

Enhanced Severe Transient Analysis for Prevention Technical Program Plan

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SUMMARY

This document outlines the development of a high fidelity, best estimate nuclear power plant severe transient simulation capability that will complement or enhance the integral system codes historically used for licensing and analysis of severe accidents.

As with other tools in the Risk Informed Safety Margin Characterization (RISMC) Toolkit, the ultimate user of Enhanced Severe Transient Analysis and Prevention (ESTAP) capability is the plant decision-maker; the deliverable to that customer is a modern, simulation-based safety analysis capability, applicable to a much broader class of safety issues than is traditional Light Water Reactor (LWR) licensing analysis. Currently, the RISMC pathway's major emphasis is placed on developing RELAP-7, a next-generation safety analysis code, and on showing how to use RELAP-7 to analyze margin from a modern point of view: that is, by characterizing margin in terms of the probabilistic spectra of the "loads" applied to systems, structures, and components (SSCs), and the "capacity" of those SSCs to resist those loads without failing. The first objective of the ESTAP task, and the focus of one task of this effort, is to augment RELAP-7 analyses with user-selected multi-dimensional, multi-phase models of specific plant components to simulate complex phenomena that may lead to, or exacerbate, severe transients and core damage. Such phenomena include: coolant crossflow between PWR assemblies during a severe reactivity transient, stratified single or two-phase coolant flow in primary coolant piping, inhomogeneous mixing of emergency coolant water or boric acid with hot primary coolant, and water hammer. These are well-documented phenomena associated with plant transients but that are generally not captured in system codes. They are, however, generally limited to specific components, structures, and operating conditions. The second ESTAP task is to similarly augment a severe (post-core damage) accident integral analyses code with high fidelity simulations that would allow investigation of multi-dimensional, multi-phase containment phenomena that are only treated approximately in established codes.

In both plant system and severe accident analyses, a wide variety of phenomena occur in a large number of SSCs. Computer limitations, complex geometries and physics that challenge modeling efforts still require that the bulk of the phenomena be modeled in an integral sense. The models are generally mechanistically based but quantities of interest are modeled as lumped parameters representing averages over zero-dimensional 'control volumes' representing major plant structures. Spatial resolution is discarded except for a few instances in which either the structure is divided into multiple control volumes or a true 1-D model is constructed in which the structure is sub-divided geometrically into nodes or cells and a discrete approximation to a governing equation is solved over each node. The latter approach yields information about the variability of the parameter within the structure but is correspondingly much more computationally intensive.

Fast-running integral codes lend themselves to parametric and perturbation studies; enabling the quantification of sensitivities and uncertainties for risk assessment and bounding analysis. The basic assumption of integral codes, however, is that the solution to the governing equations using these lumped parameters (and associated 'subgrid' correlations) adequately capture the physics of system. This is a reasonable assumption in many cases and indeed models have been validated for a number of plant transients and accident sequences using integral codes such as RELAP5 and MELCOR. There are, however, phenomena that occur during severe transients that exhibit multi-dimensional, multi-scale, or multi-phase behaviors that are not captured with average parameters. The extent to which these phenomena affect the trajectory of a severe accident sequence is not known as plant instrumentation is rarely configured to detect it. High fidelity, finite element codes can simulate the phenomena in individual structures as long as the rest of the plant is ignored or modeled as simple boundary conditions. At this time, it is simply not feasible to consider modeling the entire plant with this level of fidelity.

In this effort, the focus is on developing a generic approach to coupling fast-running integral codes to high fidelity models of specific SSCs with the goal of producing a tool that can be used to study the effect of complex phenomena in a given structure while retaining the full dynamic feedback from the balance

of plant. The burden of deciding which structure to model with high fidelity remains with the user but the normally difficult process of coupling a high fidelity simulation with a system code should be automated to a far greater extent than has been the case thus far.

Coupling codes in this manner is not trivial and poses a significant technical challenge to success. In some ways, it is more difficult than writing either a standalone integral code or a standalone high fidelity physics code. The full capabilities of the Multiphysics Object Oriented Simulation Environment (MOOSE), under development at the INL, will need to be harnessed to make the coupling efficient and transparent to the user. In the first part of this effort (SubTask 1), the work will be facilitated by the fact that the system code in question, RELAP-7, and the specific high fidelity physics solvers (PRONGHORN and BIGHORN), are being built on the MOOSE platform. The PRONGHORN/BIGHORN teams will be able to work closely with the MOOSE and RELAP-7 teams to develop a coupling approach that is applicable to a wide variety of severe transient scenarios. The lessons learned will be applied directly to the second Subtask in which a high fidelity containment thermal fluid solver will be coupled to an existing (and non-INL) integral code such as GOTHIC, CONTAIN, or MAAP. This second effort shall require close collaboration with the (external) developers of the code.

Given the desired product and the limited resources available, development of the ESTAP tool will rely heavily upon the concurrent development of associated MOOSE-based tools, RELAP-7, PRONGHORN, BIGHORN, MAMMOTH, and RAVEN. With the exception of PRONGHORN and BIGHORN which will be largely covered in the SAAP Work Package, these codes are being developed under other project funding along a schedule that coincides with that of the SAAP effort. Focusing on Subtask 1 (RELAP-7 augmentation) will allow time to develop working relationships with the developers of the existing integral containment analysis code to get it running on the MOOSE platform.

Subtask 1 is the development of an enhanced integral code for severe transient investigations and prevention. We will create a best estimate tool that will enable the investigation of complex phenomena that may occur during a severe plant transient that may lead to core damage. The scenario of interest is a main steam line break (Long Term Coping) in a PWR. The general coupling capability, however, is applicable to the modeling of a severe transients in advanced reactors as well.

Subtask 2 is the development of an enhanced integral code for containment analysis. The scenario of interest is BWR (Mark I or II) containment pressurization and venting after a loss of coolant, core damage, and depressurization of the primary loop (Containment Venting). The phenomenon of interest is the stratification of gases and aerosols in the containment.

The overall goal is not to replace integral codes for accident analysis but to efficiently augment those codes with high fidelity models as needed to investigate specific phenomena that may defy analysis with current tools. The project is expected to take five years if funded as planned.

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Enhanced Severe Accident Analysis for Prevention Technical Development Plan

1. INTRODUCTION

1.1 Purpose

The Light Water Reactor Sustainability Program (LWRS) RISMIC Pathway [1] focuses on modernization of nuclear power safety analysis tools (including methods and data); implementing state-of-the-art modeling techniques; taking advantage of modern computing hardware; and combining probabilistic and mechanistic analyses to enable a risk informed safety analysis process. The *modernized* tools will maintain the current high level of safety in our nuclear power plant fleet, while providing an improved understanding of safety margins and the critical parameters that affect them. Thus, the set of tools will provide information to inform decisions on plant modifications, refurbishments, and surveillance programs, while improving economics. The set of tools will also benefit the design of new reactors, enhancing safety per unit cost of a nuclear plant.

The Enhanced Severe Accident Analysis for Prevention (ESTAP) task is focused upon the development of high fidelity, best-estimate analysis of complex phenomena that occur during severe nuclear power plant transients. Current accident analysis tools such as RELAP5[2], MELCOR[3], ASTEC[4], GOthic[5], and MAAP[6] possess many of the physics describing the wide range of phenomena occurring during a severe transient but they do so in an ‘integral’ sense. Core and plant structures are modeled by large, zero-dimensional ‘control volumes’ characterized by a set of ‘lumped parameters’, which are often averages of spatially dependent parameters such as solid material temperature. In some cases, however, that spatial dependence may influence the progression of an accident sequence and more detailed understanding of the phenomena may lead to design changes or operator actions that can prevent core damage.

It is neither practical nor necessary to solve all of the physics in a severe accident code like MAAP using higher order finite element or other spatially resolved methods. Instead, the user shall have the capability to apply such methods to specific structures and phenomena of interest while using the fast-running, integral solvers for the balance of the simulation. This approach enables the investigation of multi-dimensional, multi-scale, multi-phase, and/or multi-physics phenomena in specific components while retaining the dynamic feedback of the system simulation. Such a modeling approach requires a flexible and computationally efficient computing platform that seamlessly converts between the lumped average boundary values used in the integral code and the spatially dependent boundary values of high fidelity solution. It must do this in a physically accurate way (conserving bulk quantities) while damping out any numerical oscillations and instabilities that often accompany such coupling. Such a capability is afforded by the MultiApps and Transfers[7] protocol in the Idaho National Laboratory’s Multiphysics Object Oriented Simulation Environment (MOOSE)[8].

A complete accident sequence that results in core damage can effectively be represented by two sets of physics. Prior to core damage the geometry is largely fixed and plant behavior is governed by conjugate heat transfer (thermal fluids) in the core and primary coolant loop. Radiation transport and kinetics may play a role if the core remains critical, i.e. during an Anticipated Transient Without Scram (ATWS) event, and fluid/aerosol dynamics and oxidation may be involved if the primary loop is depressurized, intentionally or not. Core integrity, however, is mainly a function of heat and coolant mass transfer which may be quite complex depending on the extent of boiling and buoyancy driven (natural circulation) flow. If the sequence results in significant core damage, neutron kinetics has a very limited role but other physics become important including: circulation of steam and non-condensable gases, oxidation of core materials (especially Zircaloy cladding), loss of core geometry (clad ballooning and rupture, melting of fuel, cladding, and other structures, relocation, embrittlement and fragmentation of

corium), heating of the lower head, and release of fission products from the fuel matrix and transport to the coolant. If the primary coolant boundary is breached, ex-vessel phenomena such as containment thermal-hydraulics, hydrogen accumulation, aerosol behavior, and corium spreading and interactions with concrete and other materials must be modeled.

For this reason, the task will be conducted in two parts. Subtask 1 will focus on simulating the plant while the fuel is still intact and actions can be taken to maintain a coolable geometry (Long Term Coping). The objective will be to augment the RELAP-7 code^[9] with high fidelity simulations of selected structures and phenomena. As RELAP-7 is being developed on the MOOSE platform, legacy code coupling issues will be avoided and resources can be devoted to constructing and testing a three-dimensional core thermal fluids solver (PRONGHORN) to replace the 1-D core solver in RELAP-7. The target simulation will be a PWR Main Steam Line Break that has been the subject of an international code-to-code benchmark. The reference case will be modeled in the near term using RELAP5-3D.

Another test of the capability will be to model fluid stratification in the cold leg of the AP1000 after activation of the Passive Residual Heat Removal (PRHS) system, an emergency core cooling system that is unique to this design. Cold water from the PRHS flows into the bottom plenum of the steam generator then out through the cold leg to the core via natural circulation. If mixing of this cold water and the heated water in the steam generator is incomplete, cold and hot water streams may cause excessive thermal stress on the flange and lower structures in the pressure vessel. The task will be to augment RELAP-7 with multi-phase CFD solver (BIGHORN) to be able to model this stratification and other natural circulation phenomena in advanced reactors. Ultimately a thermal stress analysis solver would be coupled to this tool. Again, the near term reference case will be modeled, to the extent that it can, with RELAP5-3D.

Substantial progress in Subtask 1 can facilitate Subtask 2 in two ways: 1) the initial state of the reactor (a BWR mark I or Mark II plant) will be captured more accurately with the improved simulation tool, and 2) many of the coupling and time-stepping issues will likely be resolved in the coupling to RELAP-7. The subtasks may, however, be largely independent and can be completed concurrently or in opposite order.

Subtask 2 will shift the focus to events occurring after the core is severely damaged, specifically the pressurization and venting of the containment (Filtered Venting). In this scenario, a 3-D model of the gas-aerosol dynamics of the containment will be used to augment a typical integral code simulation (GOTHIC) to investigate the effects (if any) of temperature distribution and fluid stratification on containment pressurization and venting scenarios. As these are stand-alone codes not maintained by the INL, close collaboration with the code developers will be necessary to resolve coupling and numeric issues. The near term reference case will be a simulation of the event with the unaugmented code.

1.2 Background

DOE-NE is engaged in a comprehensive modernization of analytic codes and methods for nuclear energy applications. The Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, administered by the office of Nuclear Energy, is charged with the application of modern computational methods to better simulate the performance of nuclear power plants during both normal operations and anticipated transients. The recently developed NEAMS Toolkit consists of a multi-physics, multi-scale computational framework which couples various physics modules aimed at predictive modeling of physical phenomena at various scales. Other DOE-NE programs such as the Advanced Reactor Technologies, Fuel Cycle R&D, the Consortium for the Advanced Simulation of Light Water Reactors (CASL), and the Light Water Reactor Sustainability (LWRS) Program, to name a few, are also developing modern computer codes and methods for nuclear energy applications.

The RISMC Pathway focuses on implementation of a comprehensive, risk-informed safety framework that characterizes and quantifies an enhanced safety construct by doing the following[1]:

- Addressing the full range of hazard types
- Better managing human performance
- Employing advanced analysis tools
- Incorporating uncertainty to better manage margin against the unknown.

The aim of RISMC is to develop method, tools, and data to quantify safety margins, accurately analyze accident progression, and enable designs by implementing an advanced risk-informed management approach that will maintain adequate safety (i.e., balancing cost with safety, as illustrated in Figure 1) in nuclear-related facilities.

The ESTAP task focuses specifically on severe transients (e.g. station blackouts) that may result in core damage. The additional types of phenomena that occur after the fuel is severely damaged are significantly different than those occurring prior to core melt so this task is divided into two subtasks.

Enhanced Long-term Severe Transient Analysis - System-level or integral codes are widely used for transient analysis and licensing because a large number of phenomena can be modeled (approximately) and because the models run quickly on single CPU computers or small workstations. For these reasons models can be constructed to perform bounding and sensitivity analyses. It is desired to retain this attribute in the augmented analyses. The idea is to use the system code as the basic engine for the simulation but replace selected 0-D and 1-D structures with 3-D structures solved using an appropriate finite element solution algorithm.

The enhanced capabilities are a direct and logical extension of the work already underway to build the next generation reactor system code, RELAP-7. RELAP-7 is being built on the MOOSE platform and currently possesses the capability to execute simple plant models. A one-dimensional, two-phase CFD solver has been added to simulate boiling in a BWR channel. The code is being designed to couple with higher fidelity simulations of the core and other components.

With the enhanced RELAP-7, many of the complex phenomena associated with severe transients in LWRs can be investigated with greater fidelity than that which can be achieved with the nominal system code. Designers and analysts will be able to simulate the effects of asymmetric coolant flow in the core during a rod ejection or other non-uniform reactivity insertion, stratified flow in components, natural circulation, and inhomogeneous mixing of multiple flow streams. Such a capability will aid designers and operators in developing mitigating systems and procedures for coping with a loss of AC power or other severe accident scenarios.

Even with modern programming, however, reactor system codes provide information about the gross behavior of the plant. In most simulation, spatial resolution and fidelity are sacrificed to obtain computationally efficiency. For conservative safety analyses and other licensing calculations this is acceptable as the ‘worst case’ scenario is meant to reveal only phenomenological ‘cliff edges’ rather than accurate information on the local level.

For ‘Best-Estimate plus Uncertainty’ analyses, higher-order phenomena and the space dependence of many variables can influence the actual trajectory of a transient. For example, thermal stratification in pipes may result in stresses that can hasten the onset of fatigue or failure due to thermal shock. A better and more detailed understanding of temperature profiles within the core or other structures can help designers and operators to take preventative or remedial measures that will lower the probability of more severe consequences.

Containment Pressurization and Venting – Two of the safety improvements mandated by NRC following the accident at the Fukushima Daiichi nuclear facility are to:

- Install reliable, hardened containment venting systems capable of operating under beyond-design-basis and severe accident conditions

- Install containment vent filtration systems to reduce the release of radioactive materials should a severe accident occur for BWR Mark I and Mark II containments [10]

Given the relatively small volumes of Mark I and II containments that depend on suppression pools and have no mitigation for hydrogen, ensuring the availability of reliable, hardened containment vents will provide plant operators with improved ways to vent containments during a wide range of beyond-design-basis events. Venting containment can help prevent or delay the loss of, or facilitate recovery of, important safety functions such as reactor core cooling, reactor coolant inventory control, containment cooling, and containment pressure control. Some analyses indicate that venting may be ineffective, or worse, deleterious to plant safety. It has been postulated that changes to venting in BWRs could have negative aspects, for example “venting has been postulated to increase the likelihood of core damage by causing pump cavitation and the eventual loss of injection to the reactor coolant system.”[10] With regard to the Fukushima accident, deliberate venting of the containment of Units 1 and 3 are now believed to have caused the hydrogen explosions as the ventilation piping was connected to the normal plant building exhaust stack. Rather than escaping to the atmosphere, hydrogen was diverted back into the building where it was able to mix with oxygen and ignite. Therefore, accident sequences and then interactions of engineered systems in those sequences needs to be better understood to determine under what conditions the containment vents are beneficial or non-beneficial.

For example, in most U.S. nuclear power plants, the net positive suction head for Emergency Core Cooling System (ECCS) pumps in licensing-basis analyses is calculated assuming that the pressure in containment is atmospheric. In reality, accidents such as a LOCA would lead to an increase in the containment pressure. The assumption of atmospheric pressure in the containment assures that in design-basis accidents, the loss of the capability of the containment to maintain pressure would not affect the ability of the ECCS to maintain core cooling. The inclusion of the pressure developed in the containment during an accident in the calculation of the available net positive suction head is referred to as containment accident (or overpressure) pressure credit. The Advisory Committee on Reactor Safeguards (ACRS) has consistently expressed concerns with the use of this margin because it represents a decrease in the safety margin available to deal with a phenomenon that is subject to large uncertainties, namely, maintenance of adequate net positive suction head for ECCS pumps during accident. This margin protects against unanticipated accident phenomena such as sump strainer blockage or an inadvertent loss of containment isolation. In some requests for containment accident pressure request, operator actions are required to establish or maintain elevated containment pressure. Of particular concerns are actions that stop or reduce operation of systems whose normal design function is to remove heat from the reactor core or containment.

The issue of evaluating containment venting with updated methods, tools, and data is also timely. As suggested in the NRC proposed order the completion date for implementation of filtered hardened vents should be no later than December 2017. Because of the uncertainties associated with containment venting, the NRC has delayed the decision on the final rule until March 2017.

For nominal safety analyses, pressurization of the containment upon venting of the primary coolant loop has been simulated by control volume (0-D) codes like MELCOR or GOTHIC. Fluids flow between large volumes representing building spaces with the flow rates governed by simple mass and energy conservation with momentum terms given by a lumped parameter equations describing the fluids as ‘slugs’ with characteristic lengths. Within the cells, however, the fluids are assumed to be stagnant and well-mixed (homogeneous). While such assumptions may yield the correct gross behavior in many scenarios, the mixing of hydrogen, aerosols, and other noncondensable gases in the containment building spaces is incomplete for substantial time periods. As with the pre-core-damage scenarios described previously, stratification of fluids in the containment can be expected, leading to a different accident progression. Again, improved understanding of these sequences can be achieved by coupling multi-phase, multi-dimensional simulations of specific containment structures to the low-order simulation of the remaining physics offered by the integral code.

2. VALUE PROPOSITION

The ability to simulate a specific scenario such as a station blackout is useful but the key attribute acquired in this effort will be the flexibility to swap in a specific high fidelity simulation of a component while retaining the full dynamic feedback of the balance of plant. This is important in severe accident analysis which is often characterized by a large number of coupled physics amenable to low order modeling but with few multi-dimensional, multi-component phenomena that are treated only approximately.

A high-fidelity, containment system analysis methods and tools development will provide more science-based understanding of the accident sequences. The development will also help the optimization of the venting system designs and operations and support understanding for several questions:

- How well will the hardened vents perform in scenarios with rapid temperature and pressure increase?
- Should the vents be designed to have a rupture disk such that the vent would work passively or should the vents be manually operated?
- When the venting should happen, is early venting preferred when the containment pressure and hydrogen concentration are low or should venting wait for containment pressure to reach or exceed design pressure?
- What should the reliability targets be for operation in order to have “reliable” hardened vents?
- What will the economics look like once a design support tool is available to answer the questions above and provide a technical basis for tradeoffs in design space?

The various Mark I and II containments have similarities, but also differences in design features and system capabilities which may result in differences in costs for severe accident-capable vents between the two types. A higher cost would likely reflect need to modify containments to prevent a molten core from causing a bypass of the suppression pool because of failure of downcomers and drain lines below the reactor vessel. Such differences have given rise to the possible benefits of developing a performance-based approach, which would require each plant owner/operator to evaluate the needed performance of the containment venting function and to implement appropriate design and procedure changes to satisfy the performance requirements.

3. END USE OF ESTAP PRODUCTS

The overall RISMC goal is to provide cost-beneficial approaches to safety by leveraging modern methods, augmented (a combination of existing and new) tools, and repurposed (existing, but used in a new way) data. The ESTAP task focuses on the second of these; augmenting existing or new severe transient analysis tools with high fidelity solutions of specific components or phenomena. The new capability will be initially applied to three ‘Use-Cases’, scenarios representing severe transients with or without depressurization of the primary coolant loop.

Use-cases describe the analysis approach (methods, tools, and data) from the user’s point of view, meaning we describe the user-specific analysis characteristics to be performed and the intended decisions to be supported. This use-case concept is particularly suited for modern methods, tools, and data development since any one safety use-case is supported by a number of technical solutions. However, no use-case requires all of the proposed advances in safety and one advance (say, for example, enhanced 3D modeling) may be used effectively in many different use-cases. In a sense, use-cases are requirements in themselves, helping to effectively communicate the RISMC objectives and products to prospective users and stakeholders and serves as a benchmarks to document and measure success. The ‘Use-Case’ concept is described in more detail in [11] along with the three stakeholder groups that stand to benefit mostly from RISMC tools.

The “Long Term Coping Strategies” use-case is the focus of Subtask 1. In these scenarios, the plant is subjected to a severe transient, perhaps resulting from a simultaneous loss of onsite and offsite AC power, and is relying on DC-powered actuators and stream-driven systems to maintain core cooling. The core remains covered. The enhanced safety analysis tools developed in this task will allow designers and analysts to investigate plant behavior in greater detail with an eye toward implementing systems and procedures that will maximize the time to core damage, in accordance with the industry-led FLEX program[12].

The coupling approach is also applicable to “Advanced Reactors”, a second use-case. New reactor designs such as the AP1000 and Nuscale are ostensibly designed to withstand such a loss of AC power by relying on passive decay heat removal and natural circulation. In both plants, there is the possibility of stratified flow conditions that can complicate heat removal but that are not captured in integral safety analysis codes.

Note: A Station Blackout (SBO) in a Boiling Water Reactor (BWR) has been the subject of RELAP-7 testing and demonstration. BWR fuel assemblies are encased in Zircaloy canisters which prevent inter-assembly coolant flow. A PRONGHORN 3-D analysis of core coolant flow is not expected to yield additional information. The two-phase flow solver (BIGHORN) being developed for RELAP-7, however, will be exploited as part of the ESTAP toolkit for multi-phase flow phenomena.

“Reactor Containment” is the use-case that is the subject of Subtask 2. In the aftermath of Fukushima, the NRC will require the installation of filtered containment venting systems at all General Electric Mark I and perhaps Mark II Boiling Water Reactors (BWRs). Deterministic containment analysis codes need to be used within the RISMIC Toolkit framework to assess the risk of taking credit of containment accident pressure. Current Probabilistic Risk Assessments (PRAs) can estimate the likelihood of pre-existing containment leakage. However, they cannot evaluate the risk of small amount of leaks and other evolving conditions that may reduce the available net positive suction head. Therefore, analyses of the impact of loss of containment integrity are needed and can be used to evaluate the likelihood of scenarios in which large amounts of containment overpressure credit are required for a significant amount of time. The previous methods are based on the licensing-basis analyses. The judgment of whether to grant containment overpressure credit for a particular application would depend on more realistic and deterministic evaluation of the required amount and duration of containment overpressure credit, the likelihood of scenarios that would require containment overpressure credit, and the operator actions required to maintain containment overpressure from adequate pump net positive suction head. The deterministic evaluation would include uncertainty quantification. The PRONGHORN code being developed with the fully coupled simulation capability for all components involved will provide more realistic simulations containment overpressure.

4. Demonstration Cases and Development of Tools

The systems codes and the higher order modules will be coupled using MOOSE. MOOSE provides efficient algorithms for solving large and stiff sets of partial differential equations, meshing utilities, and utilities for coupling different codes. In particular, a new MOOSE framework development is enabling the efficient combination of multiple, independently developed applications with the goal of achieving massive, multi-scale calculations. This development, which combines a flexible execution strategy with a sophisticated data exchange protocol, allows MOOSE-based applications to run concurrently while exchanging data, a process termed multicoupling. Multicoupling is a unique technology that will directly address the challenges of coupling phenomena spread across length and time scales and physics and thus will greatly facilitate the development of enhanced systems codes in the project.

The performance of these systems will need to be demonstrated. The Fukushima Daiichi Unit 1 (or 3) containment will be modeled by running the severe accident integral code on the MOOSE platform and coupling to it a higher order model of the containment and ventilation piping. As a reference case, a nominal integral model of this reactor accident will be prepared and executed.

4.1 Subtask 1 – Enhanced RELAP-7 Analysis for Pre-Core Damage Simulation of Severe Transients

4.1.1 Demonstration Cases

Severe transients in PWRs - The Three Mile Island Unit 1 Main Steam Line Break (MSLB) Benchmark [13] calls for the analysis to generate a number of integral parameters such as maximum power, flow rates, hot and cold leg temperatures, and core average coolant temperatures. These can be, and traditionally are, generated using a code like RELAP5 which models the core as a small number of control volumes with one set representing an average coolant channel and others combining to represent other channels of interest such as the expected hot channel. Coolant passes from one control volume to the next in a coarse representation of axial flow. Another control volume is used to represent coolant bypassing the assemblies. An example of a system code model of the TMI-1 reactor is shown in Figure 1.

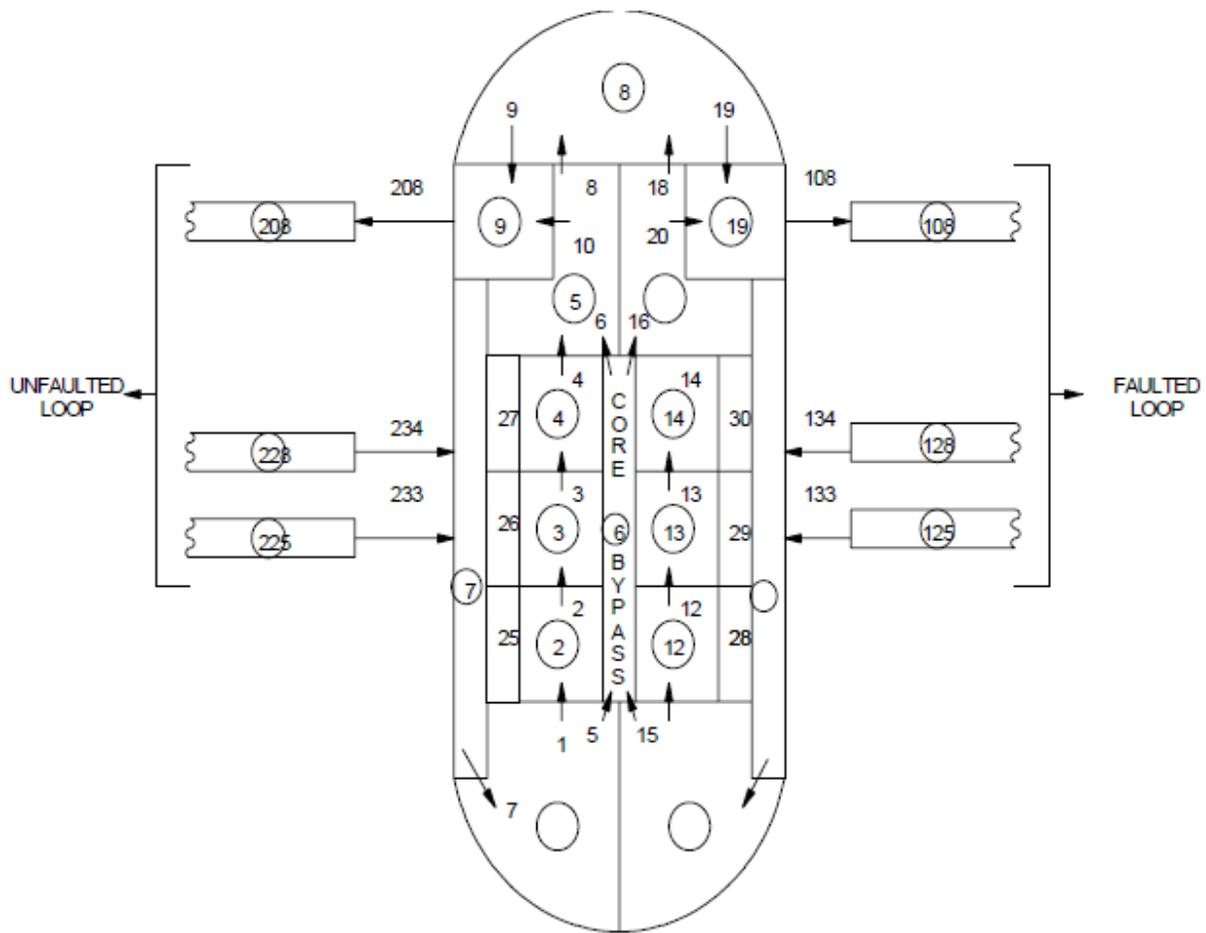


Figure 1: Nodalization of the TMI-1 Core using RETRAN [13]

The balance of plant is modeled with a similar set of control volumes representing pipes, valves, the steam generator, pumps, etc., and example of which is shown in Figure 2.

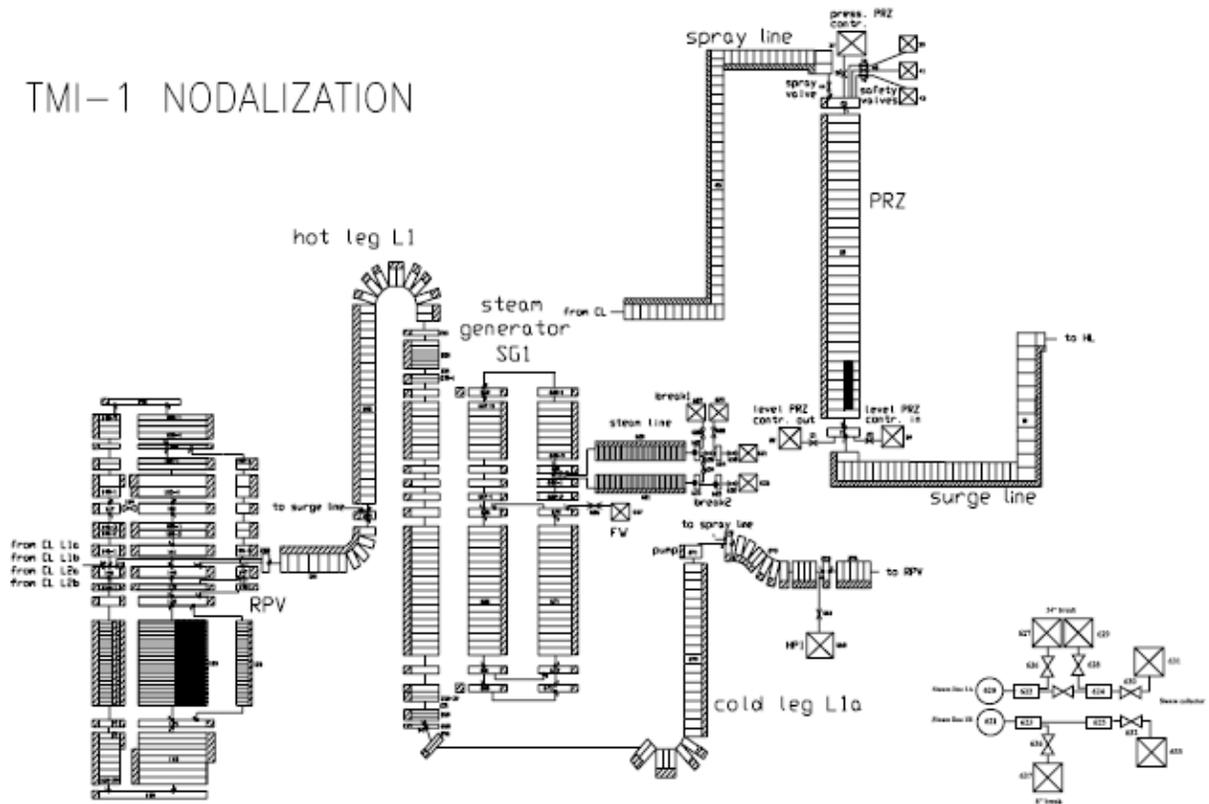


Figure 2: Nodalization of the TMI-1 MSLB benchmark [13]

Multi-dimensional, two-phase core thermal fluid codes, if they existed, were generally not applied even though full 3-D kinetic codes were used for the neutronics. For this particular problem, the 3-D core thermal fluid code PRONGHORN will be substituted for the 1-D core module in RELAP-7, enabling the simulation of asymmetric thermal fluid behavior (cross flow). PRONGHORN[14] development began under the Next Generation Nuclear Plant (NGNP) project to yield 3-D single-phase temperature profiles in a high temperature gas-cooled reactor during severe loss of forced cooling events. PRONGHORN solves the fluid equations for homogenized structures (e.g. assemblies) in the core and thus represents a balance between low-order system and high fidelity CFD models. This approach sacrifices local (pin-level) resolution to achieve greater computational speed, enabling core-wide multi-dimensional fluid modeling on a high performance cluster. PRONGHORN is now being modified to also simulate coolant flow in LWRs. A finite element model of a PWR core recently generated for PRONGHORN testing is shown in Figure 3.

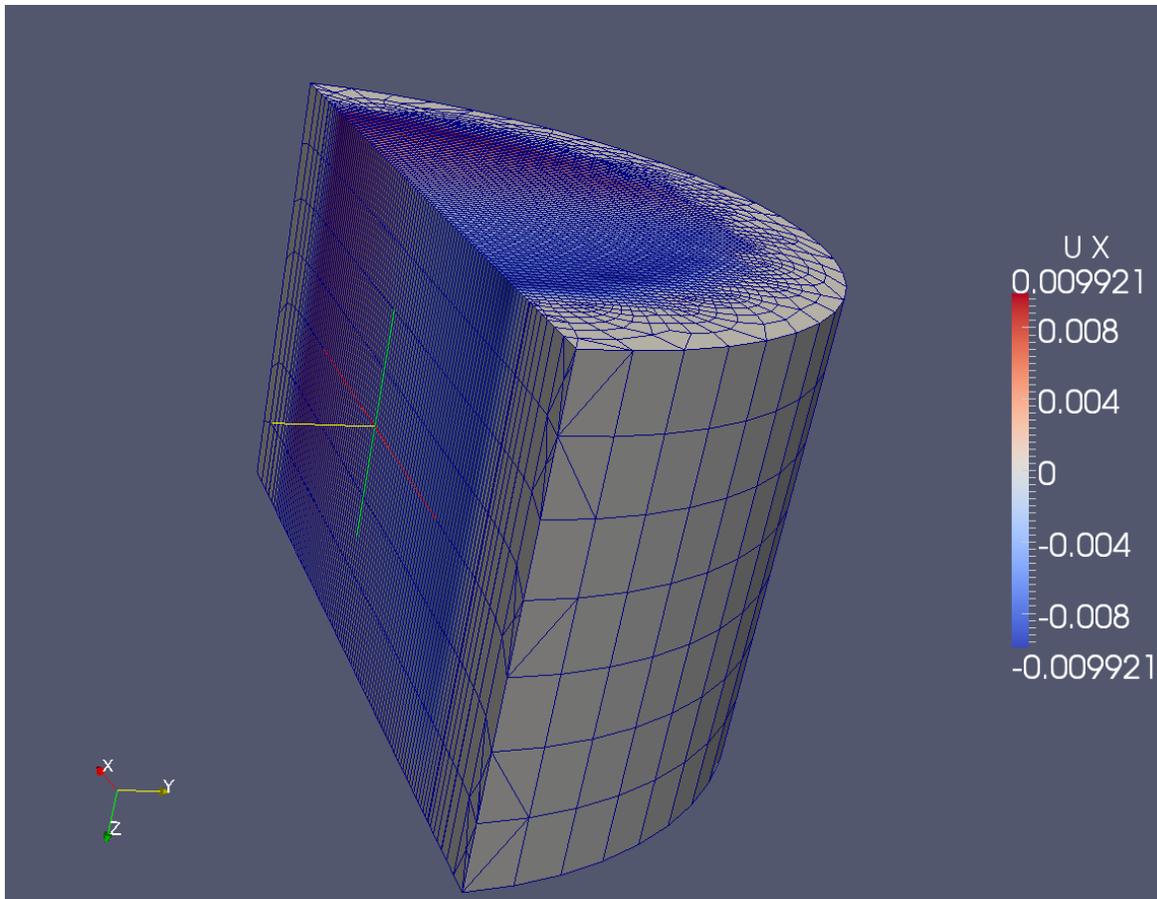


Figure 3: Finite Element Model of a PWR Core used for PRONGHORN Analysis

The geometric detail of the individual fuel assemblies is replaced by cells the thermal fluid properties of which are obtained through homogenization techniques which retain the overall fluid and heat transfer properties of an assembly. The bulk core fluid behavior can be investigated without the prohibitive computational overhead of a comparable CFD model.

The balance of plant will be modeled with RELAP-7 using the basic control volume approach to allow analysts to study the behavior in the core subjected to various actions and failures of the emergency core cooling system. For testing the PRONGHORN solver, a RELAP5-3D model of a 4-Loop Westinghouse PWR (core only) will be used. As the near-term reference case, a plant model of the TMI MSLB will be prepared and executed.

Severe Transients in BWRs - The Fukushima accident understandably focused attention on the survivability of older BWR designs, in particular those with GE Mark I and Mark II containments. Simulation of Station Blackout in this class of reactor is a primary goal of the RELAP-7 development program. As BWR fuel assemblies are contained within closed fuel channels, coolant cannot flow between assemblies as it can in PWRs. A PRONGHORN simulation of the BWR would not reveal significantly new information over what will be achieved with the 1-D RELAP-7 model of a channel. Effort in this task will therefore be focused on the PWR.

Natural Circulation in Advanced Reactors – New LWR designs intentionally rely on natural (buoyancy-driven) circulation of coolant flow for decay heat removal and, in the case of Nuscale, fission heat removal. Natural circulation has been used to enhance decay heat removal, and in rare cases, fission heat removal in existing plants but never to the extent proposed for plants like the AP1000 and Nuscale.

The behavior of these plants under severe transient conditions is still under investigation. In particular, inhomogeneous flows are not reliably captured, if at all, with system analysis codes which generally assume homogeneous mixing of flow streams. When activated, the Passive Residual Heat Removal System (PHRS) in the AP1000 injects cold water into the bottom plenum of the steam generator (Figure 4). A system code would assume uniform mixing of this cold water and the hot water in the steam generator. Some experiments suggest, however, that the cold water will flow unmixed through the primary circulator pumps into the cold leg of the reactor vessel, leading to thermal shock in pressure vessel and fuel structures near the inlet. This phenomenon must be modeled with 2- or 3-D computational fluid dynamics in the cold leg.

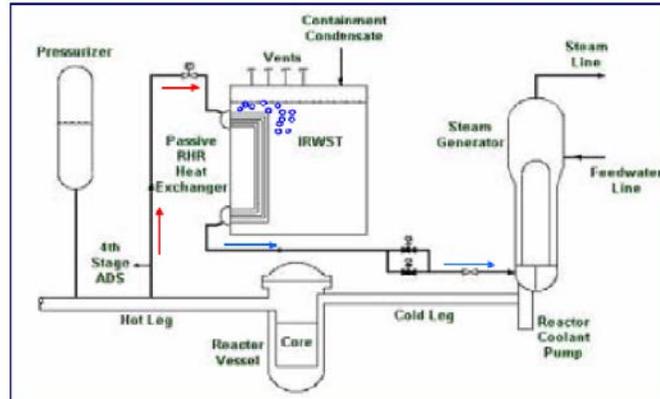


Figure 4: Schematic of the AP1000 with Activated Passive Residual Heat Removal System

For this project, the cold leg control volume in RELAP-7 will be replaced with a higher order, multiphase, multidimensional pipe model using the well-posed, 7-equation CFD formulation being developed at the INL[15]. A 1-D version of this equation set is under development for RELAP-7. Development of the multi-dimensional version (BIGHORN) will be pursued in Phase I of the ESTAP Project to specifically study stratified flow in the cold leg of the AP1000. As a near-term reference case, a RELAP5-3D model of the AP1000 will be prepared and executed.

Additional applications

Inhomogeneous mixing of coolant in the Nuscale reactor can also be modeled with an enhanced RELAP-7 code. The Chemical Volume Control System injects boric acid into the coolant near the core outlet. It then passes through the steam generator above the core, and back to the core through a downcomer. Incomplete mixing of the boric acid will lead to slugs of high concentration boric acid circulating through the primary with corresponding reactivity insertions. This phenomenon was observed during startup testing in the San Onofre Nuclear Generating Station [16]. Again, systems codes cannot capture this scenario but the multiphase formulation of the PRONGHORN coarse-grained CFD equation set will enable it.

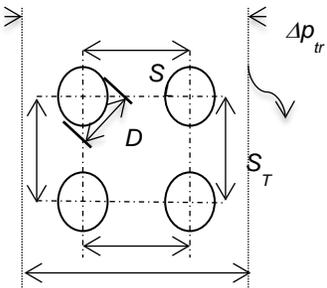
4.1.2 Development of Tools

Three-dimensional, single-phase flow in PWRs - The PRONGHORN equation set originally included time-dependent neutron diffusion, 3-D heat conduction, and gas dynamics based upon a simple Darcy flow model which neglected momentum terms in the fluid flow. This is a reasonable simplification for modeling gas-cooled reactors but is inadequate for liquid-cooled systems. The neutronics equations have since been omitted as a more capable neutron transport solver, Rattlesnake [17], has been constructed on the MOOSE system and coupled to RELAP-7. A multi-dimensional version shall be used with PRONGHORN to yield a complete 3-D model of the core.

The conjugate heat transfer equation set in PRONGHORN is being modified to include inertia and viscosity terms that enable a correction specification of the solid-wall (no-slip) boundary condition. The so-called Hazen-Dupuit-Darcy model, the momentum equation for which is given by

$$\frac{\partial \epsilon \rho \bar{u}}{\partial t} + \nabla \cdot (\epsilon \rho \bar{u} \otimes \bar{u}) + \epsilon \nabla P - \nabla \cdot \mu \nabla \epsilon \bar{u} + \tilde{W} \epsilon \rho \bar{u} - \epsilon \rho \bar{g} = 0 \quad ,$$

is a coarse mesh approximation to the Navier-Stokes equations that is computationally efficient enough for 3-D core calculations at the expense of detailed flow models within an assembly. The viscous terms for an assembly of regularly space rods is obtained from a distributed resistance model given by

$$\vec{R} = \underbrace{-p\vec{n}}_{\text{Form drag}} + \underbrace{\vec{\tau} \cdot \vec{n}}_{\text{Friction drag}}$$


Initially, the equations will be solved only for single-phase coolant (water or gas), which is adequate for resolving inter-assembly flow of water in a PWR core.

Analysis of a PWR undergoing an ATWS event will also require a 3-D time-dependent neutron transport solver and associated nuclear data generation. Under MOOSE, this capability is afforded by RattleSnake and MAMMOTH. These applications are being developed for other projects and thus can be exploited for ESTAP problems with a minimum of development.

Multiphase fluid flow occurs in boiling water reactors and steam generators. The theory can also be applied to thermal flow stratification in pipes and to the mixing of flow streams containing coolants of different compositions (e.g. boric acid). Essentially this involves solving the mass, energy, and momentum conservation equations for each stream with an additional equation and closure relations to account for the transfer of those quantities between the streams. A well-posed, seven-equation set has been implemented in RELAP-7 to model 2-phase fluid flow (liquid and vapor water) moving in one dimension [18]. This is a particularly challenging problem in that it involves the modeling of complex phenomena such as heterogeneous boiling, flashing or cavitation, and bubble collapse. Most existing 2-phase flow solvers make use of various simplifying assumptions such as Homogeneous Equilibrium in which the vapor and liquid pressures are equal and at saturated conditions. The reduced equation set is much simpler to solve. The RELAP-7 equation set reduces to this simpler set if the underlying assumptions are valid.

The PRONGHORN equation set is another such simplification that is applicable to the case of a fluid flowing through a porous medium. A reactor core can be simulated in this manner by ‘smearing’ the solid and fluid components of a flow channel using appropriate homogenization techniques and closure relations. As discussed above, this necessarily sacrifices local fidelity to attain computational efficiency and, under certain flow conditions, is useful for modeling large flow volumes with fewer computational cells. For this reason, it is being adapted for modeling large LWR (and HTGR) cores and may also be useful for the containment pressurization studies.

Simpler multiphase phenomena such as gravity-driven fluid stratification can also be treated using the RELAP-7 equations set. Separate but contacting hot and cold liquid streams in a pipe can be modeled but with interphase mass and energy transfer relations that are simpler than what is required to model boiling. These equations are not currently part of the RELAP-7 development plan and thus will be addressed in

the ESTAP task to enable the simulation of stratified coolant flow in the AP1000 (and perhaps the NuScale) reactor that would occur when cold PHRS water is injected into the primary coolant stream.

4.1.3 Research Partners and Coordinated Development

Clearly, the development of a severe accident simulator is complementary to, and heavily reliant upon, the continued development and testing of the RELAP-7 code as well as other MOOSE-based applications (MAMMOTH, RAVEN). Lessons learned in developing the multi-phase, one-dimensional computational fluid dynamics solver (BIGHORN) being developed for RELAP-7 can be applied to the development of the three-dimensional, coarse-mesh solver. The numerics of 3-D flow are considerably more difficult so close coordination with the MOOSE developers will be essential.

EPRI shall play an important role in high-level technical steering and in detailed planning and execution of demonstration cases. EPRI also would assist in engaging other industry stakeholders to support development and evaluate technical results from the method, tools, and data developments.

4.2 Subtask 2 – Enhanced Containment Analysis for Containment Pressurization and Ventilation

In Subtask 2 efforts will focus on developing a true 3-D and multi-phase flow with turbulence modeling analysis capability to analyze large open spaces within a containment or confinement building to replace the control volume approach. Initially, the thermal and fluid phenomena of the containment atmosphere will modeled with the coarse-grained solver PRONGHORN, with appropriate multiphase adaptations. This solver will be coupled with an existing containment analysis tool (GOTHIC-3D) to provide the appropriate boundary conditions and forcing functions for solving the detailed containment atmosphere problem. GOTHIC-3D already possesses a multi-dimensional finite volume solver option which is an improvement over the simple control volume solver but it may not have the fidelity needed for the filtered venting problem. A MOOSE-based finite element solver will be coupled for this particular scenario. The coarse-mesh, porous medium fluid solver will be based on the PRONGHORN method but will also need to treat natural circulation, non-condensable gases, aerosols, droplet flow and evaporation, and condensation on the containment wall. Later, three-dimensional hydrogen transport and combustion capability and fission product transport and deposition capability will be developed.

The enhanced code will be used within the RISMIC Toolkit framework to assess the risk of taking credit of containment accident pressure. Current PRAs can estimate the likelihood of pre-existing containment leakage. However, they cannot evaluate the risk of small amount of leaks and other evolving conditions that may reduce the available net positive suction head. Therefore, analyses of the impact of loss of containment integrity are needed and can be used to evaluate the likelihood of scenarios in which large amounts of containment overpressure credit are required for a significant amount of time. The previous methods are based on the licensing-basis analyses. The judgment of whether to grant containment overpressure credit for a particular application would depend on more realistic and deterministic evaluation of the required amount and duration of containment overpressure credit, the likelihood of scenarios that would require containment overpressure credit, and the operator actions required to maintain containment overpressure from adequate pump net positive suction head. The deterministic evaluation would include uncertainty quantification.

4.2.1 Demonstration Case

An early model (Mark I/II) BWR provides the most relevant case for testing the analysis capability. The Peach Bottom 2 reactor is a Mark I BWR. It is subject of an OECD Couple Code Benchmark [19] and thus design features are readily available.

4.2.2 Development of Tools

While a code like RELAP5 may be able to model some aspects of a transient involving core damage. Many pertinent phenomena are beyond its simulation capabilities. Inside the vessel, such phenomena include: circulation of steam and non-condensable gases, oxidation of core materials (especially Zircaloy cladding), loss of core geometry (clad ballooning and rupture, melting of fuel, cladding, and other structures, relocation, embrittlement and fragmentation of corium), heating of the lower head, and release of fission products from the fuel matrix and transport to the coolant. As such events are either caused by or lead to a breach of the primary coolant boundary, integral severe accident analysis codes also must be able to model ex-vessel phenomena including containment thermal-hydraulics, hydrogen accumulation, aerosol behavior, corium spreading and interactions with concrete and other materials. Codes such as MELCOR, ASTEC, MAAP, and GOTHIC, therefore, are used to estimate radiological releases from plants the reactor codes of which have suffered significant damage upon a loss of coolant. These are relatively fast-running system codes that capture, in an integral sense, the wide variety of phenomena that occur in conjunction with the massive failure of fuel and other plant structures.

GOTHIC has both lumped parameter-based method and computational fluid dynamics (CFD)-like field simulation capability. Its CFD capability contains simplified turbulence models and outdated numerical methods that could not take advantage of modern, high-performance computing such as parallelization. It also is limited by the number of elements that can be used for a specific analysis.

Another candidate code is CONTAIN[20]; a code developed at Sandia to model specifically the complex phenomena and interactions that occur in the containment building. While lacking GOTHIC's basic CFD capability, CONTAIN captures most, if not all, of the phenomena affecting the performance of a containment vessel, again in an integral sense [Figure 5]. A proposal to couple CONTAIN to the REALP-SCDAP[21] mechanistic severe accident code was never implemented.

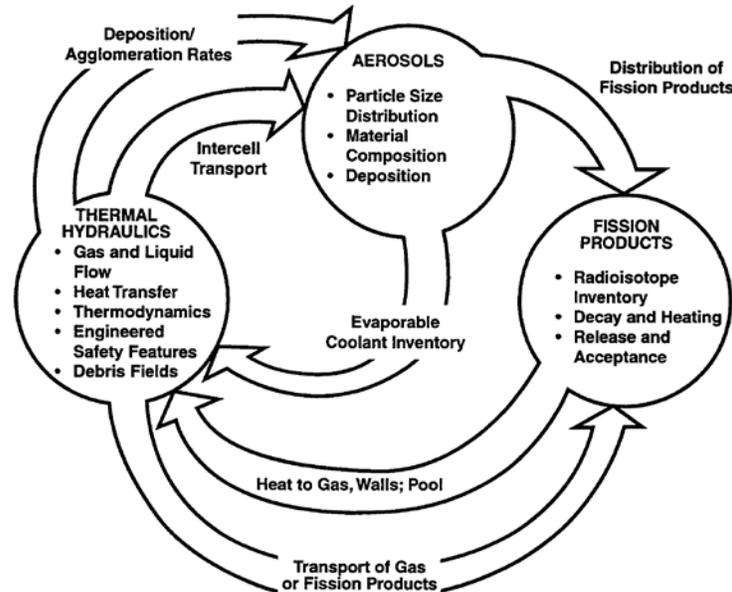


Figure 5 : Feedback mechanisms modeled in the CONTAIN code [20]

As with GOTHIC, the major portions of the containment and internals are modeled as individual 'cells' (control volumes) or small numbers of connected cells. A cell may contain both liquid and gas phases of a species with some cells representing the top part of a suppression pool and the gas space immediately above it [Figure 6]. Balance equations (average mass, energy, etc.) are solved for each cell

with terms presenting the bulk flow between cells. *Within the control volume, however, the fluids are assumed to be stagnant and well-mixed.*

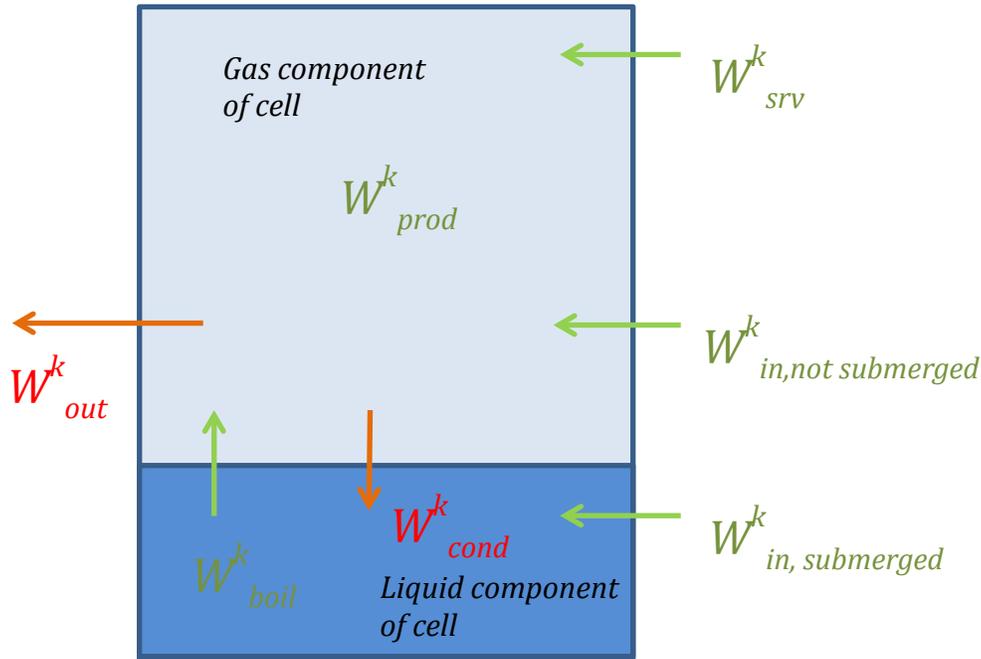


Figure 6: Mass Balance in a CONTAIN cell

The major processes that are modeled include intercell flow, hydrogen combustion, heat and mass transfer processes (e.g., convection, condensation, condensate film flow, thermal radiation, conduction, and concrete outgassing), aerosol behavior (e.g., agglomeration, deposition, and condensation), fission product behavior (e.g., decay, heating, and transport), engineering safety features (ESFS) (e.g., containment sprays, fan coolers, and ice condensers), processes associated with, but not limited to, boiling water reactors (e.g., vent clearing, gas-pool equilibration, and aerosol scrubbing), direct containment heating (DCH) caused by high pressure ejection of finely divided core debris from the reactor vessel, and core-concrete interactions (CCIS). If the flow between the cells occurs through an engineered flow path, critical or choked flow, gravitational heads, and scrubbing effects are also modeled. Like many severe accident codes, CONTAIN does not treat in-vessel processes, the user in many cases must rely upon separate analyses to determine the sources of mass and energy to the containment. Rudimentary thermal-hydraulic models can be modeled in CONTAIN and they may suffice for some cases.

As lumped-parameter, control volume-based codes, however, severe accident simulators generally do not simulate these phenomena at lower length scales. (The 3-D solver in GOTHIC addresses this deficiency to some extent. See [22] for an example.) On the other hand, it is unnecessary and impractical even to attempt to model all of the phenomena occurring in a core melt/primary breach accident with a finite element code. Again, an enhanced systems analysis will allow the user to focus on specific components or phenomena at all relevant length scales from the millimeter to the hundred meter scale, spanning from capturing boundary layers from small scales (such as condensation layer with non-condensable gas and hydrogen plume) to large-scale simulations (such as the entire containment recirculation pattern). The multi-physics phenomena simulated will include multi-phase fluids flow, heat transfer, aerosol transport and deposition in containment, condensation on the containment walls, and hydrogen transport and combustion.

For Subtask II which focuses on venting of the containment, the cell or group of cells comprising the containment atmosphere will be replaced with a coarse mesh finite element model (FEM) built for simulation using a modified PRONGHORN solver or similar application capable of multiphase fluid analysis. Density-driven stratification can then be modeled explicitly to yield a better representation of the flow near and through the ventilation system [Figure 7].

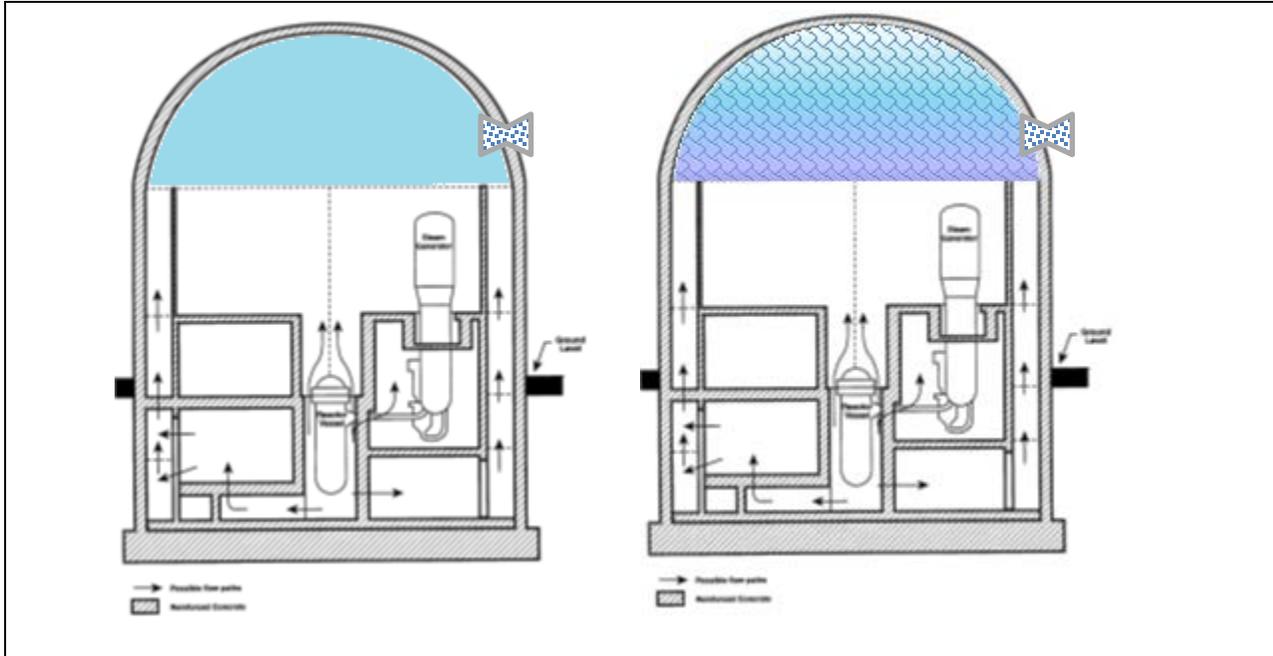


Figure 7: Schematic of vented containment building [20] showing the containment atmosphere modeled as a single control volume (left) and discretized for finite element analysis

An initial approach will be to incorporate as few phenomena into the FEM model as is necessary to model the transient. The difficult data transfer and numerical issues can be worked out more easily and the results of the FEM analysis, averaged over the volume, can be compared to the reference model before adding additional complexity. The initial phenomena will be predominantly thermal fluid in nature such as: heat and mass transfer processes (e.g., convection, condensation, condensate film flow, conduction,), aerosol behavior (e.g., agglomeration, deposition, and condensation), and BWR behavior such vent clearing, gas-pool equilibration, and aerosol scrubbing. These will fully exercise the multiphase solvers being developed.

4.2.3 Research Partners and Coordinated Development

As with Subtask I, EPRI would play an important role in high-level technical steering and in detailed planning and execution of demonstration cases. EPRI can also facilitate collaboration with the GOTHIC developers (Numerical Applications Inc.)

5. VERIFICATION AND VALIDATION

Validation and verification (V&V) of the severe accident analysis tools shall follow the RISMIC Toolkit V&V philosophy and plan as described in [1].

Data from experiments that can be used for validating various components of the enhanced solver include the following.

- 1) 3-D Core Thermal Fluids – In-core fluid temperature data for validating a 3-D thermal-hydraulic calculation do not exist. The OECD LOFT experiment provides transient system-wide plant

data which will be used for validation of the system response. The BEAVRS[23] data set provides data for validating the neutronic calculation and some thermal-hydraulic data such as reactor outlet temperatures. A BEAVRS model has been constructed for validating parts of a MOOSE PWR simulation and will be used for Subtask 1 model validation as appropriate. Test loop experiments and computational benchmarks will be used to confirm but not validate temperature and flow profiles generated by PRONGHORN.

- 2) Stratified Flow in Pipes - The OECD/NEA ROSA project [24] aimed to resolve issues in thermal-hydraulics analyses relevant to light water reactor (LWR) safety using the Japanese ROSA/LSTF facility. In particular, it focused upon the validation of simulation models and methods for complex phenomena that may occur during design basis events (DBE) and beyond-DBE transients. A key objective of the OECD/NEA ROSA project was to provide an integral and separate-effect experimental database to validate code predictive capability and accuracy of experimental thermal-hydraulic models. In particular, phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows are to be studied. The project included the following types of ROSA large-scale experiments:
 - temperature stratification and coolant mixing during emergency coolant injection;
 - unstable and disruptive phenomena such as water hammer;
 - natural circulation under high core power conditions;
 - natural circulation with superheated steam;
 - primary cooling through steam generator secondary depressurization

Results from these tests will be useful for testing different parts of the PRONGHORN and BIGHORN solvers.

- 3) Containment Pressurization - As part of the Cooperative Containment Research Program[25] between the Nuclear Power Engineering Corporation of Japan and the NRC, Sandia National Laboratories built a 1:4 scale model of a prestressed concrete containment vessel (PCCV) pressurized it to failure. The prototype for the model is the containment building of Unit 3 of the Ohi Nuclear Power Station in Japan. The design accident pressure, P_d , of both the prototype and the model is 0.39 MPa (57 psi). The objectives of the PCCV model test were to simulate some aspects of the severe accident loads on containment vessels, observe the model failure mechanisms, and obtain structural response data up to failure for comparison with analytical models.
- 4) Hydrogen Combustion – validation data was generated in the Flame Acceleration Measurement and Experiments test [26] and the Large-Scale Vented Combustion Tests Facility Experiments (LSVCTF) [27].

The FLAME experiment involves a large horizontal rectangular channel made of heavily-reinforced concrete designed and built for the US NRC. It is a half-scaled model of the upper plenum volume found in ice condenser PWR containments. In the FLAME experiments, twenty-nine sets of test were executed, during which the hydrogen mole fraction was varied from 12 % to 30 %.

The LSVCTF experiments are large-scale combustion tests also conducted in a rectangular channel. An igniter was located at the center of the volume and a vent is located on one of end walls to guarantee the flame flows. The vent area could be changed by removing or replacing the

appropriate number of panels. Temperature and pressures were measured locally to provide data regarding flame propagation.

Other validation experiments will be assessed and used as appropriate.

6. COST AND SCHEDULE

In Section 4, we described the demonstration case and initial use-case to be investigated. These activities are part of the overall RISMIC Toolkit development process, which is outlined in Figure 8. Early in the project, the initial strategy development (which this document is a part of), the stakeholder engagement to obtain input on the strategy, and the Integrated Program Plan would also be completed. Then, following the demonstration and initial use-cases, the R&D for the other high-priority use-cases would be conducted over the course of six years. As a parallel activity, the associated validation of the methods, tools, and data for specific uses-cases will be conducted over the same time period.

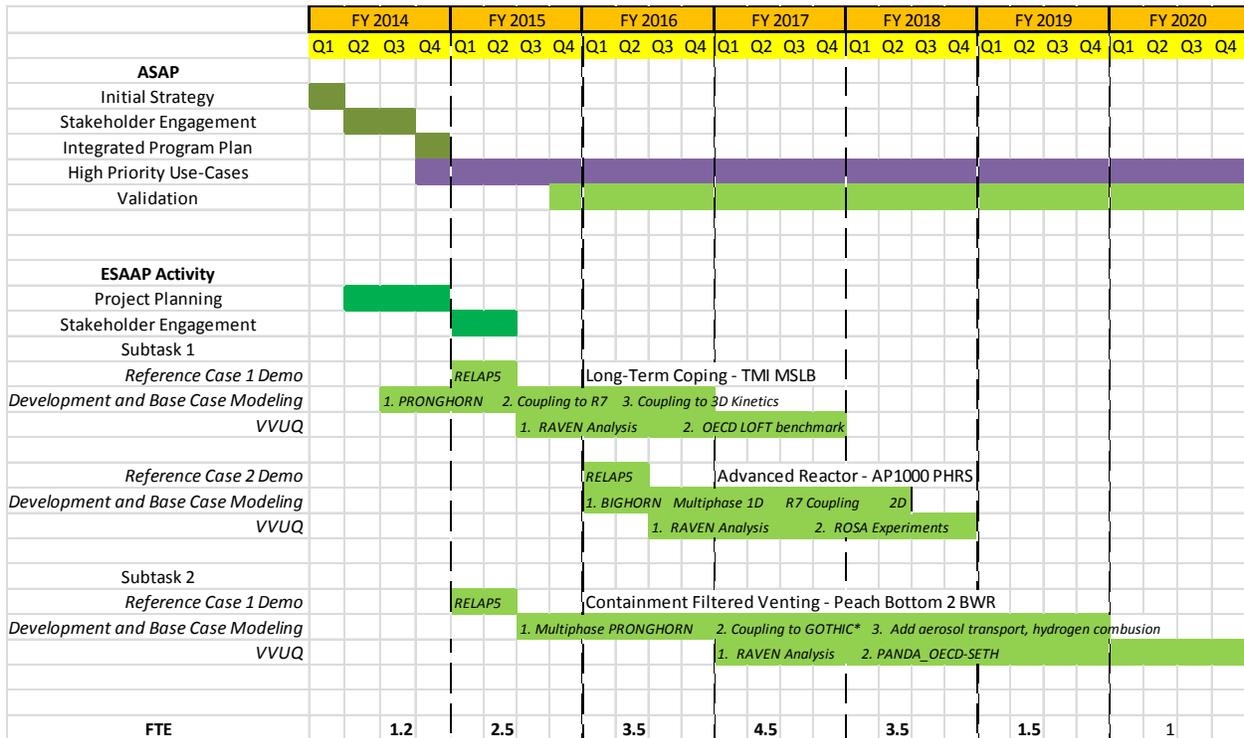


Figure 8. Overall schedule and proposed staffing levels for the ESTAP.

The cost to develop the ESTAP methods, tools, and data (including validation using existing information and data sets) is estimated to require a total of four Full Time Equivalent (FTE) development staff in the initial stage (as the program is ramped up) to an activity level of approximately four FTE per year over three years before ramping down at project closeout. The assumptions behind this activity level are based upon recent progress with RELAP-7 development, PRONGHORN development under the Next Generation nuclear Plant program, and NEAMS activities in which modern methods and tools development takes place over a couple of years. As with other MOOSE-based codes, it is expected that roughly 1/2 to 2/3 of the funding will be used for validation and uncertainty analysis. Since the schedule has been designed to address two to three use-cases concurrently, a target investment to successfully carry out the R&D is maintained at the four to six FTE/yr level. The methods, tools, and data will be made available to the users as they become available, starting with the first use cases at the end of the third year after the initiation of the program.

It is expected that DOE would bear the costs of development, conduct the workshops to inform the users of the capabilities, and perform the initial validation. It is expected that, once the use-cases are completed, the end users would fund their own adoption, installation, and training.

7. Summary

This report outlines a plan for enhancing existing severe transient analysis capabilities using recently developed computational methods and capabilities. The Enhanced Severe Accident Analysis Prevention (ESTAP) effort is split into two subtasks: 1) pre-core damage analysis using an enhanced system code (RELAP-7) and 2) containment analysis using an enhanced containment analysis code. Both subtasks will exploit the computational capabilities of the Multiphysics Object-Oriented Simulation Environment (MOOSE) under development at the Idaho National Laboratory.

The first subtask aims to enhance the ability of system analysis codes with high fidelity, spatially resolved subsystem physics solvers that reveal complex phenomena not captured by traditional control volume approaches. Such tools can provide a designer with more complete information on the behavior of the plant under extreme transient scenarios, thereby assisting in the development of preventative or remedial measures that can delay or prevent core damage. The enhancements, mainly a 3-D core thermal fluid solver and a multiphase CFD solver, can also be applied to advanced reactors that rely on passive heat removal mechanisms that are not easily modeled with system codes.

The second subtask uses a similar strategy to augment a containment analysis code with a multi-dimensional, multiphysics code that will model reactor containment fluid dynamics and physics. The enhanced code will be used for detailed analysis of containment venting during a loss of AC power and after primary depressurization.

In both tasks, the multiphysics, multicomponent modeling capabilities and computational efficiency of the system codes are largely retained. It is neither practical nor necessary to re-build an entire accident analysis tool from the ground up on the MOOSE platform. Instead, the MultiApps and Transfers code coupling protocol will be exploited to efficiently shuttle data between the base code and the high fidelity subsystem model. Even with this capable utility, retaining accuracy, numerical stability, and speed will be the primary development challenge.

Progress in the ESTAP effort will hinge on concurrent progress with the development of the RELAP-7 system code. In particular, the same improvements in the computational efficiency of the RELAP-7 two-phase flow solver will be needed to ensure computational stability and efficiency in the coarse mesh core solver (PRONGHORN) and the multi-phase CFD solver (BIGHORN). Close coordination with the RELAP-7 and MOOSE teams will be essential. For the second subtask, the additional challenge of working with an existing, non-INL code system will require coordination with the code (GOTHIC) developers.

Reference (near term) simulations of the Use-Cases addressed in this effort will be conducted using the RELAP5-3D code. Models will be constructed of a Westinghouse 4-loop reactor core, the OECD Three Mile Island Main Steam Line Break benchmark, the AP1000, and the Peach Bottom 2 Turbine trip benchmark. Data for validation of the enhanced solvers will be obtained from relevant past experiments including the ROSA series of experiments, the recently developed BEAVRS benchmark, the Cooperative Containment Research Program, and smaller experiments re-creating severe accident phenomena. The entire project is will require approximately 18 Full-Time-Equivalent developers and analysts.

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