



Developing a Licensing and Regulatory Framework for Light Water and Non Light Water Small Modular Nuclear Reactors

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Executive Summary

As an increasing number of large light water reactors are being decommissioned and as the need for clean energy grows, the path forward will be spearheaded by the commercialization of small modular nuclear reactors. Light water reactors will be the first designs to undergo the licensing process. However, in the future, it is possible advanced designs such as high temperature gas reactors or molten salt reactors will become integrated into the electrical industry. Small modular reactors have many advantages over their larger counterparts, such as having more siting flexibility, a lower maximum power, and modularity to scale to meet energy requirements. Prospective designs utilize more passive safety features that require less human interface, and feature underground siting, which enhances safety and security. These features make designs more suitable to place near large population clusters. Many of these passive safety features have been successfully implemented in large advanced reactors such as the AP 1000.

Currently, the Nuclear Regulatory Commission (NRC) has outlined several issues that need to be addressed if the commercialization of these reactors is to move forward. Industry and government alike both recognize the need to develop policies that acknowledge the unique design advancements that promote competition, while still maintaining a manageable framework to make the licensing process more cost effective and more efficient.

Issues such as siting, safety, staffing, and design considerations top the list in terms of priority. One of the main obstacles in the safety regulation process, especially for non-light water reactors, is accident scenario analysis, which varies based on reactor type. The probabilistic risk assessment tool should be utilized primarily when selecting accidents for

safety analyses. Industry will seek the assistance of national labs for development of this tool along with mechanistic source terms before applying for NRC plant design review. The emergency planning zone should be established for plants instead of individual modules, and be based on a dose criterion for severe design basis accidents.

Siting considerations include the exclusion area boundary and low population zone, which is proposed to be scalable using the same methodology for establishing the emergency planning zone. This methodology for siting would not require changes to current NRC regulation. New criterion needs to be established for geologic and seismic characteristics for underground plant sites. Staffing for control rooms is purely design based and requires extensive review on a case-by-case basis, while security staffing requirements should be made scalable based only upon the number of reactor modules in the plant and number of entry points to the plant. Design based considerations include the approach to licensing plants, which should be done by issuing licenses to individual reactor modules while referencing a license appendix. Additionally, the application process should have a phased design certification application for Title 10 Code of Federal Regulations Part 52.

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List of Acronyms

10 CFR – Title 10 of the Code of Federal Regulations

AEA – Atomic Energy Act

BDBA – Beyond Design Basis Accident

CDF – Core Damage Frequency

COL – Combined Construction and Operating License

DBA – Design Basis Accident

DC – Design Certification

DCA – Design Certification Application

DOE – Department of Energy

EAB – Exclusion Area Boundary

EPZ – Emergency Planning Zone

ESP – Early Site Permit

FSAR – Final Safety Report

HFE – Human Factors Engineering

HTGR – High Temperature Gas Reactor

ITAAC – Inspections, Tests, Analyses, and Acceptance Criteria

LOCA – Loss of Coolant Accident

LPZ – Low Population Zone

LWR – Light Water Reactor

MST – Mechanistic Source Term

NEI – Nuclear Energy Institute

NRC – Nuclear Regulatory Commission

NUREG – U.S. Nuclear Regulatory Commission Regulation

PRA – Probabilistic Risk Assessment

PRISM – Power Reactor Innovative Small Module

SFR – Sodium Fast Reactor

SMR – Small Modular Reactor

SSC – Structure, Systems, and Component

TEDE – Total Effective Dose Equivalent

TRISO – Tristructural-isotropic

Introduction

The nuclear industry is facing aging power plants approaching retirement within the next decade or so. While many plants have renewed their licenses, the decreasing price of natural gas in recent years is pushing more plants to retire since they will no longer be profitable. With increasing pressure to reduce carbon emissions world-wide in an effort to halt the detrimental effects of climate change, this stumbling block to the clean energy infrastructure would impede the progress made. Nuclear power plants provide over 11% of the world's electricity with consistent and reliable base-load power while emitting minimal greenhouse gases compared to coal plants. The United States relies on nuclear plants for almost 20% of its electricity needs. Even though new plants are being designed and utilities are applying for new licenses, they will not be able to make up for the deficit once the older plants are decommissioned. Because of this fact, there is a push to develop more economical, streamlined, and smaller nuclear reactors. These reactors would provide clean power to more diverse areas at lower costs.

Small modular nuclear reactors (SMRs) have the benefit of being able to be mass produced and distributed all over the country. The projected lower upfront capital cost for reactor modules is the primary economic incentive, making nuclear power potentially much more obtainable for governmental, industrial, and residential applications. Modularity means any number of these reactors can be installed on a single site to meet whatever electrical need the location requires. Anywhere from 1 to 12 individual modules can be located on site, depending on reactor and plant design, to provide the scalable power equivalent to large plants. Many SMR designs use the effective passive safety technology implemented in large

advanced light water reactor plants to make designs inherently safer. A lower power output, translates to a lower possible maximum fission product release. The risks of radiation exposure and ingestion in the surrounding area are projected to be smaller compared to their larger counterparts.

Despite the numerous potential benefits of these reactors, there are still obstacles to overcome both economically and policy-wise. Seeing commercial success of SMRs will depend on overall cost, both direct and indirect, as well as the efficiency of regulating and licensing these new technologies. The U.S. Nuclear Regulatory Commission (NRC) has long standing established regulations for nuclear reactors, but the hurdle facing the commission currently is how to apply these standards to the new smaller reactor designs, both traditional, and more advanced non-light water reactors (LWR). Obtaining certifications, going through the licensing process, and abiding by the established regulations requires time and capital. The longer it takes to go through these processes, the more expensive the plant will become. Many of the licensing and regulatory concerns regarding SMRs, both light water and non-light water, can be broken down into several categories: siting, safety, staffing, and design-basis considerations.

The NRC began operating in 1975 to ensure the safe use of radioactive materials for civilian applications while protecting people and the environment. Up until the Three Mile Island accident in 1979, severe nuclear accidents were considered so unlikely that the thought of one even occurring was considered to be incredible. Only afterwards were possible causes of accidents explored with the level of scrutiny the NRC uses today. Modern safety regulations are the result of the NRC's reexamination of their policies after the Three Mile Island event. When the Chernobyl accident occurred in 1986, it prompted even more focus on the types of

requirements to license new nuclear power plants. Extensive emergency planning measures were set forth during this time, including the 10 mile evacuation radius. At this point, the NRC looked to create more rigorous safety standards for the nuclear fleet (almost all LWRs) instead of licensing new plants. Many of the regulations currently in place address specifically LWRs because of these accidents.

There have been non-LWR designs that the NRC has reviewed in the past, such as the Clinch River Breeder Reactor in the early 1970s or the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor which is undergoing preliminary reviews. For the PRISM design, accident scenarios were selected using deterministic engineering judgement and risk assessment insight. Dose criterion from current regulations were met using modern source term analysis tools. Clinch River outlined accidents using a deterministic approach and used computer codes to generate transient data. In both these cases, where the regulations did not directly apply, applicants provided sufficient data, the tools they used, and information about plant systems and accident analyses for the NRC to review.

This paper will explore the various categories and the current regulations regarding each one. Much headway has already been made since the LWR SMR technology first came into public view. While there have been small SMR-type reactors proposed since the 1980s, these new designs, especially non-LWR designs, have a less developed regulatory framework. Policy options will be explored to suggest compromises between the structure in place for licensing large reactors and a new structure needed to address new SMR designs. This report will also explore if LWR regulation can be applied to non-LWR technology.

General Background

SMRs rely on nuclear fission reactions just like large LWRs do. Their smaller size, however, makes them more economical in terms of upfront capital costs for individual modules. However, since they are built to operate at a lower power than their larger counterparts, the cost per kilowatt hour of power may actually be higher than in large plants. SMRs can be coupled together in a system to support larger energy needs, depending on what is required and how many modules the plant can house. This is similar to large LWR plants that house multiple reactors at a single site. SMRs operate at less than 300 MWe compared to the larger LWRs which can generate upwards of 1500 MWe. In the near term, the designs that will see commercialization first will most likely be LWRs.

Nuclear plants generate 20% of America's electricity consumption. Unlike other forms of clean energy, they can operate for long periods of time, generating a consistent base-load power. The reactors themselves generate zero carbon emissions and no greenhouse gas emissions whatsoever, making nuclear power appealing to the Clean Energy Initiative. There are about 440 commercial reactors operating in 31 countries [1], including 99 in the U.S. [2] with four new reactors currently under construction: two at Vogtle in Georgia and two at VC Summer in South Carolina [3]. The reactors in operation and the ones currently under construction are all LWRs. Many of these plants have been in operation for decades and soon face decommissioning, which prompts the inquiry into utilizing SMRs to make up for the loss in energy generation.

Many of the SMR designs feature passive safety features. An example of these features involves using natural circulation for heat transfer and cooling pools as heat sinks for decay

heat. This enables the reactor to remove heat during scenarios such as loss of coolant accidents (LOCA) of varying degrees. Many of these systems are similar to the systems in the new large LWR design: the AP 1000. However, the smaller design means it can be constructed and fueled in one location and shipped by rail to locations across the country. Additionally, underground siting could potentially increase the number of natural barriers for intrusion, decrease the number of entry points, and decrease the impact of natural disasters such as hurricanes or tornados. Because of the lower power output, the maximum fission product generated is lower, so in the event of an accident, the size of the emergency planning zone need not be as large as those for large plants. Designers claim the smaller facilities also require less control room and security staff members.

There are some concerns to SMR technology as well. These modules are harder to inspect, because of their compact nature and integrated systems which are contained in a single structure. If there is mass production of these SMRs, there is concern that the NRC will not have adequate resources to inspect, service, or secure all potential sites for these reactors. There will either be a large demand for inspectors and staff that the NRC will have to meet or the NRC will need to change the way inspectors and staff are assigned to plants. There is also the question of whether implementation of SMRs will actually be cost effective. The cost per kilowatt-hour of electricity produced may be higher than a large reactor, assuming all other factors such as material costs, and demand are held constant. There is concern that if the costs are not economical, other aspects such as safety and security will only be designed to meet bare minimum requirements instead of exceeding established regulations.

In January 2012, the Department of Energy (DOE) supported the development of SMRs through its Licensing Technical Support Program to help expedite SMR licensing and design work [4]. In December of 2013, the DOE announced a grant to NuScale. The SMR Start Consortium was established in early 2016 to help advance the commercialization of the SMR reactor designs. In general, there has been increased interest in SMR technology as many companies have revealed new designs. The current struggle in the industry is how to make such developing designs commercially feasible.

There are currently four designs for LWR SMRs from four different companies. NuScale is scheduled for a design submission in December of 2016. Its design is a 45-MWe module that stands 65 feet tall with a 9-foot diameter. Up to 12 modules can be monitored from one control room. The reactor and containment vessel operate inside a water-filled pool. No pumps are needed to circulate water through the reactor since natural convection is utilized instead. The module is entirely passively cooled and NuScale claims it can shut down safely without outside power. A 12 module plant protected area is projected to be 42 acres [5]. Westinghouse has designed an SMR with 255 MWe output. An operating site is estimated to be 15 acres of land. Westinghouse also designed the AP 1000, and thus passive safety systems from the large LWR have been adapted for the SMR design. It also uses gravity and convection to shut down and maintain plant safety [6]. Another SMR design, developed by mPower, is capable of generating 195 MWe. Up to 10 modules can be operated in one plant. B&W states that the module can go for 14 days without outside intervention in the case of a station blackout. Plants have a projected area of 36 acres [7]. Finally, Holtec possesses a design for a 160 MWe output reactor taking up only 4.5 acres for a single module plant. Holtec claims that the passive safety features

will not release radioactivity regardless of the severity of an accident. Pumps and motors are replaced by gravity and natural circulation to run all significant plant systems. A passive containment cooling system addresses decay heat removal from the spent fuel pool. Plants can consist of up to 10 modules [8]. For size comparison, the average 1000 MW nuclear facility needs about 1.3 square miles or around 832 acres of land [9].

The SMR industry in the U.S. has not seen any recent designs that have gotten past the conceptual phase other than LWR designs. Unlike the LWR designs which use water as moderator and coolant, non-LWR designs use different materials for those roles, and operate using fuel of different enrichments. The appeal of these designs is that they can operate at much higher temperatures than LWRs and thus have higher thermodynamic efficiencies. Various designs have different safety features, use varying enrichments of fuel, and require different system designs. Some have different applications than just power production. Therefore, if common regulations are to be established for both types of reactors, they must accommodate differences in technology.

Since 2010, the question of how regulations will apply to SMRs has evolved. SECY-10-0034¹ is one of the first major documents to outline the many challenges that will need to be overcome with regulating SMRs. The issues of highest importance are SMR emergency planning, source terms, plant staffing, and physical security. Many white papers have been published by the Nuclear Energy Institute (NEI) that discuss the issues and make recommendations for potential changes to the way SMRs are regulated in these areas.

¹ SECY Papers are commission papers written by NRC staff to inform the general commission about policy, rulemaking, and adjudicatory matters

However, most of the current recommendations pertain primarily to LWRs, with the non-LWRs to be addressed at a later date. SECY-14-0095¹ was written in 2014 with a status update of the progress made through the white papers and other commission recommendations. Since then, issues that have been closed include environmental issues, risk approaches to SMR licensing, system design issues, operational programs for multi-modular designs, installation of modules during operations, and classification of structures, systems, and components (SSCs). Closed issues include those that have undergone regulatory guidance or revision as well as those where changes were not deemed necessary for specific topics. However, the issues of highest importance are still in discussion.

Safety

Accident Scenario Analysis

The beginning of any conversation dealing with the safety of a nuclear power plant is considering the possibilities of what could potentially go wrong. The core of safety review is determining which accident scenarios to analyze and the criteria for specific designs. A range of design basis accidents (DBAs) are chosen for analysis, some that potentially release radioactivity into the environment and some that potentially do not, but can still cause major problems for the plant systems. Then, once the accidents are chosen, an evaluation is performed on how the plant's safety systems deal with these scenarios. There is a minimum threshold of radiological exposure to the public that is the base criterion for assessment of these scenarios. The safety systems must meet this criterion in order for the plant to be deemed safe for operation, although the criterion depends upon the accident being analyzed. The height of safety concern is how much fission products or radionuclides are released into the environment in any case.

Accidents currently fall into two categories: DBAs and beyond design-basis accidents (BDBAs).

The former type is an accident a facility must be designed and built to withstand without loss to the SSCs necessary to ensure public health and safety. An example is a LOCA. BDBAs are accident sequences that are possible but are deemed too unlikely to warrant the extensive analysis and requirements for DBAs. This could mean accidents involving significant fuel damage or containment leakage. These accidents are analyzed to fully understand the capability of a design.

Some of the established set of accident scenarios for large LWR safety analyses can be adapted to SMR LWRs due to similarities in technology. However, some accident scenarios can be eliminated by design. For example, integral reactor designs do not utilize an external coolant pipe to pump coolant into the core. This potentially eliminates the possibility of certain types of LOCAs occurring. For non-LWR SMRs, the technology differs and the systems are not the same as in LWRs, so some accidents cannot be examined for these advanced SMRs. For example, high temperature gas reactors (HTGRs) use graphite as a moderator and gas as coolant, such as helium. The fuel for some of these designs is layered tristructural-isotropic (TRISO) particles in either prismatic blocks or pebble beds. Vendors claim the TRISO is incapable of cracking or melting, which, if the NRC accepts that claim, eliminates one particular set of accident scenario. However, other possible accident scenarios arise including large water quantities entering the vessel. This could add a large positive reactivity insertion as well as cause graphite corrosion. Loss-of-forced-circulation accidents can cause the core to heat up, causing vessel damage. Some accidents may be the same for non-LWRs and LWRs, particularly accidents which do not depend on reactor vessel designs, such as station blackouts or steam generator tube rupture.

A tool developed to help quantify likelihood of occurring accidents is probabilistic risk assessment (PRA). Through a series of models and event trees, the results of PRA include frequency of accidents that cause damage to the nuclear core, frequency of accidents that release radioactivity, and consequences in terms of injury to the public and environmental damage. Damage to the core is measured by a parameter known as core damage frequency (CDF). PRA is most effective if the systems being analyzed for accident potential are familiar and well understood. There is considerable data from operating LWR systems that help make PRA for LWRs more precise, but PRA can have large margins of error for first-of-a-kind designs. Many of the older regulations are based on deterministic models that do not quantify risk. Regulatory requirements were based on experience, test results, and expert judgement. With the development of PRA, however, the NRC has moved towards a “risk-informed” and ultimately “performance-based” approach which incorporates some level of PRA. The NRC expects to apply this approach for SMRs.

At the end of an accident scenario analysis, the established criterion must be met in order for the design to be deemed safe. For example, Title 10 of the Code of Federal Regulations (10 CFR) Part 50.46 outlines the acceptance criteria for handling LOCAs for LWRs. To make the regulatory framework more suitable for SMRs of both LWR and non-LWR design, regulations pertaining to specific accidents should be changed. New acceptance criteria should be made based on the results of PRA, whether that be CDF or likelihood of accident occurrences in specific systems of the plant. A white paper published by NEI in 2013 concerning the methodology and criteria for the technical basis of SMR EPZ gives a potential threshold for internal events. This threshold is a total mean CDF for SMR plants of $1\text{E-}5$ per year [10]. This is

an example of standard that could be used for all SMR types if all SMRs extensive PRA to obtain similar measurable parameters. This methodology in general will accommodate all types of SMRs with more flexibility than current regulations. A parameter such as this can be used to set acceptance criteria for accidents, as well as define the line between DBAs and BDBAs for each plant.

Therefore, to address different accident selections for various designs, current NRC regulations must remain technology neutral. The NRC must revise the current regulations addressing specific accidents like LOCAs and set criteria more exclusively based on PRA. DBAs and BDBAs should be defined by their likelihood being above or below the PRA parameter threshold. Lack of data for accurate PRA will be the challenge for non-LWR SMR designs, especially for first-of-a-kind plant designs. If the NRC moves forward in altering current regulations, there is still a minimum degree of uncertainty with PRAs that they should be willing to accept. It may be difficult for applicants to provide accurate PRA-based DBAs and BDBAs for non-LWR SMR designs and first-of-a-kind plants. Nevertheless, in order for accident selection regulation to be applicable for all types of SMR plant designs, these changes should be made.

Mechanistic Source Terms

For a complete analysis of safety when designing any plant, mechanistic source terms (MST) are used for analytical evaluation. These are selected radionuclides to calculate the fission product release based on DBAs. Source terms depend on a number of factors including: the inventory of fission products and other radionuclides in the core, progression of core damage, fraction of radionuclides released from the fuel, retention of coolant systems, and the performance of the containment system. Using the design based parameters, a source term can

be developed specifically to help calculate emergency planning zone and siting distances. The main risk these MSTs address is release of fission products into the environment.

The appeal of the LWR SMR designs is that they use essentially the same fuel, only in smaller quantities. MSTs previously developed for large plants can potentially be adapted to SMRs instead of developing all new source terms. The fuel, moderator, and coolant are all the same material for all LWRs. The accident selection will potentially vary as previously discussed, such as the elimination of large break LOCAs from safety analysis consideration. Other accidents like small break LOCAs, where pipes break while the reactor remains pressurized, may still apply to some SMR designs. Since the United States is familiar with the fuel and systems of LWRs, it will be easier to develop these MSTs for various LWR based designs, since many of them have already been considered.

Since the MST is design dependent, these source terms will be inherently different from design to design for non-LWRs. They utilize different enrichments of fuel coupled with different materials for both moderator and coolant. Using a MST developed for LWRs in the evaluation of a HTGR or a sodium fast reactor (SFR) would yield inaccurate results. Therefore, new MSTs must be employed based on the technical aspects of each design. The underlying question is if the designs of non-LWR SMRs are understood well enough such that MSTs can be developed for these new designs. If there is not sufficient information, a policy must be established that outlines the process of obtaining this information.

Emergency Planning Zone (EPZ)

The results of the safety analysis must meet the acceptance criteria for the established EPZ. SECY-11-0152, which focuses on developing an emergency planning and preparedness

framework for SMRs, cites 10 CFR Part 50.47 and U.S. Nuclear Regulatory Commission Regulation (NUREG) 0396 as foundations for current EPZ distances. For conventional large LWRs the EPZ plume exposure pathway is 10 miles from the reactor site and ingestion exposure pathway is 50 miles [11]. The objective for this zone is to encompass the areas where projected dose from DBAs exceed the protective action guide limits. The exact size and shape of each EPZ is a result of evaluation of specific conditions of the site, geographical features, and population information. The plume exposure pathway is based on dose received directly from fission product exposure. The ingestion exposure pathway limits the area from which food and water is consumed, which could potentially host settled out radioactive material. Based on NUREG-0396, Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, the distances of 10 miles and 50 miles were chosen based on the consequences of a fission product release tailored specifically to large LWR accidents. The designed based LOCA was used as the subject of study resulting in exposures not exceeding 25 rem to the thyroid and 5 rem to the whole body (external) at the distances given [12]. An example of an EPZ from the NRC website is shown in Figure 1.

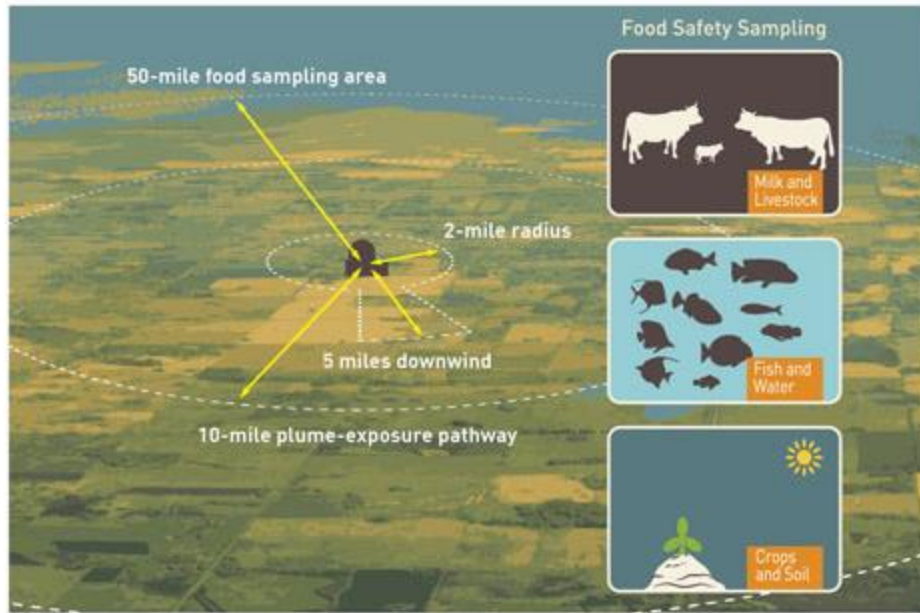


Figure 1: Emergency Planning Zone

Source: "Emergency Planning Zones." U.S. NRC. NRC, n.d. Web. 26 July 2016

For SMRs, the EPZ is proposed to be scalable. Because of the lower inventory of fission products, the release due to an accident is expected to be less severe than a large LWR. This is a siting incentive, since the plants can be located closer to more populated areas and the areas of application increase dramatically. However, there is concern over the accident implications of up to 12 of these modules being located at one plant. Under worst case scenario conditions, the entire fission product inventory of all operating modules is released into the environment. However, the probability of such a scenario occurring may be extremely low, and accident analyses are not based upon absolute worst-case scenarios. Deciding the severe accident scenario on which to base the EPZ is a concern that applies to all SMRs regardless of the technologies: LWR or non-LWR. Deciding an EPZ distance would be an additional step in the application process for the building of a plant if EPZ is indeed scalable, since the same EPZ will not be applicable to every SMR design. While 10 CFR Part 50.33 allows for EPZ to be adjusted

on a case-by-case basis for reactors with an authorized power level of less than 250 MW, it doesn't provide the scalability that the SMR applicants desire.

The NEI white paper concerning the technical basis for establishing SMR EPZs from 2013 recommends for EPZ to use an expected offsite dose of 1 rem within a two hour period of release as a benchmark, and the distance from the plant at which this dose is less than or equal to this dose will be the established EPZ distance [10]. This is the scalable alternative to using a base 10 mile exposure radius and 50 mile ingestion radius. A scalable dose-based EPZ requirement will mandate NRC to update regulations, but this approach would appeal to utilities, since the smaller EPZ incentive is preserved. The NRC may resist these changes, since erring on the conservative side is more appealing to the general public, and since the dose calculations may not be considered accurate enough to warrant an EPZ radius change.

Siting

For the application process for constructing a new plant, utilities can obtain an early site permit (ESP) before obtaining the actual license for the plant under 10 CFR Part 52. The ESP takes into consideration the location of a desired site for plant construction, approving site safety issues, environmental protection issues, and emergency planning [13]. 10 CFR Part 52.17 details the contents of this application. Under 10 CFR Part 50, the application for a construction license reviews siting concerns. For both of these processes, in terms of siting, the main debate between regulating bodies and industry is how close is too close to population centers. As stated earlier, SMRs could potentially be placed where large LWRs cannot. A lot of prospective sites are where retiring coal plants are located near large population areas.

The site surrounding a reactor includes an exclusion area boundary (EAB), low population zone (LPZ), and population center distance. These areas are also proposed to be scalable for SMRs. To determine the distances for EAB and LPZ, calculations must be done to show that the total radiation dose to an individual's whole body due to a fission product release accident will not exceed 25 rem as stated in 10 CFR Part 100.11. The EAB criterion established by Part 100.11 states that the total effective dose equivalent (TEDE) must not exceed 25 rem within a 2 hour release. The LPZ criterion is: TEDE will not exceed 25 rem for the entire duration of the release at the outer boundary of this zone. This is also established by Part 100.11. The population center distance must be at least one and one third times the distance from the reactor to the outer boundary of the LPZ. This criterion is based solely on dose calculations, and the $\frac{4}{3}$ multiplying distance will not change based on design. That being said, the distances of the zones themselves may vary depending on the dose calculations for each design. Similarly to the EPZ, the size of the EAB and LPZ will vary depending on plant design, and the smaller these areas are, the closer the plants can be located to population centers. Since the current regulations are already dose based, there will not need to be revision to the current regulations to apply them to SMR designs.

Natural hazards and man-related hazards must also be considered during the siting process. Natural hazards include meteorology, geology, and seismology. Man-related hazards are accidents associated with infrastructure such as airports, dams, and transportation routes. Proximity to natural environment such as wetlands or endangered species must also be considered. For SMRs, the requirements involving natural hazards may be different because many plant designs are sited underground and will be affected differently by earthquakes. New

advancements in earth science, earthquake engineering, and seismology may warrant a revision or creation of new requirements for the siting requirements of some SMR designs. Additionally, since some plant designs are located underground, they may be less likely to be affected by natural hazards such as hurricanes or tornados than a plant located above ground. Siting criteria taking these factors into account may need to be changed as a result as well.

Staffing

Control Room Staff

Staffing for both operation and security purposes is also a prevalent issue for SMRs. These proposed modular facilities are, as stated before, capable of housing up to 12 individual reactors depending on the vendor. For large LWRs, 10 CFR Part 54 details on-site staffing requirements, which are scalable. For example, a site with two nuclear power units operating with one control room requires two senior operators and three operators. However, for SMRs, these modules can all be controlled from one control room, creating the possibility that they need less staff than larger plants. The question becomes how many operators should be on site for plant management. Too many operators become redundant and financially unnecessary, however too few operators could result in work overload and human input impairment leading to further problems.

Operator tasks may change depending on the level of automation and required input for various designs. The current regulation in 10 CFR Part 50.34 requires a human factors engineering program to determine the tasks and responsibilities of control room staff for nuclear reactors. NUREG-0711, Human Factors Engineering Program Review Model (HFE)

Revision 3 [14] details the methodology to review such a program. HFE was revised in September 2012, making the objectives of the program up to date, but whether or not the methodology from NUREG-0711 can be applied to SMR technology is a concern. Currently, HFE programs are submitted with applications along with exemptions if the staffing number is different from regulatory requirements. Table 1 depicts the current operator requirements for power units.

Number of nuclear power units operating ²	Position	One Unit	Two units		Three units	
		One control room	One control room	Two control rooms	Two control rooms	Three control rooms
None	Senior Operator	1	1	1	1	1
	Operator	1	2	2	3	3
One	Senior Operator	2	2	2	2	2
	Operator	2	3	3	4	4
Two	Senior Operator	--	2	3	3 ³	3
	Operator	--	3	4	5 ³	5
Three	Senior Operator	--	--	--	3	4
	Operator	--	--	--	5	6

Table 1: Minimum Requirements per Shift for On-Site Staffing of Nuclear Power Units by Operators and Senior Operators Licensed Under 10 CFR Part 55

Source: "Control Room Staffing for Small Reactors." Tech. Washington D.C.: NEI, 2011. Print.

If the guidance from NUREG-0711 can be used to evaluate HFE programs for the different SMR designs, then the HFE programs can be reviewed on a case-by-case basis by the NRC. Applicants can submit formal exemptions in the design certification (DC) stage of the application process. Data and tests included in the individual HFE programs can be generated

from plant reference simulators, which provides insight into the operational tasks required for the specific control rooms. Training simulators can serve multiple functions: a mechanism to train future operators, and a mechanism to collect information needed for HFE analyses. Additionally, positions such as shift supervisor and shift technical advisors can undergo a separate HFE analyses while operators are being trained. This may require extensive work on the applicant's part, but a case-by-case HFE review may be necessary because standardized control room requirements like in current regulations cannot be used. The differences in technology, operator tasks, level of automation, and lack of operational experience for these advanced SMR designs are all reasons why standardized requirements cannot be implemented.

Security Staff

Security staff is another concern for SMR plants. 10 CFR Part 73 details requirements to protect against radiological sabotage and prevent theft or diversion of special nuclear material. An NEI White Paper was published in 2015 to address a consequence-based physical security framework for SMRs and other new technologies. It states elements of the physical security program include having physical barriers, identifying target sets, and staffing a security organization to assess, interdict, and neutralize threats [15]. The minimum number of armed responders for large plants is 10 as stated in 10 CFR Part 73.55 (k)(ii). However, new innovations for SMR facilities may reduce the number of armed responders needed. Some SMR plants feature designs that inhibit severe core damage, making sabotage and external attacks less serious threats, and reduce the potential for off-site radiation releases. SMR plant sites would also be much smaller than large LWR plant sites, which would reduce the need for security staff, since there would be less area that needs to be secured. Another major security feature is

that because many plants will locate their modules underground, the number of access points is minimized and the natural physical barriers that inhibit intrusion increase. Not only are severe accidents and radioactivity releases major security concerns, but having the plant shutdown in the middle of operation may be highly undesirable for electrical grids, plant systems, utilities, or the refueling schedule of the modules. While the safety features could double as security measures, there is still a necessity for armed response. If the minimum number of armed responders were to be reduced or made flexible, it would provide an incentive to the SMR utilities to construct these plants. However, there would need to be a revision in current regulation by the NRC.

For LWR and non-LWR SMRs, the security staffing numbers could be determined using the HFE program, but unlike control room staffing requirements, need not undergo extensive review on a case-by-case basis. Alternately, these numbers could be based on plant characteristics such as size of the plant, number of modules, number of vital components, total power generated, or number of access points. The factor should be chosen based on protecting the most important parts of the plant. The requirements could be broken down into a table similar to the current control room staffing requirements for large LWRs. Scaling the minimum number of responders in this way is appealing for both the NRC and applicants, since the incentive of lower security staffing is preserved while the reviewing process is expedited.

Nonproliferation is another security concern for the nuclear industry. SMR plants have a smaller quantity of fuel than larger plants, which is one of the advantages from a nonproliferation perspective. Depending on the design, spent or fresh fuel may not be located on site, and thus there is no central target for nuclear material theft. The areas of the plants

themselves are smaller, presenting smaller targets for proliferation. For various SMR plants, those with 12 modules present more of a risk than for plants with only 3, which is another factor contributing to the idea that the minimum number of armed responders should not be a concrete number. There are many technical aspects and policy issues regarding nonproliferation, but they are beyond the scope of this research paper.

Design-Based Considerations

Modular Licensing

A recommended approach outlined in SECY-11-0079, which addresses the license structure for multi-module facilities related to SMRs [16], describes three options for handling the licensing of multi module facilities. The first option is to grant a single facility license to the plant. This is, administration-wise, the easiest option since it addresses the plant as a whole and acknowledges the standardization of modules. It does not, however, deal with module specific operating problems. This approach may reduce the lifetimes of specific modules as well, since not all modules will begin operation when the facility begins operation. The second approach is to grant a master facility license and individual module licenses. This approach addresses the possible need for sub-licenses prevalent in the first approach. The master facility license would not include the authorization to operate a nuclear reactor, and so is valid for the entire lifetime of the plant. This master license would instead address the common SSCs, performance-based criteria, and requirements for the overall plant. The problem with this approach is that the NRC would need new regulations defining how the master facility license would fit within the existing technical and legal requirements for licenses. The final option is to issue individual module licenses, which can be processed with a single application and can be applied to

multiple modules. This approach preserves individual module lifetimes. The Atomic Energy Act of 1954 (AEA) allows for licenses to single facilities and individual modules.

If licenses for individual modules are pursued, there are two options in addressing the common SSCs for each module. One option is to attach the common SSCs of subsequent modules to the license of the first module. This helps support the initial licensing process, and complications would not arise until the end of the license term of the first module. Then, there is question about the decommissioning process since the SSCs of all modules would be attached to the first. The second option is to define license conditions for common SSCs in a license appendix. The license appendix would establish performance-based criteria and common standards for all modules. Instead of holding all modules to the license conditions of the first, this approach supports independence in licensing and decommissioning while still holding all modules to a common standard. The challenge would be developing the performance-based criteria initially.

Application Process

There are two processes an applicant can go through to obtain the license to build a nuclear power plant: 10 CFR Part 50 and 10 CFR Part 52. Under 10 CFR Part 50, an application for a nuclear plant calls for an evaluation to obtain a construction permit, then an operating license for the plant. The construction permit will take into consideration the preliminary safety analyses of the prospective plant design to be reviewed by the NRC. After construction begins, an application for an operating license must be submitted, accompanied by a Final Safety Report (FSAR). The construction license grants applicants permission to build the plant and the

operating license grants permission to load fuel and operate the reactors. Under Part 50, design changes can be made during the review process by revising the application.

The licensing process described in 10 CFR Part 52 combines the two separate licenses from Part 50 into a single license called a combined construction and operating license (COL). The information submitted must meet the technical requirements set forth by Part 50. Prior to the COL application, applicants have the option to apply for an ESP which can include a specific plant design or may include an envelope of potential reactor designs that meet the desired site parameters. The ESP determines ahead of time whether or not a site is eligible for construction of a power plant. It can be referenced in a COL application, reducing the time spent reviewing the application. A DC may also be referenced in the COL, which “pre-qualifies” a plant design. The DC is broken up into multiple tiers of information for submission. Tier 1 contains plant specific design parameters pertaining directly to safety of the plant. This includes, but is not limited to design descriptions, significant site parameters, and interface requirements. This tier involves a federal rule making process. Tier 2 covers all other necessary information to meet the requirements as set forth by Part 50. Tier 2* is information not included in Tier 1 that is not part of the federal rule, but still requires NRC permission to alter. At the end of construction, the facility must validate the plant by making sure it meets the inspection, tests, analyses, and acceptance criteria (ITAAC). The results are submitted to the NRC so it can be verified that the plant operates in such a manner that it has been licensed to. This process is deemed more streamlined, especially for familiar systems and technologies, but if changes need to be made in Tier 1 of the DC, it requires a federal rule making, public comment, and NRC permission. Too

many changes lead to hang ups and delays, not to mention more time and money spent during the entire process of getting a plant up and running.

There are pros and cons to going through either process. Part 52 was designed to streamline the lengthy application process for familiar designs and well known technology. The obstacles new designs would run into would be in the design certification application (DCA), where certain changes to specifications would draw out the process. Especially for non-LWR SMRs, there is greater potential for alterations necessary simply because of a lack of hard data or because more in-depth analyses are needed for unfamiliar technologies. Because the operating license is obtained before construction of the plant is even finished, any changes to the DCA would need extensive review by both the public and the NRC before the design is given permission to change. This draws out the process and could potentially cost the applicant a great deal more than with the Part 50 process. No design has ever gone through the Part 52 process completely. However, under the Part 50 process, it is possible that the applicant obtains a construction license and builds a plant, but does not meet the requirements to obtain an operating license. Then, the applicant is left with a large investment with no return. The Part 52 process could, in fact, be more efficient than the Part 50 process if no federal rulemaking changes are required. These are some of the costs and benefits of each process.

In order to improve regulatory efficiency of SMR designs through Part 52, a phased submission of the DCA could be implemented. A phased DC submission would entail Tier 2 be submitted first for NRC review, then Tier 1. This would allow sufficient time for changes and revisions to be made before the submission of Tier 1 and ITAAC. This approach still meets the regulatory requirements of technical sufficiency and completeness. Formal NRC policy changes

will not be necessary to implement the phased DCA submittal alternative. This new approach is designed to ease the review process for both NRC staff and for applicants in a timely manner.

Recommendations

Safety

To move regulations towards a more technology-neutral methodology for selecting accidents, PRA based acceptance criteria is suggested to be relied on more heavily than in current regulations. Regulations pertaining to specific designs like LWR LOCAs should not be applied for all designs. The NRC should establish a criterion parameter calculated using PRA such as CDF to separate DBAs and BDBAs from the spectrum of considered accidents for the purposes of establishing EPZ and siting distances. Applicants should be required to submit PRA for accidents with a minimum degree of uncertainty in order for NRC to review the designs. This will place less of a burden on the NRC to fit technology-dependent regulations to designs that may not use the same technology. Applicants will likely need to consult external expertise to develop accurate data for non-LWR and first-of-a-kind plant designs such as national labs for data or testing. This approach will require applicants to invest more time and resources up front to provide the PRA, but once the technologies of various reactor designs are better understood, it will be an easier process overall.

In developing MSTs, the applicant is responsible for understanding the systems of the design and for providing data sufficient enough for a DBA source term. Similar to the development of PRA, external expertise will likely need to be consulted for the development of new MSTs. Source terms should be developed for the more probable accident scenarios as

determined by PRA results. This heavy investment into the research of the potential fission product releases and other accident scenarios will pertain mostly to non-LWR and first-of-a-kind plant designs. For LWR SMR plant designs, it is recommended the applicants use a hybrid of current methodology from large LWR plants (such as evaluating small break LOCAs) and new MSTs for DBAs when undergoing source term analyses. This will take new design features into account, while still using an approach to MST analysis familiar to industry and the NRC.

The EPZ should be plant based as opposed to module based. For example, a facility licensed to operate 6 modules should establish an EPZ based on the 6 modules, even if there are only 3 installed at a certain point in time. The radius should be based on the off-site doses of DBAs determined from the plant PRA. This is assuming the applicant can provide sufficient dose calculations from MSTs and precise PRAs for more severe DBAs. If the applicant cannot supply such data, the NRC can require the 10 mile/50 mile radii. The plume exposure and ingestion exposure pathways should not be fixed to a specific value in order to preserve the incentive of SMRs having the potential to be located closer to population areas. The NRC should determine a maximum off-site dose, most likely based on health physics evaluations, and use this off-site dose to establish EPZ instead of having a constant distance like for large LWRs. This will require the NRC add more detail in methodology of regulations pertaining to case-by-case emergency planning like 10 CFR Part 50.47.

Siting

Siting criteria can be scaled down for SMRs similarly to EPZ. In determining these radii, the DBAs and BDBAs based on the PRA calculated threshold should be considered. The dose quantity may be subject for revision with more detailed analyses in the future, but the current

reactor regulations, like 10 CFR Part 50.34 and Part 100.11, can remain in the near term. This will require no regulatory alterations by the NRC. This keeps the incentives for SMRs. Assuming applicants can prove, through the most probable DBA MST analyses, that these criteria are met, the EAB and LPZ have the potential to be the same distance. Therefore, it is recommended that the NRC regulation siting criteria for LPZ and EAB not be changed, since the dose limits are already established, but the accident scenario analyses used for these dose calculations follow the recommendations stated earlier in this paper.

Siting considerations such as meteorology should not be much different than for large plants. Effects of construction and meteorology on potential fission product release accidents still follow the same methodology set forth by current regulations. The NRC should make changes in acceptance criteria regarding geologic and seismic data for SMR plants located below ground. Evaluations of these plants should use external expertise like national labs or experts in earthquake engineering. These evaluations should take into account the location of emergency systems of the plant: if they are above ground or below ground. In conclusion, the only regulation changes the NRC should make would be regarding geologic and seismic acceptance criteria for underground SMR plant sites.

Staffing

It is recommended to avoid a common standard in control staffing regulation for SMRs. However, because of lack of operational experience with advanced SMR control room designs, it will not be feasible to develop new requirements without operational SMR plants to base them off. Instead, the NRC should keep current control room staffing regulations in place, but prepare to review exemptions from applicants on a case-by-case basis. Applicants should

commit to developing a training simulator for control room staff before the application is submitted for NRC review. These recommendations do not require any regulatory changes, but require increased attention to understanding the HFE objectives set forth by NUREG-0711, which provides a reasonable basis for HFE implementation for SMRs as stated in the 2011 NEI White Paper: Control Room Staffing for Small Reactors [17].

The NRC should change security staffing regulation from a concrete minimum number to one that is flexible. The main factor of the security staffing assignment should be number of modules, independent of how much power they produce. The other important factor should be number of entry points. It is recommended the NRC adjust current regulations such as 10 CFR Part 73 to require security staffing on a scalable basis based on number of modules and number of entry points, similar to the control staff requirements in Table 1. These changes will benefit both the SMR industry and the NRC.

Design-Basis Considerations

It is recommended that licensing for SMRs be done using the individual license approach described previously. This method allows for the preservation of individual module lifetimes, and does not require extensive changes to the licensing process. A license appendix should be created to address SSC requirements. These recommendations are based on the potential benefits of maintaining pseudo-independent licenses for each module without needing to go through the entire process multiple times. Other benefits of this approach include preservation of individual operating lifetimes and standardized performance-based criteria for common SSCs. The NRC should incorporate this new licensing process into 10 CFR Part 50 and 10 CFR Part 52 for multi module facilities.

For the application process, it is recommended that the NRC do one of two things. The first is to require all first-of-a-kind SMR designs go through the Part 50 process instead of the Part 52. In this way, the applicant can make changes to the designs easier, and the NRC can gain considerable experience for licensing that type of plant. Applicants will most likely resist this action because if they feel their designs are fully developed and that they understand the Part 52 process well enough, they will want to apply using the Part 52 process. The NRC benefits from this option since it has more experience reviewing designs through the Part 50 process. The second option is to alter the timeline of Part 52 to incorporate a phased DC submission. This approach may slightly increase the timeline for applicants to submit the required material. This will not require significant policy changes on behalf of the NRC and it could potentially spread the NRC's work load out over a set period of time. Applicants will still be allowed to choose either Part 50 or Part 52. However, there is no sure way to tell this phased submission will indeed expedite the licensing process, since no application has fully completed Part 52 before. In order to maximize benefits to both the NRC and to applicants, it is recommended the NRC and applicants utilize a phased submittal process regarding the DCA in Part 52.

Conclusions

The many policy issues addressed are obstacles in the progress of commercializing SMR technology as stated in SECY-10-0034 and SECY-14-0095. These recommendations are made to potentially introduce the thought process in pursuing the step forward in determining policy for SMRs. While the most likely candidates for the first commercial SMR plants will be LWR designs, it is important to keep these advanced non-LWR designs in mind. These policy

recommendations are made to make the regulatory process more efficient for SMRs without detracting from the extremely thorough NRC procedures.

For safety regulation, accident safety analyses should be done mainly for DBAs and BDBAs determined through PRA, where applicants will consult national labs for assistance in these analyses. MSTs for LWRs should use a hybrid methodology of current MSTs while developing new MSTs to account for plant differences from large LWRs. For non-LWRs, applicants should look to national labs for assistance in developing new source terms specific to the PRA based DBAs and BDBAs. EPZs should be established for the plant instead of individual modules. A dose criteria for a severe DBA release should be made to establish the distance of this zone. Siting distance requirements should follow established regulation, but for the probable DBAs resulting from PRA. New acceptance criteria should be developed for underground SMR plant sites taking into account current advances in geologic and seismic studies. Control room staffing requirements for SMR plants will not require changes in regulation, but should be considered on a case-by-case basis examining individual submissions of HFE. Security staffing regulation should be changed to be scalable based on number of entry points and number of reactor modules. Individual licenses should be issued to each reactor module, each referencing a license appendix which contains acceptance criteria for common SSCs. Finally, a phased DCA process should be implemented in Part 52 of the application process, which will not require changes in regulations.

These recommendations could potentially lead to the expedited licensing and regulatory processes needed for the SMR industry. While these recommendation positions are not technologically heavy, they may lead to more sophisticated analyses on how to better improve

the NRC infrastructure regarding LWR and non-LWR SMR technology. As more and more large nuclear plants are being decommissioned, the need for this type of development has never been higher.

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