events are that the reactor will overheat and challenge the specified acceptable fuel design limits, and that the RCS pressure will challenge the design overpressure limit. Therefore the phenomena and processes of interest are related to the mismatch between the heat source and heat sink, and the pressurization and pressure relief processes and phenomena.

3.3.7. Reactivity and Power Distribution Anomaly Events

The reactivity and power distribution anomalies events includes:

• Control rod bank withdrawal at zero power event
• Control rod bank withdrawal at power event
• Single control rod withdrawal at power event
• Control rod assembly drop accident at cold and zero power (BWR)
• Control rod drop at power (PWR)
• Rod ejection accident (PWR)
• Moderator dilution events (PWR)
• Reactor coolant pump startup event (PWR)
• Flow controller malfunction causing an increase in core flow rate (BWR)

The safety concerns for the reactivity and power distribution anomalies events are that the reactor will overheat and challenge the specified acceptable fuel design limits, and that the RCS pressure will challenge the design overpressure limit. Therefore the phenomena and processes of interest are related to the mismatch between the reactor power distribution and the core cooling capability, and the mismatch between the heat source that causes RCS pressurization and the pressure relief processes and phenomena.

3.3.8. Boiling Water Reactor Instability Events

BWR instabilities occur when an operating condition becomes unstable after some perturbations in system parameters. As a consequence, state variables identifying the reactor working conditions are observed to oscillate in different ways depending on the modalities of the departure from the stable operating point. Power oscillations can, for large amplitudes, have an adverse influence on the fuel integrity. From the point of view of the BWR safety, the most important type of power instability is the reactivity oscillations excited by thermal-hydraulic mechanisms.

Two types of instability by reactivity have been characterized:

1) In-phase (core-wide) instability. In this case, all the variables (power, mass flow, pressure, etc.) oscillate in phase determining a

limit cycle; from the point of view of safety, this type of instability has relatively small relevance, unless it is associated with an ATWS.

2) Out-of-phase instability. In this case, the instabilities occur when a neutronic azimuthal mode is excited by thermal-hydraulic mechanisms causing asymmetric power oscillations, while part of the reactor operates at high-mass flow and low-power level, in the other part the opposite happens; this behaviour warrants detailed studies because of safety implications.

3.3.9. Anticipated Transient without Scram Events

The anticipated transients without scram (ATWS) include all anticipated operational occurrences (AOOs) with an assumed failure of the reactor protection system to trip the reactor. All PWRs have an ATWS mitigation circuit (AMSAC) to assist in event mitigation by tripping the main turbine and actuating the auxiliary feedwater system. The Babcock & Wilcox and Combustion Engineering PWRs also have a diverse scram system that successfully trips the reactor and the transient response is essentially the same. The other PWRs rely on the AMSAC actuation and reactivity feedback and operator action to mitigate the ATWS event, including injection of boric acid. The BWR strategy for mitigating ATWS events combines the alternate rod injection system, the standby liquid control system to inject boric acid and negative reactivity, and an automatic recirculation pump trip. The BWR ATWS response for some events may involve large power oscillations. The safety concern for the ATWS event is to maintain core cooling during the mismatch between reactor power and the heat sink, and for RCS pressure to stay below the RCS pressure limit that is applicable for the ATWS event.

3.3.10. Station Blackout

The station blackout (SBO) scenario causes a prolonged loss of all AC power. The reactor immediately trips and then only steam-driven cooling water sources or possibly special safety systems designed for SBO-like events are actuated. For PWRs the event is typically mitigated by the turbine-driven auxiliary feedwater pump (or an alternative pump that does not rely on normal or emergency site AC power). Loss of cooling to the reactor coolant pump seals is assumed (unless an alternative seal cooling system is available) and a SBLOCA results from seal failure. So, the event can start as a decrease in secondary heat removal event with natural circulation conditions, followed by a SBLOCA leading to core uncovery. For BWRs the event is mitigated by the turbine-driven ECCS pumps, which are of different designs. A loss of DC battery power can lead to a loss of control to these pumps and a gradual reduction in vessel inventory due to boiloff leading to core uncovery. The safety concern for the SBO event is to maintain core cooling until a source of electrical power to the safeguards systems can be restored.

3.3.11. PWR Thermal Shock

PWR overcooling events have a risk of causing a thermal shock to the reactor vessel welds. Overcooling can result from loss of secondary pressure control (steam line breaks and valve failures), steam generator overfeeds, and following actuation of the high pressure injection system due to the direct injection of cold water into the cold legs. These events are partially mitigated by the engineered safeguards that close the MSIVs and isolated main feedwater. Operator action is required to throttle or terminate high pressure injection system to prevent excessive overcooling and to control pressure. The phenomena of interest are associated with the sources of the overcooling and the pressurization and pressure relief processes.

3.4. Users Interaction

3.4.1. Developers

A developer is a scientist or engineer that develops additional physics modules to expand or improve the capabilities of RELAP-7.

This user will typically have a background in modeling and simulation techniques and/or numerical analysis but may only have a limited skill-set when it comes to object-oriented coding and the C++ language. They will be responsible for following and enforcing the appropriate software development standards. They will be responsible for designing, implementing and maintaining the software.

3.4.2. Analysts

These are users that will run RELAP-7 on an end application to perform nuclear reactor system safety analysis. They will also perform various post processing steps on the simulations they perform. These users may interact with developers of the system requesting new features and reporting bugs found and will typically make heavy use of the input file format.

3.4.3. Interface with Users

The system shall interact with users in two ways: via command line and via graphical user interface.

The system shall use the command line as defined by MOOSE.

The system shall employ a GUI for building a model of a power plant and specifying elements of a virtual control room.

4. SYSTEM REQUIREMENTS

The system is dependent on MOOSE for performance requirements and only specifies technical/functional requirements that are performed by the system.

4.1. Technical/Functional Requirements

4.1.1. General Functional Requirements

F1.1	The system shall allow specifying global gravity vector
F1.2	The system shall check the integrity of the input file prior to the execution

4.1.2. Pipe Functional Requirements

F2.1	The system shall provide a straight pipe with single phase flow
F2.2	The system shall provide a straight pipe with two phase flow
F2.3	The system shall allow specifying a position of the pipe
F2.4	The system shall allow specifying a length of the pipe
F2.5	The system shall allow specifying initial condition on the pipe
F2.6	The system shall allow specifying cross-sectional area, heat flux perimeter and hydraulic diameter
F2.7	The system shall allow specifying heat flux coming to/from the pipe
F2.8	The system shall allow specifying wall temperature
F2.9	The system shall allow prescribing a friction coefficient
F2.10	The system shall allow prescribing a heat convective coefficient
F2.11	The system shall allow prescribing a interfacial drag coefficient for two-phase flow
F2.12	The system shall allow choosing the fluid properties used in the pipe
F2.13	The system shall allow specifying heat transfer geometry
F2.14	The system shall allow for a bend pipe
F2.15	The system shall allow for controlling the wall temperature
F2.16	The system shall allow for controlling the convective heat transfer coefficient

4.1.3. Pipe with Heat Structure Functional Requirements

F3.1	The system shall allow for a pipe with a heat structure attached to it
F3.2	The system shall allow specifying adiabatic boundary condition in the heat structure
F3.3	The system shall allow for specifying convective boundary condition in the heat structure
F3.4	The system shall allow for specifying temperature boundary condition in the heat structure
F3.5	The system shall allow for specifying additional heat source in the flow channel
F3.6	The system shall allow for specifying initial temperature in the heat structure
F3.7	The system shall allow for specifying materials in the heat structure

4.1.4. Inlet Functional Requirements

F4.1	The system shall allow prescribing mass flow rate and temperature for single phase flow
F4.2	The system shall allow prescribing mass flow rate and temperature for two phase flow.
F4.3	The system shall allow prescribing density and velocity for single phase flow
F4.4	The system shall allow prescribing density and velocity for two phase flow
F4.5	The system shall allow prescribing stagnation enthalpy and momentum for single phase flow
F4.6	The system shall allow prescribing stagnation enthalpy and momentum for two phase flow
F4.7	The system shall allow prescribing stagnation pressure and temperature for single phase flow
F4.8	The system shall allow prescribing stagnation pressure and temperature for two phase flow
F4.9	The system shall allow reversible inlet conditions

F4.10	The system shall allow for controlling the mass flow rate and temperature for single phase flow.
F4.11	The system shall allow for controlling the mass flow rate and temperature for two phase flow.
F4.12	The system shall allow for controlling the density and velocity for single phase flow.
F4.13	The system shall allow for controlling the density and velocity for two phase flow.
F4.14	The system shall allow for controlling the stagnation pressure and temperature for single phase flow.
F4.15	The system shall allow for controlling the stagnation pressure and temperature for two phase flow.

4.1.5. Outlet Functional Requirements

F5.1	The system shall allow prescribing an outlet pressure for single phase flow
F5.2	The system shall allow prescribing an outlet pressure for two phase flow.
F5.3	The system shall allow reversible outlet conditions
F5.4	The system shall allow for controlling the outlet pressure for single phase flow.
F5.5	The system shall allow for controlling the outlet pressure for two phase flow.

4.1.6. Closed End Functional Requirements

F6.1	The system shall allow prescribing a solid wall boundary condition for single phase flow
F6.2	The system shall allow prescribing a solid wall boundary condition for two phase flow

4.1.7. Free Boundary Functional Requirements

F7.1	The system shall allow prescribing a free inflow/outflow boundary condition for single phase flow

F7.2	The system shall allow prescribing a free inflow/outflow boundary condition for two phase flow

4.1.8. Heat Structure Functional Requirements

F8.1	The system shall provide a 2-D heat structure
F8.2	The system shall allow specifying geometry of the heat structure and its blocks
F8.3	The system shall allow axisymmetric heat structure
F8.4	The system shall allow Cartesian heat structure
F8.5	The system shall allow multiple blocks within one heat structure
F8.6	The system shall allow defining material properties in the heat structure blocks
F8.7	The system shall allow defining initial temperature of the heat structure
F8.8	The system shall allow specifying power profile in the heat structure
F8.9	The system shall allow specifying power fraction of the reactor power
F8.10	The system shall allow specifying power density in the heat structure

4.1.9. Core Channel Functional Requirements

F9.1	The system shall provide core channel with single phase model
F9.2	The system shall provide core channel with two phase model

4.1.10. Reactor Power Functional Requirements

F10.1	The system shall provide for fully coupled native point kinetics model with thermal-hydraulic reactivity feedback for simulating strong reactor transients.
F10.2	The system shall support specifying reactor power in order to simulate normal transient reactor operation, such as startup, shutdown, and load following.

4.1.11. Junction Functional Requirements

F11.1	The system shall allow for connecting 2 single-phase pipes together
F11.2	The system shall allow for connecting 2 two-phase pipes together
F11.3	The system shall allow for connecting one pipe perpendicular to another pipes (T-shape) (single phase flow model)
F11.4	The system shall allow for connecting one pipe perpendicular to another pipes (T-shape) (two phase flow model)
F11.5	The system shall allow for connecting multiple pipes on one end to multiple pipes on the other end (all pipes with single phase flow model).
F11.6	The system shall allow for connecting multiple pipes on one end to multiple pipes on the other end (all pipes with two phase flow model).

4.1.12. Pump Functional Requirements

F12.1	The system shall provide a pump model for single phase flow.
F12.2	The system shall provide a pump model for two phase flow
F12.3	The system shall allow for driving the pump with a turbine component
F12.4	The system shall allow for controlling the pump head.

4.1.13. Jet Pump Functional Requirements

F13.1	The system shall provide a jet pump model for BWR analysis.

4.1.14. Turbine Functional Requirements

F14.1	The system shall provide power turbine model for single phase flow
F14.2	The system shall provide power turbine model for two phase flow
F14.3	The system shall provide Terry turbine model for single phase flow
F14.4	The system shall provide Terry turbine model for two phase flow

4.1.15. Separator/Dryer Functional Requirements

F15.1	The system shall provide a model to remove water from the two-phase flow and provide high quality steam to the power turbine

4.1.16. DownComer Functional Requirements

F16.1	The system shall provide representative water level in the BWR downcomer

4.1.17. Valves Functional Requirements

F17.1	The system shall provide the control and functions of a valve for single phase flow
F17.2	The system shall provide the control and functions of a check valve for single phase flow
F17.3	The system shall provide the control and functions of a valve for two phase flow
F17.4	The system shall provide the control and functions of a check valve for two phase flow

4.1.18. Wet Well Functional Requirements

F18.1	The system shall provide a model for representative pressure, water level, water and gas mixture temperatures in the wet well of a BWR

4.1.19. Accumulator Functional Requirements

F19.1	The system shall provide a model for representative water level in the accumulator during transient conditions

4.1.20. Pressurizer Functional Requirements

F20.1	The system shall provide a model for representative pressure and water level.
F20.2	The system shall provide control functions for the electric heater.
F20.3	The system shall provide control functions for the pressurizer spray.
F20.4	The system shall provide control functions for the power operated relief valves.

F20.5	The system shall provide control functions for the safety valves.

4.1.21. Subchannel Functional Requirements

F21.1	The system shall provide a subchannel model for single phase flow
F21.2	The system shall provide a subchannel model for two phase flow.

4.1.22. Models

F22.1	The system shall incorporate heat conduction model.
F22.2	The system shall incorporate single phase flow model.
F22.3	The system shall incorporate a well-posed seven-equation, two-pressure model for two-phase flow water and steam situations.
F22.4	The system shall incorporate a non-condensable gas model.
F22.5	The system shall incorporate the IAPWS-95 water-steam equation of state.
F22.6	The system shall incorporate the nitrogen equation of state.
F22.7	The system shall incorporate the Boron transport model
F22.8	The system shall incorporate the closure correlations specified in the Theory Manual
F22.9	The system shall come with AECL-IPPE 1995 critical heat flux lookup table
F22.10	The system shall allow for conjugate heat transfer between a pipe and a heat structure
F22.11	The system shall allow for variable body force vector (acceleration and gravity)

4.1.23. Natural Circulation Scenarios

F23.1	The system shall solve a single phase natural circulation problem
F23.2	The system shall solve a two phase natural circulation problem

4.1.24. Water Hammer Scenarios

F24.1	The system shall solve a single phase water hammer problem.
F24.2	The system shall solve a two phase water hammer problem.

4.1.25. LBLOCA Scenarios

F25.1	The system shall simulate the LOFT Experiment L2-5, which simulated a double-ended cold leg break.

4.1.26. SBLOCA Scenarios

F26.1	The system shall simulate Loss-of-Fluid Test (LOFT) Experiment L3-7, which simulated a 1-inch cold leg break.

4.1.27. Increase in Heat Removal Events Scenarios

F27.1	The system shall solve the PWR main steam line break benchmark problem specified in NEA/NSC/DOC(99)8.

4.1.28. Decrease in Heat Removal Events Scenarios

F28.1	The system shall simulate LOFT Turbine Trip Experiment L6-7/L9-2 (NUREG/CR-3257).

4.1.29. Reactivity and Power Distribution Anomaly Events Scenarios

F29.1	The system shall simulate LOFT Rapid Control Rod Withdrawal Experiment L6-8B2 (NUREG/CR-2930).

4.1.30. Boiling Water Reactor Instability Events

F30.1	The system shall simulate the Ringhals 1 Stability Benchmark problem specified in NEA/NSC/DOC(96)22.

4.1.31. Anticipated Transient Without Scram Events

F31.1	The system shall simulate LOFT Anticipated Transient Without Scram Experiment L9-3 (NUREG/CR-2717)

4.1.32. Station Blackout

F32.1	The system shall simulate the station blackout scenarios for a PWR such as Analyses of Natural Circulation During a Surry Station Blackout (NUREG/CR-5214, EGG-2547).
F32.2	The system shall simulate the station blackout scenarios for a BWR.

4.1.33. PWR Thermal Shock

F33.1	The system shall perform Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor (NUREG/CR-3761, EGG-2310).

4.1.34. Multiphysics MultiApp Coupling

F34.1	The system shall allow for using wall temperature from an external MOOSE-based application.
F34.2	The system shall allow for using power distribution from an external MOOSE-based application.
F34.3	The system shall allow for modifying the cross-sectional area of the flow channel.

4.2. Design Inputs, Outputs and Design Constraints

4.2.1. Input File Structure

DI1.1	The system shall use the input file syntax as defined by MOOSE.
DI1.2	The system shall employ the reactor component based input file.

4.2.2. Design Constraints

DC1.1	The system shall be implemented in MOOSE, INL's HPC development and runtime framework.
DC1.2	The system shall be written with object oriented programming language C++.

DC1.3	The system uses SI units.

4.2.3. Output File Structure

DO1.1	The system shall use the output capabilities of MOOSE framework.

4.2.4. Extensibility

E1.1	The system shall provide a way to supply proprietary data for fuel material properties.
E1.2	The system shall provide a way to supply proprietary closure correlations.
E1.3	The system shall allow the use of arbitrary fluid properties.

4.3. System Interfaces

SI1.1	MOOSE
SI1.2	libSBTL: Spline based table lookup library for water and steam properties as defined by IAPWS95 standard.

4.4. Installation Considerations

Detailed installation instructions, including the following installation considerations, will be included in the RELAP-7 User's Manual.

IC1.1	The user shall obtain a valid license
IC1.2	MOOSE Environment Setup: The user's system environment must be set up for MOOSE.
IC1.3	SSH Key Setup: The user's SSH key must be set up to allow the user to establish a secure connection between their computer and GitLab.
IC1.4	Code Checkout: It is necessary to build LibMesh before building any application. Once LibMesh has been successfully compiled, the system can be compiled.

IC1.5	Verification Testing: Once the system has been compiled successfully, it is recommended to run the tests to make sure the version of the code installed is running correctly.

5. REVIEW AND CONTROL OF THE SOFTWARE REQUIREMENTS

The system software requirements shall be controlled and verified as specified in PLN-4005. Once approved, the system requirements establish the functional baseline for the software.

6. DOCUMENT MAINTENANCE

The technical lead is responsible for maintaining this SRS. This SRS is controlled per LWP1201, “Document Management.” Revisions to this SRS will occur on an as needed basis as a result of reviews, audits, and requested changes. Modifications to this SRS must be reviewed and approved by the same organizational roles as the original release.

7. REFERENCES

The following are references for this SRS. All Idaho National Laboratory (INL) policies and procedures referenced are the current version at the time this SRS was approved.

- ASME NQA-1-2008 with the NQA1a2009 addenda, “Quality Assurance Requirements for Nuclear Facility Applications,” American Society of Mechanical Engineers, First Edition, August 31, 2009.
- ISO/IEC/IEEE 24765:2010(E), “Systems and software engineering Vocabulary,” First Edition, December 15, 2010.
- LWP1201, “Document Management.”
- LWP-13620, Managing Information Technology Assets, Revision 17, 7/28/2014.
- PLN-4005, SQAP for MOOSE and MOOSE-Based Applications, Revision 3, 3/26/2016.
- MOOSE-SRS Revision 0, Draft
- INL/EXT-14-31366, RELAP-7 Theory Manual, Revision 2, March 2016
- RELAP5 Assessment: - LOFT Turbine Trip L6-7 /L9-2, NUREG/CR-3257, July 1983.
- Experiment data report for LOFT anticipated transient-without-scram Experiment L9-3, NUREG/CR-2717; EGG-2195.

- Analyses of natural circulation during a Surry station blackout using SCDAP/RELAP5, NUREG/CR-5214; EGG-2547.
- RELAP5 Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor, NUREG/CR-3761, EGG-2310.
- Experiment data report for LOFT anticipated transient experiment series L6-8, NUREG/CR-2930.
- Ringhals 1 Stability Benchmark, NEA/NSC/DOC(96)22.
- PRESSURISED WATER REACTOR MAIN STEAM LINE BREAK (MSLB) BENCHMARK, NEA/NSC/DOC(99)8.

8. DEFINITIONS AND ACRONYMS

This section defines, or provides the definition of, all terms and acronyms required to properly understand this SRS.

8.1. Definitions

Component: A component in the system is a simplified representation of the functions of the physical component in a NPP.

8.2. Acronyms

0-D	Zero Dimensional
1-D	One Dimensional
2-D	Two Dimensional
AC	Alternate Current
AECL	Atomic Energy Canada Limited
AMSAC	ATWS mitigation circuit
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram

BWR	Boiling Water Reactor
CASL	Consortium for Advanced Simulation of Light Water Reactors
CHF	Critical Heat Flux
CHT	Conjugate Heat Transfer
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
GUI	Graphical User Interface
HPC	High Performance Computing
IAPWS	The International Association for the Properties of Water and Steam
INL	Idaho National Laboratory
IPPE	Institute of Physics and Power Engineering
IT	Information Technology
LBLOCA	Large Break Loss of Coolant Accidents
LOFT	Loss Of Fluid Test Facility
LPSI	Low Pressure Safety Injection
LWR	Light Water Reactor
MIT	Massachusetts Institute of Technology
MOOSE	Multiphysics Object Oriented Simulation Environment
MSIV	Main Steamline Isolation Valve
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NPP	Nuclear Power Plant
NQA	Nuclear Quality Assurance
ODE	Ordinary Differential Equation

OOP	Object Oriented Programming
PDE	Partial Differential Equation
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program
RISMC	Risk Informed Safety Margin Characterization
SBLOCA	Small Break Loss of Coolant Accidents
SBO	Station Blackout
SI	The International System of Metric Units
SMR	Small Modular Reactor
SRS	Software Requirements Specification