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Oak Ridge National Laboratory Support of Non-light Water Reactor Technologies: Capabilities Assessment for NRC Near-term Implementation Action Plans for Non-light Water Reactors

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OAK RIDGE NATIONAL LABORATORY
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Reactor and Nuclear Systems Division

OAK RIDGE NATIONAL LABORATORY SUPPORT OF NON-LIGHT WATER REACTOR TECHNOLOGIES: CAPABILITIES ASSESSMENT FOR NRC NEAR-TERM IMPLEMENTATION ACTION PLANS FOR NON-LIGHT WATER REACTORS

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CONTENTS

ACRONYMS

ADTR	Advanced Demonstration and Test Reactor
CASL	Consortium for Advanced Simulation of Light-water Reactors
CFD	computational fluid dynamics
DSRS	design-specific review standard
DOE	US Department of Energy
FFH	fusion-fission hybrid
FHR	fluoride salt-cooled high-temperature reactor
HFIR	High Flux Isotope Reactor
HTGR	high-temperature gas-cooled reactor
IE	initiating event
LMR	liquid metal reactor
LWR	light water reactor
MACCS2	MELCOR Accident Consequence Code System
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
NCSU	North Carolina State University
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NEUP	Nuclear Energy University Program
NRC	US Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PIRT	phenomena identification and ranking table
RANS	Reynolds Averaged Navier-Stokes
SFR	sodium-cooled fast-spectrum reactor
THORS	Thermal Hydraulic Out-of-Reactor Safety

ABSTRACT

The Oak Ridge National Laboratory (ORNL) has a rich history of support for light water reactor (LWR) and non-LWR technologies. The ORNL history involves operation of 13 reactors at ORNL including the graphite reactor dating back to World War II, two aqueous homogeneous reactors, two molten salt reactors (MSRs), a fast-burst health physics reactor, and seven LWRs. Operation of the High Flux Isotope Reactor (HFIR) has been ongoing since 1965. Expertise exists amongst the ORNL staff to provide non-LWR training; support evaluation of non-LWR licensing and safety issues; perform modeling and simulation using advanced computational tools; run laboratory experiments using equipment such as the liquid salt component test facility; and perform in-depth fuel performance and thermal-hydraulic technology reviews using a vast suite of computer codes and tools. This expertise is addressed below.

1. NRC STRATEGY 1

1.1 KNOWLEDGE, TECHNICAL SKILLS, AND CAPABILITY TO PERFORM NON-LWR REGULATORY REVIEWS

ORNL has significant experience training students, staff, and other organizations to understand and analyze non-LWRs. Technical publications provide one pathway to accomplish these training and communication objectives, such as writing a journal article providing an overview of MSR technologies and concepts [Gehin 2016]. ORNL also strongly contributed to a recent Department of Energy (DOE) Advanced Demonstration and Test Reactor (ADTR) Options Study report [Petti 2016] that described several advanced reactor technologies, evaluated representative point designs, and discussed processes for developing, licensing, and deploying advanced non-water-cooled reactors. Workshops and symposia sponsored by ORNL provide a second avenue for communication. Such events include sponsoring and hosting the 2015 Workshop on Molten Salt Reactor Technologies and 2016 Molten Salt Reactor Workshop, and hosting the Advanced Reactor Technical Summit III sponsored by the US Nuclear Infrastructure Council in 2016. As a third communication pathway, ORNL staff are frequently asked to deliver talks at conferences, briefings to key decision makers, university seminars, and university lectures to educate undergraduate and graduate students about the status and future for advanced reactors in the US and the world. A fourth approach used by ORNL involves knowledge management and formal training sessions and for advanced reactor technologies. ORNL staff coordinated with US Nuclear Regulatory Commission (NRC) staff to produce a knowledge management NUREG (NUREG/KM-0007) for liquid metal reactors (LMRs). Recent training examples include a session for DOE staff on MSRs, an upcoming training session for NRC staff on MSRs, and lecturing at IAEA training sessions on specific issues or technologies. ORNL staff also supported previous NRC design-specific review standard (DSRS) effort, which involved a systematic review of integral PWRS, high-temperature gas-cooled reactors (HTGRs) and LMRs, including training for the NRC staff.

2. NRC STRATEGY 2

2.1 COMPUTER CODES AND TOOLS TO PERFORM NON-LWR REGULATORY REVIEWS

2.1.1 Functional Area: Reactor Kinetics and Criticality

Significant ORNL research activities involve developing computational tools for modeling and simulating non-LWR reactors, applying tools to analyze advanced reactors, and pursuing validation activities using existing benchmarks to validate code features and identifying or performing experiments needed to fill in gaps.

2.1.1.1 Methods

Recent ORNL development of computational tools for analyzing non-LWR reactors includes development and validation of SCALE nuclear analysis methods for HTGRs [Gehin 2010, Ilas 2012, Ilas 2010, and Sunny 2010]. Significant efforts were also made to enable criticality and fuel cycles simulations of MSRs in SCALE [Powers 2013a]. Recent work includes developing computer codes for MSR reactor kinetics and criticality analysis in SCALE [Betzler 2017a, Betzler 2017b, and Betzler 2017c] and multiphysics simulations using MPACT [Collins 2017a and Collins 2017b]. Salt properties were also added to PARCS for analysis of a fluoride-salt-cooled high-temperature reactor (FHR) [Qualls 2017]. ORNL initiated efforts during 2016 to examine the use of SCALE for analysis of sodium-cooled fast reactors (SFRs), including applying SFR multigroup structures to generate SCALE libraries for SFR calculations [Bostelmann 2017]. This work establishes the foundation for follow-on activities that would continue to improve SFR capabilities in SCALE. ORNL activities in support of advanced reactor modeling and simulation in general include work to develop the Nuclear Energy Advanced Modeling and Simulation (NEAMS) Workbench to provide a user-friendly interface for front-end modeling and back-end results visualization for reactor modeling and simulation tools [Rearden 2017]. Under FY17 NRC funding, ORNL is integrating the advanced Monte-Carlo code Shift into SCALE to provide a reference depletion simulation capability for advanced reactors and generating nodal data for PARCS using advanced Monte Carlo methods.

2.1.1.2 Analysis Applications

A significant number of recent projects and efforts at ORNL have applied modeling and simulation tools to advanced reactors to perform analyses and simulations. ORNL has been heavily engaged with neutronic and fuel cycle analysis of interest to the DOE Fuel Cycle Options Campaign. These efforts covered a range of reactor technologies including uranium- and thorium-fueled MSRs with thermal or fast neutron energy spectra and with fuel cycle approaches ranging from once-through to continuous (full) recycle [Brown 2015, Betzler 2016a, Betzler 2016b, Gehin 2016, Powers 2014a, and Sunny 2015]. ORNL also analyzed subcritical externally-driven system concepts for this effort [Brown 2016c, Fratoni 2013, and Powers 2011], including accelerator-driven system and fusion-fission hybrid (FFH) concepts. Another significant effort at ORNL involved neutronics, thermal-hydraulics, and fuel performance analyses that led to the development of an ORNL pre-conceptual point design for an FHR demonstration reactor (DR) as part of the DOE ADTR Options Study [Brown 2017, Brown 2016b, Qualls 2017, and Qualls 2016]. Other recent non-LWR analysis applications at ORNL include:

1. Neutronic and fuel cycle analysis of the Transatomic Power MSR through the DOE Gateway for Accelerated Innovation in Nuclear [Betzler 2017c];
2. Applying SCALE Sensitivity and Uncertainty capabilities to critical experiments with salt targets as part of an international collaboration between ORNL and Research Centre Rez in the Czech Republic [Brown 2016a and Powers 2017];
3. Neutronic analysis of space reactor concepts [Betzler 2016c]; and
4. Neutronic assessment of novel fuel concepts using TRISO coated fuel particles in HTGRs and FHRs [Powers 2016a, Powers 2014b, and Powers 2013b].

2.1.1.3 Experimental/Validation Needs

Validating computer codes for specific analysis applications remains important and involves both code validation itself and identifying and/or performing experimental work to generate validation data needed for code development and qualification activities. ORNL staff currently partners with researchers from the University of California, Berkeley to create and document benchmark specifications and computer models for experiments performed at ORNL's Molten Salt Reactor Experiment (MSRE) in the 1960s. This project, funded by the DOE Nuclear Energy University Programs (NEUP), aims to submit one or two benchmarks for code comparison and validation use to the International Reactor Physics Experiment Evaluation Project benchmark handbook. ORNL also outlined and performed efforts needed to validate SCALE for use in analyzing HTGR pebble bed reactors [Ilas 2012 and Ilas 2010].

2.1.2 Functional Area: Fuel Performance

ORNL staff has been involved in several recent projects related to developing and applying thermo-mechanical fuel performance codes and models for non-LWR reactors, especially related to application analysis or conceptual design of TRISO coated fuel particles. One example involves fuel performance assessment of TRISO fuel in an FHR as part of developing the ORNL FHR DR for the DOE ADTR Options Study [Brown 2017, Brown 2016b, Qualls 2017, and Qualls 2016]. Substantial efforts were also spent analyzing and assessing the use of TRISO particles for high-fluence applications including LWR fuel concepts and a salt-cooled FFH concept [Powers 2014c, Powers 2013c, and Powers 2011]. Current ORNL staff also authored a comprehensive journal article providing a review (and comparison) of available fuel performance models for TRISO fuel [Powers 2010], which has been cited significantly in the literature because of the gap it filled and the understanding it provides.

2.1.3 Functional Area: Thermal-Fluid Phenomena

The thermal-fluid research at ORNL is focused on delivering outstanding solutions to challenging fluid, heat transfer, and irradiation engineering problems in nuclear systems by leveraging a unique integration of design, simulation, fabrication and experimentation expertise. Thermal hydraulics modeling and simulation at ORNL rely on a combination of custom code development, the leveraging of open source software, and the extension of commercial codes.

For high-resolution computational fluid dynamics (CFD) research, ORNL leverages the Direct Numerical Simulation and Large Eddy Simulation capabilities of the open-source spectral element code, Nek5000, and the open-source code, PHASTA, a code for Parallel, Hierarchic, Adaptive, Stabilized finite-element Transient Analysis. A multiphysics coupling interface was recently developed at ORNL to couple the Nek5000 code with the SHARP analysis framework, allowing detailed simulations of advanced nuclear reactor designs [McCaskey 2014]. ORNL also supports integration of the Nek5000 code with uncertainty quantification and calibration capabilities of an open-source DAKOTA Toolkit [Delchini 2016a]. ORNL has partnered with North Carolina State University (NCSU) to support the continued development of a unique, multi-interface, multiphase simulation capability within the framework of the PHASTA code, which was initially developed by Rensselaer Polytechnic Institute. ORNL also uses and extends several other open-source high-resolution CFD codes, including Hydra-TH, Nalu and OpenFOAM.

For CFD based on the Reynolds Averaged Navier-Stokes (RANS) equation, ORNL partners with commercial code developers to extend the capabilities of their simulation products to address the unique simulation needs for nuclear reactors. Notably, ORNL supports the development and validation of innovative multiphase simulation capabilities in the STAR-CCM+ code as part of the Consortium for Advanced Simulation of LWRs (CASL) thermal hydraulics team and the NEAMS Steam Generator High Impact Problem team [Pointer 2016]. ORNL also maintains a RANS-based analysis capability for the

HFIR core design and safety analyses, constructed within the COMSOL Multiphysics simulation framework [Renfro 2014, Jain 2015].

For subchannel analysis, ORNL supports the development of CTF (previously COBRA-TF) code in partnership with NCSU [Salko 2016]. ORNL focuses on enabling the CTF code to run full multichannel analyses on parallel computing platforms, extending the code for higher void fraction flow regimes found in boiling water reactors (BWRs) [Wysocki 2016], and integrating it with neutronic, fuel performance, and coolant chemistry codes and models. ORNL also partners with Pacific Northwest National Laboratory to support the continued development of the COBRA-SFS code for assessment of thermal hydraulics in spent fuel storage [Robb 2015].

ORNL has recognized expertise in the development, extension, and validation of codes for the assessment of system thermal hydraulics, system dynamics, and reactor safety. ORNL develops TRANSFORM, a unique system analysis code implemented within the Modelica simulation framework [Fugate 2015]. Implementation of TRANSFORM provides most of the capabilities of other system codes, but it is uniquely well suited for system dynamics analysis, as well as instrumentation and control design. ORNL also evaluates and validates the NRC system analysis code, TRACE, for LWR applications and develops modified branches of TRACE for advanced reactor applications. Similarly, ORNL evaluates and validates the system safety analysis code RELAP5 for LWR and advanced reactor applications [Carbajo 2017]. ORNL maintains material property functions for a variety of reactor coolants and structural materials for use in these codes.

Code benchmarking, including verification and validation, continues to be an important element of ORNL's software quality assurance activities to ensure that nuclear design and safety analysis tools can be used to accurately predict the behavior of the facilities to which they are applied. Experiments are sometimes developed to support code benchmarking activities, including thermal hydraulics experiments designed and built for specific applications and coupled with state-of-the art instrumentation to characterize detailed thermal and fluid behavior [McDuffee 2014, 2017].

2.1.3.1 Sodium-Cooled Fast Reactors

The Thermal Hydraulic Out-of-Reactor Safety (THORS) facility was a liquid-sodium heat transfer test setup that was operational at ORNL from the early 1970s through the mid-1980s. Flow and heat transfer experiments were performed to measure temperature, pressure drop, and bulk flow rates in blocked and unblocked SFR wire-wrapped pin bundles. In most tests, the sodium coolant remained as a single-phase liquid, but it was allowed to boil in a few cases. Through support of the DOE NEAMS program, ORNL recently recovered most of the THORS experimental design and data [Carbajo 2016] to validate the Nek5000 and STAR-CCM+ CFD codes and their underlying turbulence models (shear stress transport, k-epsilon and k-omega) for liquid-sodium applications. In addition, the DAKOTA Toolkit was also coupled with the Nek5000 and STAR-CCM+ CFD codes to understand uncertainty propagation in SFR safety applications [Delchini 2016b]. A best practice guidance document was also developed to enable application of codes and software packages for safety assessment of advanced SFR designs [Brown 2016d].

2.1.3.2 Molten Salt Reactors (MSRs)

Through continuing R&D efforts over the last several decades, ORNL has pioneered the development of novel MSR technologies. With current support from DOE's Office of Nuclear Energy, ORNL is developing an advanced high-temperature reactor concept that integrates technologies from several other thermal power plant designs. This new class of reactors, known as FHRs, features low-pressure liquid fluoride salt cooling, coated-particle fuel, a high-temperature power cycle, and fully passive decay heat rejection. ORNL is also performing R&D on advanced salt compatible alloys, ceramic composite materials, TRISO fuel fabrication, and liquid-salt compatible component development. Sustained analysis efforts include CFD analysis of components with STAR-CCM+, implementation of salt properties into TRACE, and system level analyses with TRACE, RELAP5-3D, and MoDSIM codes [Hale 2014, Richard 2014, Yoder 2014b, Carbajo 2017]. As part of the FHR technology development, ORNL designed and constructed a liquid salt component test facility [Yoder 2012, 2014a] to establish baseline operations with the coolant salts at temperature with materials and in developing appropriate components suitable for salt operations.

2.1.4 Functional Area: Severe-Accident Phenomena

ORNL has extensive institutional experience in performing research for severe accidents in LWRs. For example, in the 80s and 90s, ORNL led NRC-sponsored programs in the following areas:

- Accident Sequence Precursor,
- Source Term/Fission Produce Release,
- Severe Accident Sequence Analysis
- Detailed Assessment of BWR In-Vessel Strategies

Many hot cell experiments were also performed to study the fission product release behavior and speciation, as well as Zircaloy cladding oxidation and rupture. Several codes and models were developed, most of which have now been assimilated into the most recent suites of MELCOR, SCDAP/RELAP and MAAP codes.

The accidents at Fukushima Daiichi have renewed ORNL's severe accident research focus with several modeling and simulation activities [Carbajo 2012; Wang 2012; Robb 2013, 2014a, and 2014b] through Fukushima accident analyses and event reconstruction studies. Recent ORNL efforts have led to improvements in ex-vessel core melt spreading and coolability modeling in the MAAP and MELCOR codes. The response of BWR upper internals to impact the progression of accident sequence was also analyzed using the MELCOR code [Robb 2017]. ORNL also participated in a multi-institute technology gap evaluation study to address the key knowledge gaps in severe accident phenomenology that affect reactor safety. These areas are not being addressed directly by the nuclear industry or by the NRC [Farmer 2015]. In addition, ORNL also uses MELCOR to evaluate the overall dynamic response of HFIR following a large-break loss of coolant accident leading to core damage. ORNL has varied experience in applications of other severe accident codes, including, SCDAP /RELAP, CONTAIN, and SAS4-A, M codes.

2.1.5 Functional Area: Offsite Consequence Analysis

ORNL routinely performs confirmatory simulations using the MELCOR Accident Consequence Code System (MACCS2) to perform offsite consequence analysis. For example, the environmental impact of an early site permit application by PSEG Power, LLC was evaluated using MACCS2 [NRC 2015]. Using the MACCS2 code, ORNL has also evaluated the probability-weighted consequences, including the health risks, for a variety of advanced LWR designs located at the PSEG site, as well as the consequences of dispersion and deposition of radionuclides released for a proposed Advanced Neutron Source research reactor. In addition, ORNL intends to use the MACCS2 code to support off-site dose consequence analysis for the low enriched uranium conversion of the ORNL HFIR.

2.1.6 Functional Area: Materials and Reactor Component Integrity

ORNL has a rich history in nuclear material science. ORNL advances nuclear materials science by conducting innovative research and development projects for a broad spectrum of nuclear fusion and fission power and fundamental science programs. Current studies support research and development of structural, functional, and plasma-facing materials for fusion energy; structural and core component materials for both the current light water reactor fleet and future nuclear fission reactors; naval and other specialized nuclear applications; and highly accident-tolerant nuclear fuels and core components for advanced fission reactors. ORNL staff studies radiation-induced changes in physical, mechanical and structural properties of materials through simulation and analysis of experimental or commercial reactor irradiated materials. ORNL staff has had lead roles in material studies related to the International Thermonuclear Experimental Reactor fusion energy project, the LWR Sustainability program, and the DOE Advanced Reactor Technology program.

3. NRC STRATEGY 3

3.1 ADVANCED NON-LWR REGULATORY REVIEW PROCESS

3.1.1 Functional Area: Regulatory Issue Relationships

ORNL is frequently called upon to examine regulatory issues. For example, ORNL evaluated the historical, current, and proposed uses of thorium in nuclear reactors including performance of qualitative and quantitative evaluations of reactor safety issues and identification of key knowledge gaps and technical issues that need to be addressed for the licensing of thorium fuel [Ade 2013]. ORNL also conducted an evaluation of the safety characteristics and licensing approach for FHRs; a type of MSR. The FHR work evaluated a General Design Criteria based approach to licensing an MSR [Flanagan 2012].

3.1.2 Functional Area: PRA and Event Selection

ORNL has developed PRA and event selection approaches for emerging plant designs, including non-LWRs. For example, two risk-based surrogates have been developed for LWRs in order to show compliance with quantitative health objectives. The LWR surrogates consist of a preventative component (core damage frequency) and a mitigation component (conditional containment failure probability). ORNL has leveraged the LWR approach (as contained in SECY-89-102, Regulatory Guide 1.174, and NUREG-1860) to propose risk-based surrogates for broad use in non-LWR designs [Flanagan 2013] and specific use in HTGRs [Ball 2014]. Furthermore, ORNL identified a preliminary list of initiating events (IEs) for HTGRs and SFRs. IEs caused by internal events such as equipment failures and human errors, internal plant hazards such as internal fires and floods, and external plant hazards such as seismic events and accidents at nearby facilities were considered for all operating states [Muhlheim 2013]. ORNL staff subsequently evaluated multi-unit IEs for advanced reactors [Muhlheim 2014]. Finally, ORNL performed dynamic system modeling of various non-LWR systems and components including HTGRs, FSRs, and MSRs [Cetiner 2013, Hale 2014, and Hale 2015].

3.1.3 Functional Area: Identify Regulatory Framework Gaps

ORNL worked directly on identifying regulatory gaps for advanced reactors, including non-LWRs. A major NRC effort included creation of performance-based, risk-informed DSRS for near-term integral PWR designs. The DSRS effort also involved a systematic review of HTGRs and LMRs. ORNL subsequently proposed and co-led a DOE initiative to propose a set of technology-neutral advanced reactor design criteria for use by non-LWR designers and NRC staff. ORNL further interacted with NRC staff to support adapting the DOE proposal into an NRC Regulatory Guide on advanced reactor design criteria. Currently, ORNL staff is performing a gap analysis and subsequent proposed update to NUREG-0800, Chapter 4 to make the review guidance relevant to HTGRs and SFRs [Poore 2016 and Belles 2016]. ORNL is repeatedly called upon to support other regulatory gap initiatives such as expert support of phenomena identification and ranking table (PIRT) development of advanced non-LWRs. For example, ORNL led an expert panel for an HTGR design [Ball 2007a]. The HTGR PIRT was followed by a project to incorporate evaluations of risk to determine important gaps in the knowledge base and further recommend how these gaps might be addressed [Ball 2007b].

As the institution that developed and operated the only two MSRs, ORNL has an extensive MSR knowledge base that can be leveraged for technical and regulatory gap analyses of this technology. An overview of fast-spectrum MSR options was prepared for the DOE Advanced Reactor Options program [Holcomb 2011] and numerous MSR technology and trade-off studies, along with dynamic system modeling projects, have been performed at ORNL over the last decade [Greene 2010, Holcomb 2009, Holcomb 2010, Qualls 2011, and Qualls 2016].

3.1.4 Functional Areas: Develop Regulatory Review Roadmap and Prototype, Research, and Test Reactor Guidance

Government, academia, and industry often seek out ORNL expertise for reactor design, licensing and safety issues, and regulatory, material, and testing needs. For instance, ORNL frequently collaborates with academia on NEUP projects related to advanced reactors. ORNL worked with NASA on the Jupiter Icey Moon Orbiter space reactor project to resolve licensing and safety issues, and establish the process to setup and operate a ground test facility. At the request of DOE, ORNL identified a safety and licensing research, development, and demonstration path forward for FHRs that included both test and power reactors, as well as the role of safety standards in the approval process [Flanagan 2012]. Subsequently, a roadmap was prepared describing the remaining FHR principal technology challenges [Holcomb 2013]. ORNL frequently performs technology-readiness-level assessments for the major systems, structures, and components of various designs [Holcomb 2014a and Holcomb 2014b].

ORNL staff provided a revision for Chapter 7 of NUREG-1537, the non-power reactor standard review plan, to the NRC staff. In addition, ORNL provided significant input toward the preparation of interim staff guidance to NUREG-1537, related to an aqueous homogeneous reactor. Industry has further sought out ORNL expertise to team in a DOE cost-share advanced reactor initiative. ORNL is utilizing its licensing and safety expertise to perform a gap analysis of the actions needed to make NUREG-1537 applicable to non-power MSR designs.

4. NRC STRATEGY 4

4.1 INDUSTRY CODES AND STANDARDS TO SUPPORT ADVANCED NON-LWRS

At the direction of DOE and in cooperation with NRC, ORNL is conducting a pilot project to evaluate the application of codes and standards in NRC regulatory guides to SFRs; specifically, what codes and standards are currently applicable and where gaps exist. The effort is an attempt to quantify and prioritize the scope of work needed to provide appropriate codes and standards for non-LWR licensing. In addition, ORNL staff are leading or lending significant insight for several American Nuclear Society Standards including ANS 20.1, Nuclear Safety Criteria and Design Process for FHRs, ANS 20.2, Nuclear Safety Design Criteria and Functional Performance Requirements for liquid-fuel MSRs, and ANS 54.1, Nuclear Safety Criteria and Design Process for SFRs.

Materials evaluation for advanced non-LWRs is also an ORNL strength. For example, cladding and component alloys have been evaluated for compatibility with a high-temperature fluoride salt environment and high-temperature HTGR environments [Muralidharan 2011]. Materials work at ORNL supports necessary regulatory gap analyses needs for non-LWRs as well as needed codes and standards evaluations.

5. NRC STRATEGY 5

5.1 TECHNOLOGY-INCLUSIVE POLICY ISSUES IMPACTING REGULATORY REVIEWS, SITING, PERMITTING, AND LICENSING OF NON-LWR NUCLEAR POWER PLANTS

ORNL often evaluates licensing policy issues for non-LWRs. For example, ORNL identified and evaluated regulatory implications concerning digital control and protection systems proposed for use in HTGR designs. This included new and emerging measurement technologies with high potential to improve operations, maintenance, and accident response designed with passive safety features [Ball 2012 and Wilson 2012].

ORNL also developed a GIS-based power plant siting tool. For example, the siting of large and small LWR plants as well as advanced non-LWR plants have been evaluated in support of DOE, EPRI, and private industry [Mays 2012 and Belles 2012]. ORNL is actively tracking the Clinch River early site permit application for policy impacts for advanced non-LWRs; particularly the emergency protection zone [Belles 2016].

6. SUMMARY

ORNL's vast experience with HTGR, SFR and MSR fuel and thermal-hydraulic codes and tools will immediately benefit the NRC in execution of the Near-Term Implementation Action Plan. ORNL is well positioned to leverage domestic and international experience to assist in evaluating and down-selecting appropriate codes for use by the NRC staff to review non-LWR applications. In addition, ORNL has extensive capability to support the NRC staff with issues surrounding the non-LWR regulatory review process including PRA expertise and development of advanced reactor design criteria. ORNL has demonstrated leadership in the area of reactor materials, codes, and standards. Finally, ORNL is an active participant in resolving prototype, research, and test reactor issues.
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APPENDIX A. REFERENCES

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