Risk-Informed and Performance-Based

Human-System Considerations

for Advanced Reactors

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Executive Summary

Advanced nuclear reactor technologies present a combination of new opportunities and important topics for discussion for both the U.S. Nuclear Regulatory Commission (NRC) staff and the nuclear industry. The NRC staff anticipates that reactor developers will propose novel approaches to reactor operations. These operations-related areas include staffing, training, qualifications, human factors engineering, and an increased reliance on autonomous operations for safety significant key systems, structures, and components (SSCs). As such, the NRC staff is developing a risk-informed, performance-based, and technology-inclusive regulatory framework to appropriately consider the role of humans and human-system integration in ensuring safe nuclear operations within the context of advanced reactors.

This paper proposes methods by which the proposed Title 10 of the *Code of Federal Regulations* (10 CFR) Part 53, in conjunction with relevant regulatory guidance documents, can provide a predictable and technology-inclusive framework for the regulation of the human-system aspects of advanced reactor operations.[1] Specifically, the primary objective of this white paper will be to inform the Part 53 rulemaking by proposing operations-related content for inclusion in the rule.[2] A secondary objective will be to facilitate the consistent treatment of operations-related content of advanced reactor applications that are received prior to the Part 53 rule being finalized.

Disclaimer

The reader is advised that not all the references cited within this paper have been endorsed by the U.S. Nuclear Regulatory Commission (NRC). For example, the referenced report by the International Atomic Energy Agency on "Non-baseload Operation in Nuclear Power Plants: Load Following and Frequency Control Modes of Flexible Operation" and the Nuclear Energy Institute's white paper on "Micro-Reactor Regulatory Issues" should not be interpreted as containing NRC endorsed positions. References such as these are discussed within this paper for informational purposes. Similarly, the included discussion regarding SLOWPOKE reactors is not associated with any official interactions with the government of Canada. Rather, the material related to the SLOWPOKE and SLOWPOKE-2 designs is publicly available and is provided solely for informational purposes.

[1] Current applicants under Title 10 of the Code of Federal Regulations (10 CFR) Parts 50 and 52 have access to a regulatory exemption process to address certain topics described within this paper. However, the 10 CFR Part 53 rulemaking presents an opportunity to reduce such reliance on exemptions. While this paper primarily serves to inform the regulatory framework for licensing under Part 53, a secondary function is to inform proposed exemptions from Part 50 and 52 requirements for near-term applicants.

[2] Throughout this paper, frequent reference is made to informing the content of the Part 53 rule. However, it should also be understood that this objective is, by extension, also inclusive of informing the Part 53 statements of consideration, as well as the associated regulatory guidance documents.

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Introduction

Advanced nuclear reactor technologies present a combination of new opportunities and considerations for both the U.S. Nuclear Regulatory Commission (NRC) staff and the nuclear industry. In addressing this changing technological environment, a joint Division of Reactor Oversight (DRO) and Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) working group was formed in order to consider the regulatory implications of novel approaches to reactor operations that are envisioned for advanced reactor designs. The primary objective of this working group is to support development of the "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors" that is associated with the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 53 rulemaking. For clarity, the term *advanced reactor* is defined within the scope of the proposed Part 53 rule (and thus of this paper) and is based on the definition in Nuclear Energy Innovation and Modernization Act (NEIMA, Pub. L. 115-439).[3] As described in SECY-20-0032, "Rulemaking Plan on "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN-3150-AK31; NRC-2019-0062)," the NRC staff considers an advanced reactor to include:

- all designs that are not light-water reactors (LWRs), regardless of size;
- all small modular reactor designs;
- microreactors; and
- fusion reactor systems (per the Staff Requirements Memorandum (SRM) to SECY-20-0032).

The NRC staff stated its interpretation of NEIMA's definition of an advanced nuclear reactor in SECY-20-0032. NEIMA states that such a reactor will have "significant improvements compared to commercial nuclear reactors under construction" as of January 14, 2019. Therefore, "Generation III+" designs are excluded from the definition because the AP1000 reactors were under construction at the time of NEIMA's enactment.

<u>For simplicity, this paper will use the term "advanced reactors" to refer to this wide variety of reactor technologies in a collective sense.</u>

The Part 53 working group came to an early conclusion that a framework needed to be established that could address the diverse and novel operational characteristics and considerations associated with advanced reactor facilities. These operational characteristics and considerations include automation of operations (e.g., reactivity control), staffing and qualifications of operations personnel, evolution in control room concepts, and the application of human factors engineering (HFE). In particular, the NRC staff recognizes that the concept of operations for an advanced reactor facility would likely be substantially different from operations generally associated with existing technologies. According to NUREG-0711, "Human Factors Engineering Program Review Model," a concept of operations ("ConOps") "…defines the goals and expectations for the new system from the perspective of users and other stakeholders and

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[3] It should be noted that the Energy Act of 2020 (Division Z of the Consolidated Appropriations Act, 2021) included a different definition of "advanced nuclear reactor." However, that legislation does not affect or modify the definition enacted in NEIMA. Thus, this paper utilizes the NEIMA definition.

defines the high-level considerations to address as the detailed design evolves" (NRC, 2012a, p. 113). From a personnel and human factors engineering standpoint, the concept of operations encompasses roles and responsibilities, staffing, qualifications, training, and the management of operations under various conditions.

In order to provide a more predictable framework for advanced reactor licensing, the NRC staff is developing a regulatory framework that can be applied to a wide range of advanced reactor technologies to facilitate sound conclusions regarding whether a given concept of operations provides, at minimum, reasonable assurance of adequate protection.. In a related manner, advanced reactor stakeholders need a regulatory process that is predictable and that allows for designers to make informed decisions, including those associated with the concept of operations for a proposed design. Therefore, a risk-informed, performance-based, and technology-inclusive regulatory framework is warranted that appropriately considers the role of humans and human-system integration in ensuring safe nuclear operations for advanced reactors. For this reason, this paper provides a discussion to supplement concepts described in preliminary draft Part 53, subpart F, "Operations," and provides a technology-inclusive framework for the licensing and oversight of human-system aspects of advanced reactor operations.

Problem Statement

The NRC staff is conducting near-term licensing reviews for advanced reactors under the current regulations in 10 CFR Parts 50 and 52. However, the existing regulatory framework for power reactors is primarily aligned with operation of large LWR designs. The regulatory framework for advanced reactors should be capable of addressing novel operational concepts for a wide variety of advanced reactor technologies and should reduce the need for exemptions from current regulatory requirements. Furthermore, given that some advanced reactor designs may present very low radiological risk, some requirements in the current regulatory framework for operation of large LWRs may be unnecessary for reasonable assurance of safety. For these reasons, the staff is developing a risk-informed, performance-based, and technology-inclusive regulatory framework that appropriately considers the role of humans and human-system integration in ensuring safe nuclear operations and addresses areas such as staffing, training, qualifications, HFE, and considerations associated with autonomous operations.

New Reactor Technologies and Safety Characteristics

The nuclear industry in the United States has been host to considerable advances in reactor technologies over the preceding decades. These advancements have evolved out of the combined efforts of both the U.S. Department of Energy (DOE) and private stakeholders. Such new technologies have included evolutionary changes in areas like passive safety and modular construction. Examples of advanced reactor technologies that are under various stages of development include small modular reactors (SMRs), non-LWRs, and fusion-based technologies. In the case of non-LWR technologies, certain concepts that were originally introduced decades ago are now being revisited in an updated fashion. Such new technologies warrant the careful consideration of unique design attributes that represent departures from the large LWR designs that have long dominated the domestic nuclear power landscape. For these reasons, the NRC staff is developing new regulatory approaches that take the changes in nuclear technology into account.

In 2008, the NRC issued a final Policy Statement on the Regulation of Advanced Reactors. This policy statement reinforced the Commission's expectations regarding advanced reactors and included new items to be considered during the design of these reactors. Key excerpts from this policy statement include the following:

> …the Commission expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.
>
> Among the attributes that could assist in establishing the acceptability or licensability of a proposed advanced reactor design, and therefore should be considered in advanced designs, are:

- Highly reliable and less complex shutdown and decay heat removal systems. The use of inherent or passive means to accomplish this objective is encouraged (negative temperature coefficient, natural circulation, etc.)…
- Simplified safety systems that, where possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe shutdown conditions. Such simplified systems should facilitate operator comprehension, reliable system function, and more straightforward engineering analysis.
- Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems, with an emphasis on minimizing the potential for accidents over minimizing the consequences of such accidents…
- Designs that incorporate the defense-in-depth philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for, and consequences of, severe accidents. (73 FR 60612; October 14, 2008)

As highlighted by the preceding excerpt, the NRC recognizes the desirability of attributes such as simplified safety features of a passive or inherent nature, reductions in required human actions, incorporation of defense-in-depth, and minimization of the risks associated with severe accidents in advanced reactor designs. Accordingly, these and related attributes, and the implications for licensing reactors that have them, will be addressed in greater detail.

Smaller Source Term Sizes and Reduced Accident Consequences

Advanced reactors could vary in size from very large to very small; such variations are expected to have potential implications for both source term sizes and accident consequences. Building upon earlier pre-application activities and work by the NRC, the NRC staff used SECY-10-0034, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs," to discuss a number of issues related to SMR policy and licensing. A key consideration associated with SMR development identified in SECY-10-0034 was that of accident source terms (NRC, 2010). In discussing this area in SECY-10-0034, the staff observed the following:

> Accident source terms are used for the assessment of the effectiveness of the containment and plant mitigation features, site suitability, and emergency planning. Other radiological source terms are used to show compliance with regulations on dose to workers and the public. Design and license applicants and the NRC will need to establish appropriate bounding source terms for high-temperature gas-cooled reactors and other SMRs. (NRC, 2010, p. 4)

It should be noted that the SMR designs under consideration at that time ranged from 10 MWe to over 300 MWe (NRC, 2010, Enc. 1, p. 1).

In 2019, the Nuclear Energy Institute (NEI) published a white paper on "Micro-Reactor Regulatory Issues." As part of this paper, NEI included the observation that "…a performance-based, consequence-oriented regulatory framework is needed for micro-reactors, because the existing regulatory framework for power reactors does not appropriately consider the extremely low potential consequences of accidents that are expected for these designs" (NEI, 2019a, p. 1).

SECY-20-0093, "Policy and Licensing Considerations Related to Micro-Reactors," was developed to inform the Commission on certain licensing topics related to microreactors that may necessitate future departures from current regulations, related regulatory guidance, and past precedent. This SECY identified potential policy issues related to licensing microreactors and described the NRC staff's approach to facilitating licensing submittals for the deployment and operation of microreactors (NRC, 2020d). Highlighting the potential for reduced radiological consequences to result from accidents at microreactor facilities, the staff observed in the SECY paper that:

> Although no regulatory definition has been established, micro-reactors are small (on the order of tens of megawatts thermal (MWt)), have simpler designs with inherent safety features, and, in the unlikely event of an accident, are anticipated

> to have lower potential radiological consequences with a correspondingly lower impact on public health and safety. (NRC, 2020d, pp. 1-2)

Smaller source term sizes and expected reduced accident likelihood and consequences serve to directly limit the size of the hazard posed by a reactor facility. Ultimately, limiting the relative size of the hazard posed by a reactor facility is the most reliable means of ensuring safety because of this design aspect's ability to remain effective across a wide range of possible events (e.g., internal and external initiating events, equipment failures, design deficiencies, maintenance errors, or hostile actions).

<u>From these discussed considerations, the following key points should be noted:</u>

- Accident source terms serve as a measure of the efficacy of mitigation features.
- Advanced reactor designs may present low potential accident consequences.
- Limiting the hazard posed by a reactor facility reduces the potential for accident consequences and is the most reliable means of ensuring safety.

Passive Safety Features and Inherent Safety Characteristics

Sandia National Laboratory's 2020 report on "Human Factors Considerations for Automating Microreactors" evaluated specific issues that were envisioned to stem from potential changes in the role of personnel at microreactor facilities. This was done to inform the NRC on microreactor-related licensing topics that could warrant departures from existing regulations and guidance. This report noted that microreactor design elements related to safety features and low levels of nuclear material may result in such facilities looking to automate more functions than existing nuclear power plants. In particular, passive safety features and inherent safety characteristics were anticipated to substantially influence the design and operations of microreactor facilities. Specific insights provided on these types of safety features included the following:

> With a passive safety system, a microreactor design would not require significant human or automation intervention to maintain a safe state. Thus, human and/or automated tasks may serve as a secondary safety check rather than a primary function to operate the reactor. (SNL, 2020, p. 10)
>
> …the typical causes of failure for active systems generally do not exist for a passive system—i.e., loss of power or failure of operator action. By contrast, passive systems can fail as a result of modes such as mechanical or structural failure of an SSC, or even malicious human intervention. (SNL, 2020, p. 30)
>
> In contrast to passive safety, inherently safe systems are those which are absolutely reliable. The classification of absolute reliability must be qualified by a detailed consideration of the range of characteristics of the SSC that support the safety function. For example, control of reactivity often involves reactivity feedback mechanisms inherent to a system preventing reactivity excursions from occurring (e.g., moderator temperature feedback). (SNL, 2020, p. 31)

The NRC staff has previously also recognized similar potential implications of these inherent safety characteristics for certain advanced reactors, noting in SECY-20-0093 that:

> Micro-reactors are smaller and generally simpler than the SMRs the NRC has reviewed recently, and they may be able to rely on inherent characteristics for safety functions. This degree of simplicity and inherent safety may result in few to no operator actions being credited for maintaining plant safety. (NRC, 2020d, Enc. 1, p. 5)

Design attributes such as inherent safety characteristics, passive safety features, and automated safety systems work in different ways to support safety. These attributes can be considered in terms of a progression that influences the concept of operations that will be implemented for a given facility. The following outline summarizes these attributes along with specific considerations that have relevance to the concept of operations:

- <u>Inherent safety characteristics</u>: these characteristics rely upon the intrinsic attributes of a hazard in order to limit the behavior of that hazard. Inherent safety within a design can limit undesired departures from safe operation.
- <u>Passive safety features</u>: these systems do not rely upon external inputs to achieve desired safety functions and rely instead upon laws of nature, material properties, and stored energy; within such a context, human beings and automation tend to take on a secondary safety check function (SNL, 2020, p. 10). However, it must be noted that passive safety features remain vulnerable to human errors (e.g., incorrect system alignments, degradation, or other failure mechanisms).
- <u>Automated safety systems</u>: these systems rely upon active safety systems. Automated safety systems reduce the potential for human error but are still vulnerable to it. Furthermore, automated safety systems are also vulnerable to design deficiencies, dependent on external power, and unable to directly limit the behavior of a hazard.
- <u>Manual operator actions</u>: these operations rely upon human intervention to ensure that safety systems can perform their credited functions. Manual operator actions are subject to human error and limitations.

It is important to note that the further a given applicant needs to go down this list of design attributes in the course of making a safety case (such as by relying upon active safety systems in lieu of inherent safety characteristics), the greater the degree of emphasis that the NRC staff will likely need to place on the HFE aspects of the application.

Among the concepts described above, those associated with inherent safety are especially noteworthy within the context of this paper, and clearly articulated definitions in this area is of particular importance. The International Atomic Energy Agency (IAEA) observed in a 1991 publication that "[s]afety related terms such as passive and inherent safety have been widely used, particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other" (p. 7). In the course of addressing this, the IAEA (1991) went on to provide the following description:

> An inherent safety characteristic is a fundamental property of a design concept that results from the basic choices in the materials used or in other aspects of the design which assures that a particular potential hazard can not become a safety concern in any way. (p. 10)

Building upon this discussion, the IAEA (1991) went on to define "inherent safety characteristics" as being "safety achieved by the elimination of a specified hazard by means of the choice of material and design concept" (p. 13). Going forward, it will be important for both the NRC staff and industry to establish and maintain a common understanding of what constitutes inherent safety characteristics within advanced reactor designs.

<u>The following are key points that should be taken away regarding passive safety features and inherent safety characteristics:</u>

- Passive safety features and inherent safety characteristics can influence the role of personnel at advanced reactors facilities.
- Passive safety features tend to place humans into a defense-in-depth role.
- While passive safety features can still fail under certain conditions, inherent safety characteristics can be considered to be absolutely reliable[4].
- The incorporation of inherent safety characteristics, passive safety features, and automated safety systems influence the concept of operations and can affect the emphasis placed on the HFE aspects of an application review.

In summary, the NRC staff recognizes the desirability of attributes such as reduced accident consequences, passive safety features, and inherent safety characteristics in advanced reactor designs. The NRC staff also recognizes that the specific safety characteristics and features of an advanced reactor can influence both the selected concept of operation for the design and the depth and scope of the HFE review for an application. The next section of this paper considers aspects of these concepts of operations in greater detail.

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[4] In their 2020 report on "Human Factors Considerations for Automating Microreactors," Sandia National Laboratory made the following observations regarding the need to carefully consider the "absolute" nature of inherent safety characteristic reliability:

> The classification of absolute reliability must be qualified by a detailed consideration of the range of characteristics of the SSC that support the safety function. For example, control of reactivity often involves reactivity feedback mechanisms inherent to a system preventing reactivity excursions from occurring (e.g., moderator temperature feedback). In this case, it is generally difficult to postulate an external perturbation that would give rise to a loss of reactivity control. However, for cooling or containment functions, it is more likely that passive systems can exhibit failures under a range of external perturbations such that they are not absolutely reliable. Under some circumstances, however, even cooling functions may be ultimately reliable should the power level of the reactor be sufficiently low that residual heat can always be rejected to the atmosphere. (p. 31)

New Concepts of Operation

The emergence of advanced reactor designs and technologies may involve changes that allow consideration of significant departures from past operational practices within the U.S. The areas that will be highlighted in this section are multi-module operation, automated reactivity manipulations, autonomous reactor operations, and load-following. However, before proceeding, it should be noted that there is a considerable degree of overlap between these areas and, therefore, it is useful to consider the operational features that will be discussed here within the framework of the greater whole of a given design.

Multi-Module Operation

Some SMRs are designed for deployment and operation of multiple reactor modules as part of a larger power reactor facility. A primary consideration associated with multi-module operation is how the design and staffing allow for the control of multiple modules from a common control room. The NRC staff, in SECY-10-0034, previously identified that this issue, as well as how to address it, would necessitate further evaluation:

> During pre-application discussions with the NRC staff, SMR designers have indicated that they are evaluating whether the function and task analyses for normal operation and accident management conducted for their SMR designs support control of more than two modules from one control room and support operation with a staffing complement that is less than that currently required by the Commission's regulations. The NRC staff believes that resolution of this issue is required to support the design development, and the staff's review, of design and license applications for most of the SMR designs… (NRC, 2010, p. 13)

The NRC staff's most recent experience in navigating such considerations has come by means of the licensing efforts associated with the NuScale design certification process. In that example, the NRC staff reviewed a staffing plan supporting the operation of up to twelve modules. Thus, this area is one in which the NRC staff has recently gained experience. In particular, recent insights gained from reviewing the results of task analyses and functional analyses for multi-module operational concepts have been especially informative when considering the various operating models that might be employed at advanced reactor facilities.

Automation of Plant Operations

The NRC staff expects that another area of difference between current, large LWR operations and those of advanced reactors will be the degree of automation potentially employed in reactivity control systems. This becomes even more significant within the context of designs that could operate in an entirely autonomous manner. As identified within SECY-20-0093, "[a]utonomous operation necessitates evaluating the implications of reactivity operations being initiated and performed by automation rather than licensed operators…" (NRC, 2020d, Enc. 1 p. 7). However, fully informed consideration of autonomous reactivity control requires, in part, familiarity with the principles of automation and autonomous operation. The following subsection provides an overview of automation concepts that are useful for evaluating the implications of autonomous reactivity operations.

Overview of Automation Concepts

NUREG-0700, "Human-System Interface Design Review Guidelines," contains the following discussion of automation in Chapter 9, "Automation System":[5,6]

> Automation is a device or system that accomplishes (partially or fully) a function or task…
>
> Historically, the concept of automation was associated with control tasks. However, in modern plants, the role of automation extends to other applications as well, such as supporting operator decision making and managing the [human-system interface] HSI. In addition to its broad application, automation is more interactive. That is, while in the past, tasks were performed either by personnel or automation, today's automation can be designed to work with personnel, each "agent" having defined roles and responsibilities. (NRC, 2020c p. 9-1)

Automation is implemented in a range of ways which form "levels" that are based upon the degree of automation involved. NUREG-0700, Table 9.1, "Levels of Automation for NPP Applications," summarizes these levels of automation as shown below (NRC, 2020c, p. 9-3):

Level	Automation Tasks	Human Tasks
(1) Manual Operation	No automation	Operators manually perform all tasks.
(2) Shared Operation	Automatic performance of some tasks	Operators perform some tasks manually.
(3) Operation by Consent	Automatic performance when directed by operators to do so, under close monitoring and supervision	Operators monitor closely, approve actions, and may intervene to provide supervisory commands that automation follows.
(4) Operation by Exception	Essentially autonomous operation unless specific situations or circumstances are encountered	Operators must approve of critical decisions and may intervene.
(5) Autonomous Operation	Fully autonomous operation. System cannot normally be disabled but may be started manually	Operators monitor performance and perform backup if necessary, feasible, and permitted.

The level of automation employed for various applications at an advanced reactor facility may not necessarily be static in nature. For example, the staff are proactively considering technologically advanced automation techniques not currently used in the operating fleet, such as adaptive automation. Adaptive automation is capable of dynamically changing allocation between the automation and the operator in response to situational changes (NRC, 2020a).

[5] The term "user-interface interaction and management" refers to the means by which personnel provide inputs to an interface, receive information from it, and manage the tasks associated with access and control of information. User-interface interaction and management comprise a wide range of tasks operators undertake when accessing information and controls needed to operate the plant. (NRC, 2020c, p. 2-1)

[6] Computerized operator support systems represent a type of automation that "…supports the operator's decision-making activities, such as situation assessment and response planning" (NRC, 2020c, p. 9-4).

As described above, the highest level of automation is represented by fully autonomous operation. Autonomous operation is unique in that it presents the capability to, at its hypothetical extreme, allow for the unattended operation of a reactor facility. However, there are other facets to autonomous operation and, more broadly, automation in general, that need to be considered as well. For instance:

- The design of automation can dramatically influence the manner in which humans interact with the system; should it be necessary for a human to manually perform operations under certain circumstances (i.e., when needed for safety or desired for defense-in-depth), then the way in which automation is implemented will affect the human's ability to safely perform such operations.
- Automated systems possess varying degrees of "transparency," that is, how well the system shows a human what it is doing when behaving automatically.
- There can be substantial differences in the level of trust that humans place in automated systems, a primary contributor to which is the perceived reliability of the automated system itself.

The NRC staff has long recognized that incorporating higher levels of automation into plant designs would create new operational considerations for nuclear power plants. For instance, the following 1997 discussion from NUREG/CR-6400, "Human Factors Engineering (HFE) Insights for Advanced Reactors Based Upon Operating Experience," shows examples of the types of issues that were envisioned to potentially arise as functions were reassigned to an automatic control system in lieu of a human:

> In discussing problems that might be anticipated with future [local control stations], Hartley et al. (1984) pointed to the allocation of an increasing number of local control functions to automatic or semiautomatic systems (as opposed to human operators). The difficulties they anticipated were the same as those that can arise from increasing automation in the control room, i.e., the potential loss of operators' situation awareness, and hands-on control skills (O'Hara, 1993) as their primary role becomes one of monitoring rather than controlling. (p. 41)

As discussed above, it is possible for increased degrees of automation to contribute to degradations in certain aspects of human performance. The key insight here is that, while automation is capable of substantially enhancing both operational performance and efficacy of facility operations, the aggregate effects of increased levels of automation can be quite complex, and their introduction may also be accompanied by other operational effects that must be adequately considered (e.g., degradation of operator proficiency with manual control, reduced situational awareness on the part of operators, etc.).

Autonomous Operations

The ability of an advanced reactor facility to operate autonomously would represent a major technological advancement for power reactor facilities in the U.S. In its 2019 white paper on "Micro-Reactor Regulatory Issues," NEI discussed the technological potential for, and implications of, the autonomous operation of microreactor facilities. Specifically, NEI highlighted that:

> Micro-reactors are expected to be simple to operate, driven partially by the significant reduction in number of systems and components, and decreased reliance on human actions to ensure safety. Many micro-reactors are also expected to include automatic and/or remote operations/monitoring features. Some may be fully automatic and not require any operators, other than to initially commission the reactor and go critical, or to load and unload the fuel. (p. B-1)

It should be noted that only limited operating experience is available concerning the autonomous operation of nuclear reactors. More specifically, through outreach to both the U.S. DOE and international counterparts, the staff found that autonomous power reactor facilities were not available for purposes of comparison to U.S. nuclear reactors. However, the staff found that Canada's SLOWPOKE-2 reactor possessed attributes that can provide a limited degree of operating experience in the area of autonomous operation.

The SLOWPOKE-2 facility is currently operated by the Royal Military College of Canada. This reactor is a light water moderated, pool-type reactor. The reactor core (which produces a nominal power level of 20 kW) is contained within a beryllium neutron reflector, thus allowing for a relatively small critical mass to be used in its design. Although originally equipped with an analog control system, the Royal Military College of Canada has since replaced the SLOWPOKE-2 control system with a digital version that supports increased functionality and improved interactions with users. Most notably, however, this SLOWPOKE-2 nuclear reactor is licensed in Canada for unattended operation in automatic mode (RMC, n.d.).

In evaluating the potential for licensing facilities that employ automated reactivity control and autonomous operations, examining the NRC's existing regulatory framework is informative for the development of the requirements of 10 CFR Part 53. Current regulations mandate the presence of licensed operators at reactor facilities, as well as their responsibilities regarding the conduct of reactivity manipulations. Specifically, the applicable requirements of 10 CFR 50.54 "Conditions of Licenses," are summarized below:

> (i) Except as provided in § 55.13 of this chapter, the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator as provided in part 55 of this chapter.
>
> > [For reference, 10 CFR 50.2, "Definitions," states, in part, that "[c]ontrols when used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor."]
>
> (j) Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to part 55 of this chapter present at the controls.

> (k) An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.
>
> (l) The licensee shall designate individuals to be responsible for directing the licensed activities of licensed operators. These individuals shall be licensed as senior operators pursuant to part 55 of this chapter.

As indicated by the regulations above, the existing regulatory framework on its face precludes certain approaches to autonomous operations (for example, an autonomous design operating without licensed operators). In contrast, if the NRC chooses to accommodate autonomous operations at advanced reactor facilities, Part 53 will need to address a range of automation that is anticipated to be diverse in extent and approach, including, for example, the fully automated control of reactivity. In a related manner, the requirements associated with licensed operators and reactivity are also germane when discussing the topic of load-following in the following section of this paper.

<u>Drawing upon the preceding discussion of the automation of plant operations, the following points should be reiterated:</u>

- Advanced reactors may wish to incorporate the management of reactivity manipulations by means of fully automated systems rather than by licensed operators.
- Automation is implemented in levels that span from manual to autonomous operation.
- Autonomous operation (full automation) has the potential to support unattended reactor operations.
- Even in an autonomous design, there may still exist a need for humans to implement manual operations under certain circumstances, such as for defense-in-depth.
- Automation generally enhances operational performance, however other operational effects must be considered as well (e.g., operators losing manual control proficiency).

Load-Following

Due to the potential for advanced reactor designers to incorporate load-following capabilities into their designs, this topic will be examined here also. For the purposes of clarifying what is meant by “load following” within the context of a nuclear power plant, the IAEA describes this process in the following manner:

> Load following means to change the generation of electricity to match the expected electrical demand as closely as possible. A generating unit is said to be load following when its output is varied, either in a planned way, or in response to instructions or signals from the grid control centre [sic], to allow generation to match demand. (IAEA, 2018, p. 6)

It should be noted that “load following” may be implemented using two distinct approaches. Under the first approach, a request is transmitted by a grid control center to the plant operators; the plant operators must then take action to implement the requested change in plant output. Under the second approach, the grid control center directly changes plant output. As the former approach is generally permissible under existing regulations (i.e., the power change is ultimately implemented by licensed operators), the focus of the present discussion will be on the latter approach, which is not permissible under the existing framework.

Regarding this subject for micro-reactors in particular, the NRC staff observed within SECY-20-0093 that:

> One currently envisioned application for micro-reactors is as power production or backup power for remote locations, for example, on a micro-grid, where the micro-reactor might be the primary source of power. Such applications would likely benefit substantially from the ability to let grid demand dictate reactor power without operator manipulation. This operational configuration, however, is precluded by the current regulations. Pursuant to 10 CFR 50.54(i)-(l), any operational change in reactor power level or reactivity may be performed only through a licensed operator manipulating the controls of the facility directly or through the manipulation of other apparatus and mechanisms as authorized by a licensed operator. (NRC, 2020d, Enc. 1 pp. 6-7)

Before continuing further, a clear distinction should be established between "load-following" and "remote operations."[7] To this end, the following example is helpful in differentiating the two concepts:

- At a facility utilizing a remotely operated design, a licensed reactor operator who is not co-located with the facility would be manipulating and supervising reactor power levels, as well as other plant equipment and parameters.
- In contrast, at a facility utilizing a load-following design, a remotely located, non-licensed load dispatcher (or perhaps even automation) would directly control the electrical generation output of the facility, thereby causing changes in reactor power levels.

As noted previously, the existing regulatory framework in the U.S. does not permit load-following operations where a grid control center can directly adjust plant output at commercial nuclear facilities. However, while load-following is not currently practiced by U.S. commercial nuclear facilities, that is not the case internationally. The IAEA's 2018 report on "Non-Baseline Operations in Nuclear Power Plants: Load-Following and Frequency Control Flexible Operations" provides a considerable amount of relevant information on load-following at commercial nuclear facilities. The following excerpts illustrate some relevant highlights contained within this report:

> A plant operator [at a load following nuclear plant] could also refuse a request if it could place the plant outside its accepted capabilities and agreed envelope of operational parameters, or if it is precluded by the plant equipment conditions. More importantly, the plant operator would reject the request if it could place the unit at risk for safety and reliability. (p. 7)

[7] The subject of the remote operation of reactors is not addressed specifically within this paper. However, it should be noted that remote operation is the focus of a separate project currently being undertaken by the NRC's Office of Nuclear Regulatory Research.

> The reactivity management standards can also be influenced by the regulatory environment. For example, in the USA, direct reactivity changes may only be made by a licensed reactor operator… This would require the grid system operator or corporate energy planners to contact the plant's main control room and communicate the desired power change and ramp rate to the plant operators. The operators would then perform the activity. In France, on the other hand, reactivity (power) changes can be the direct result of grid system operator actions. The role of the licensed reactor operator in overseeing plant reactivity changes cannot be overemphasized. (p. 71)
>
> Changing the design or operation of a plant that is optimized for baseload operation will have an impact on organizational and human performance issues. This impact can be managed by appropriate training, programmes [sic], processes and procedures, some of which can be common between the plant owner/operator and grid operator. (p. 102)

<u>Key points that should be emphasized here include the following:</u>

- Load-following where a grid control center can directly adjust plant output is not currently practiced by commercial nuclear facilities in the U.S. because the practice is precluded by existing NRC regulations; however, that is not the case internationally.
- A nuclear power plant needs to be able to refuse load-following requests when complying with such requests would violate technical specifications or result in unsafe conditions.

Conclusion

As has been shown in the preceding subsections, concepts of operation that are markedly different from those traditionally present in the domestic commercial nuclear industry are likely to be encountered for some advanced reactor facilities. Factors such as multi-module operation, automation of plant operations, and load-following each represent important considerations for which the NRC staff will need adequate supporting information in order to make an appropriate safety determination. As will be further explored later in this paper, these changes have implications for both the review of license applications and for the creation of the Part 53 regulatory framework.

Defense-in-Depth and Advanced Reactor Operations

The NRC staff observed in SECY-10-0034 that “[t]he Commission has had a long-standing policy of ensuring that defense-in-depth (DID) is incorporated into the design and operation of nuclear power plants” (NRC, 2010, p. 3). To inform the incorporation of DID within an advanced reactor application, the guidance described in NEI 18-04, as endorsed by RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” provides one acceptable process for determining the adequacy of DID for non-LWR advanced reactors. It is important to note that the proposed rulemaking framework being developed in 10 CFR Part 53 is closely aligned with the methodology described in RG 1.233, which provides a process by which the content of applications will permit understanding of the system designs and their relationship to safety evaluations for a variety of non-LWR designs.

With regard to the role played by humans in achieving this DID, NUREG/CR-6947, “Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants,” states that “[o]perators contribute to the plant's defense-in-depth approach to safety and serve a vital function in ensuring its safe operation” (NRC, 2008, p. 3). Recognizing that this relationship could be affected by reduced human roles at highly automated advanced reactors, the NRC staff recently stated in SECY-20-0093 that “[a]utonomous operation necessitates evaluating the implications of… potentially eliminating human operators as a diverse means of defense-in-depth for the assurance of reactor safety” (NRC, 2020d, Enc. 1 p. 7). The staff notes that it will evaluate the staff requirements memorandum (SRM) on SECY-93-087 “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs” as part of its ongoing evaluations for this paper to ensure insights on the role of manual actions are appropriately documented, or that any inconsistencies are addressed.

In RG 1.233, the NRC staff endorsed the principles and methodology described in Nuclear Energy Institute (NEI) 18-04, Revision 1, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development” as an acceptable means of informing the licensing basis of non-LWRs (NRC, 2020b). This RG outlines that “[d]efense in depth, or the use of multiple independent but complementary methods for protecting the public from potential harm from nuclear reactor operation, is an important part of the design, licensing, and operation of nuclear power plants” (NRC, 2020b, p. 18).

NEI 18-04 provides a set of guidelines for establishing the adequacy of overall DID capabilities at non-LWR plants (NEI, 2019b). These guidelines express DID in the form of layers according to the following progression:

1. Prevent off-normal operation and AOOs [Anticipated Operational Occurrences]
2. Control abnormal operation, detect failures, and prevent DBEs [Design Basis Event]
3. Control DBEs within the analyzed design basis conditions and prevent BDBEs [Beyond Design Basis Event]
4. Control severe plant conditions and mitigate consequences of BDBEs

5. Deploy adequate offsite protective actions and prevent adverse impact on public health and safety (p. 61)

In outlining this layered approach to DID, NEI 18-04 (2019b) also reinforced the principle that, from a qualitative standpoint, “no single design or operational feature no matter how robust, is exclusively relied upon to satisfy the five layers of defense” (p. 61). Going further, NEI also clarified that “[t]he no single design or operational feature criterion is noted within this context to imply no excessive reliance on programmatic activities or human actions and that at least two independent means are provided to meet this objective” (p. 61).

<u>Within this paper’s scope, the key principles here are that DID approaches should not rely solely on a single operational feature or excessively upon human actions (or programs).</u> While the role of humans in DID at advanced reactors is an area that may need further development, these principles should be borne in mind during subsequent discussions within this paper.

Implications for License Reviews; Current Framework Limitations

As discussed in the preceding sections, new technologies, inherent safety characteristics, and non-traditional concepts of operation (when compared to large LWRs) are likely to be associated with advanced reactors. Developments such as these have substantial implications for human-system considerations within the existing regulatory framework and license application reviews, as well as for the associated regulatory guidance.

<u>Key points that have been covered thus far within this paper have included the following:</u>

- Advanced reactor designs may present low potential accident consequences.
  - Reducing accident consequences limits the hazard posed by a reactor facility and is the most reliable means of ensuring safety.
- Inherent safety characteristics can be considered absolutely reliable.
- Passive safety features and inherent safety characteristics can influence the role of personnel at advanced reactors facilities.
  - Passive safety features can still fail under certain conditions; they tend to place humans into a DID role.
  - The attributes of inherent safety characteristics, passive safety features, and automated safety systems work in distinctly different ways to support safety, while also influencing the role of the human as a layer of DID.
    - These attributes may have different impacts on both the concept of operations and on the NRC staff's assessment of the concept of operations (including the degree of emphasis placed on the HFE aspects of an application review).
- Automation may be implemented in levels that extend up to fully autonomous operation.
  - Automation generally enhances operational performance, but human interactions with automation are complex and may have certain undesired effects as well.
  - Autonomous operation has the potential to support unattended reactor operation.
  - Fully automated systems have the potential to control plant operations at advanced reactors (including reactivity manipulations) without action by licensed operators.
    - Even in an autonomous design, humans may still be needed to implement certain manual operations (i.e., DID), conduct plant monitoring, and perform periodic maintenance.
- Current DID philosophy advises that no single operational feature (no matter how robust) should be exclusively relied upon in an advanced reactor's approach to DID.
  - A related element of this philosophy also discourages excessive reliance upon human actions or operational programs in DID approaches.
- Advanced reactors are likely to incorporate novel features that are not currently utilized domestically by power reactors within the U.S., such as certain load-following capabilities.

- Load-following (i.e., the direct control of plant output by a grid control center) is not currently practiced by commercial nuclear facilities in the U.S. due to being precluded by existing regulations; however, it is permitted in some cases internationally and operating experience exists.
- Load-following requests must be able to be refused if violation of technical specifications or unsafe conditions would result, regardless of whether a human or automation is controlling the reactor.

These points should be taken into consideration when developing the human-system aspects of the Part 53 regulatory framework In order to provide a more predictable framework for advanced reactor licensing, the NRC staff is developing a regulatory framework that can be applied to a wide range of advanced reactor technologies to facilitate sound conclusions regarding whether a given concept of operations provides, at minimum, reasonable assurance of adequate protection. In a related manner, advanced reactor stakeholders need a regulatory process that is predictable and that allows for designers to make informed decisions, including those associated with the concept of operations for a proposed design. Therefore, a risk-informed, performance-based, and technology-inclusive regulatory framework is warranted that appropriately considers the role of humans and human-system integration in ensuring safe nuclear operations for advanced reactors. With that goal in mind, the following subsections address selected regulatory topics associated with human systems and operations, namely the areas of staffing, operator licensing, training, human factors engineering, and control room operations.

Staffing

The NRC has identified that advanced reactors will likely have staffing needs that differ from large LWRs. In SECY-10-0034, the NRC staff noted (within the context of SMR designs), an existing perspective "…that operator staffing may be design dependent and intended to review the justification for a smaller crew size for the advanced reactor designs by evaluating the function and task analyses for normal operation and accident management" (NRC, 2010, p. 13). For microreactor designs, the NRC staff also highlighted in SECY-20-0093 a trend towards reduced operator staffing, as illustrated by the following:

> In some cases, micro-reactor designers have proposed having no licensed operators. This may include using non-licensed personnel to perform functions that have historically been reserved for licensed operators (such as reactivity manipulations). Such cases would represent departures from existing practice for reactor facilities, given that the NRC has historically required licensed operators in the licensing of reactor facilities (both power and nonpower). (NRC, 2020d, Enc. 1 p. 5)

In a related manner, in considering the matter of staffing at microreactor facilities, Sandia National Laboratory observed that:

> Currently, 10 CFR 50.54 provides a minimum number of onsite staff. However, this number may not be appropriate for microreactor personnel. Thus, future

> regulatory guidance may need to provide flexibility for licensees to define their own staffing levels as determined by their validated analysis. (SNL, 2020, p. 24)
>
> For microreactors, a probabilistic risk assessment (PRA) and HRA would need to be completed for the novel control room design. The analyses would need to assess the need for physical human intervention onsite in the event of an emergency or off-nominal situation. (SNL, 2020, p. 25)

Regarding proposals for not using licensed operators at certain power reactor facilities, the overall matter of statutory authority for such hypothetical licensing actions should be addressed. This question was discussed by the NRC staff in SECY-20-0093. In that paper, the staff outlined the following understanding of the existing statutory requirements and how they relate to pertinent regulations:

> The staff has confirmed that licensing facilities without licensed operators does not require a change to the Atomic Energy Act of 1954, as amended. In the absence of rulemaking to establish a new category of reactors that would not require licensed operators, exemptions from existing regulations would be necessary. (NRC, 2020d, Enc. 1 p. 6)

For the purposes of informing the development of Part 53 as it relates to this topic, the following selected requirements of 10 CFR 50.54(m) illustrate key aspects of the existing regulatory framework:

> (m)(1) A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation...
>
> (2) Notwithstanding any other provisions of this section, by January 1, 1984, licensees of nuclear power units shall meet the following requirements:
>
>> (i) Each licensee shall meet the minimum licensed operator staffing requirements in the following table… [table excluded for brevity]
>>
>> (ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit…
>>
>> (iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.
>>
>> (iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a

> senior operator license or a senior operator license limited to fuel handling to directly supervise the activity...

These current regulations prescribe staffing levels for reactors licensed under Parts 50 and 52 and, historically, have served the industry and NRC well for the existing fleet of large LWR designs. However, in considering the staffing levels related to SMRs, the NRC staff recognized that such designs tend to rely more on passive safety features and inherent safety characteristics (and rely considerably less on human actions) which, in some cases, may warrant staffing levels that are not consistent with 10 CFR 50.54(m).

The NRC staff developed NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," to support the review of exemption requests from 10 CFR 50.54(m). In short, the process described in NUREG-1791 establishes how elements of an HFE program (as described in NUREG-0711, "Human Factors Engineering Review Model") can be used to provide a technical justification for reduced staffing levels (NRC, 2005).

At present, NUREG-1791 has been applied twice by the NRC staff. The first such instance occurred during a standard design review. The second such instance is currently ongoing, with the staff once again applying NUREG-1791 to review a staffing-related topical report.

NUREG-1791 has proven to be a useful guidance document for considering staffing level deviations below the baseline described in 10 CFR 50.54. However, there exist some practical limitations that prevent it from addressing certain considerations associated with advanced reactor designs. Notably, <u>NUREG-1791 does not address reducing licensed operator staffing levels to zero, such as might be the case for a fully autonomous plant design.</u> Thus, there is still some assumption of staffing inherent in this existing review and exemption process.

Another challenge associated with applying NUREG-1791 is that it relies heavily upon the human factors analyses, design activities, and validation activities described in NUREG-0711. For instance, NUREG-1791 and NUREG-0711 rely upon performance-based simulator testing data to draw conclusions about the adequacy of a given design. Extensive human factors analyses may potentially not be cost-justified or practical for designs that either do not provide for licensed operators or that do not rely upon any credited operator actions in order to demonstrate safety. However, without the data derived from these HFE-related tests, the NRC staff may, in turn, not have sufficient data to support a NUREG-1791 review. This consideration was identified by the NRC staff in SECY-20-0093, which included the following statement:

> …[NUREG-1791] is predicated on the assumption that an applicant has a human factors engineering program that is capable of providing the necessary supporting analyses. As micro-reactor designers work toward reducing the number and role of plant personnel, this assumption may need to be reevaluated, and an alternative means for establishing an appropriate technical basis may be necessary. (NRC, 2020d, Enc. 1 p. 6)

In summary, the NRC staff previously recognized the limitations of 10 CFR 50.54(m), which was developed as a prescriptive requirement. In response, the staff subsequently developed NUREG-1791 in order to allow increased flexibility to LWRs and provide guidance for assessing exemptions to the regulations in 10 CFR 50.54(m). However, licensing future applications for advanced reactors by exemption under the current Part 50 and 52 framework may not be a practical long-term regulatory framework. Therefore, an alternative means of making a safety case that is not reliant upon NUREG-1791 and NUREG-0711 would be beneficial, especially if such an alternative means were to rely upon alternate analyses, namely ones that can be appropriately scaled with the risk of the facility.

Operator Licensing

The NRC, as well as the Atomic Energy Commission which preceded it, has a long history of licensing reactor operators in the U.S., with regulations pertaining to operator licensing dating back to 1956. The NRC issues licenses to operators at power reactors in the form of both Reactor Operator (RO) and Senior Reactor Operator (SRO) licenses. The regulations that address operator licensing have a basis in statute, in part by way of the Atomic Energy Act of 1954, as amended (AEA). Specifically, Section 107 of AEA (42 U.S.C. 2137) states the following:

> "The Commission shall—
>
> (a) prescribe uniform conditions for licensing individuals as operators of any of the various classes of production and utilization facilities licensed in this Act;
>
> (b) determine the qualifications of such individuals;
>
> (c) issue licenses to such individuals in such form as the Commission may prescribe; and
>
> (d) suspend such licenses for violations of any provision of this Act or any rule or regulation issued thereunder whenever the Commission deems such action desirable.

A combination of regulations and regulatory guidance documents address the operator licensing processes of both power and non-power (i.e., research and test) reactors, an overview of which is provided below:

> <u>Part 55, "Subpart E - Written Examinations and Operating Tests", Section 55.40, "Implementation":</u>
>
> > (a) The Commission shall use the criteria in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," in effect six months before the examination date to prepare the written examinations required by §§ 55.41 and 55.43 and the operating tests required by § 55.45. The Commission shall also use the criteria in NUREG–1021 to

evaluate the written examinations and operating tests prepared by power reactor facility licensees pursuant to paragraph (b) of this section.

(b) Power reactor facility licensees may prepare, proctor, and grade the written examinations required by §§ 55.41 and 55.43 and may prepare the operating tests required by § 55.45, subject to the following conditions:

(1) Power reactor facility licensees shall prepare the required examinations and tests in accordance with the criteria in NUREG-1021 as described in paragraph (a) of this section;

(2) Pursuant to § 55.49, power reactor facility licensees shall establish, implement, and maintain procedures to control examination security and integrity;

(3) An authorized representative of the power reactor facility licensee shall approve the required examinations and tests before they are submitted to the Commission for review and approval; and

(4) Power reactor facility licensees must receive Commission approval of their proposed written examinations and operating tests.

(c) In lieu of paragraph (b) of this section and upon written request from a power reactor facility licensee pursuant to § 55.31(a)(3), the Commission shall, for that facility licensee, prepare, proctor, and grade, the written examinations required by §§ 55.41 and 55.43 and the operating tests required by § 55.45. In addition, the Commission may exercise its discretion and reject a power reactor facility licensee's determination to elect paragraph (b) of this section, in which case the Commission shall prepare, proctor, and grade the required written examinations and operating tests for that facility licensee.

(d) The Commission shall use the criteria in NUREG–1478, "Operator Licensing Examiner Standards for Research and Test Reactors," for all test and research reactors to prepare, proctor, and grade the written examinations required by §§ 55.41 and 55.43 and the operating tests required by § 55.45 for non-power reactor facility licensees.

<u>Guidance Documents Referenced in 10 CFR 55.40 for Implementing the Operator Licensing Process</u>

- <u>NUREG-1021, "Operator Licensing Examination Standards for Power Reactors":</u> Under the power reactor operator license examination process detailed in NUREG-1021, the scope of the examination *does not vary* based upon the facility.

The operating test is divided into two major categories, a walkthrough portion and dynamic simulator scenarios, with differences depending on whether the SRO or RO license level is being applied for. The walkthrough consists of 4 to 5 administrative Job Performance Measures (JPMs) and 10 to 11 systems JPMs, depending on the license type being applied for. The dynamic simulator scenarios consist of at least two scenarios that are *administered in a simulator facility*. For written examinations, applicants at both the RO and SRO levels take a 75 question RO exam, but SROs also take an additional 25 question section that is specific to their license level and emphasizes administrative functions (NRC, 2017).

- <u>NUREG-1478, "Operator Licensing Examiner Standards for Research and Test Reactors":</u>
  Research and test reactors (RTRs) vary widely in their complexity, with power levels ranging from 5 watts up to 20 MW. To take this into account, the non-power reactor operator licensing examination process detailed in NUREG-1478 *applies a graded approach for developing examinations* conducted at RTR facilities. These facilities are classified into three levels of complexity: Complex, Moderate, and Simple. Complex facilities are those licensed to operate at 500 kilowatts or greater, Simple facilities are AGN-200 (5 watts) series reactors, and Moderate facilities comprise all remaining RTR facilities. It should be noted that the RTR license examination process *does not require* a simulator facility (NRC, 2007).

  As an example, the operating test for a Complex facility is divided into three major categories, with certain differences depending on whether the SRO or RO license level is being applied for. These categories are: Category A (administrative topics), Category B (facility walkthrough), Category C (integrated facility operations). The written examination for such a facility would consist of 60 questions. Based upon the perspective that there are minimal differences in the knowledge, skills, and abilities required of RO and SRO applicants that can be tested in a written examination format at RTRs, the NRC only writes a *single written examination* for both RO and SRO applicants at such facilities (NRC, 2007).

<u>In summarizing key aspects of the regulations and guidance discussed above, the following should be noted:</u>

- The NRC has required licensing for nuclear power plant operators since the 1950s.
- The AEA requires the NRC to prescribe uniform conditions for operator licensing.
- All license examinations are either prepared by the NRC staff, or reviewed for approval by the staff, prior to administration.
- The NRC staff administers all licensing examinations that are currently given.
- The existing operator licensing examination process for <u>power</u> reactors does not vary in scope based upon the facility in question; it also requires a simulator facility.
- The existing <u>non-power</u> reactor operator licensing examination process applies a graded approach for developing examinations, does not require a simulator facility, and utilizes a single written examination for all applicant license levels.

When taking these regulations and guidance documents into consideration, it is possible to contemplate advanced reactor designs and operational concepts that would not align well with the existing operator licensing framework for power reactors. This is consistent with the perspective expressed in NEI's white paper on "Micro-Reactor Regulatory Issues," which noted that "…to the extent that licensed operators are needed at micro-reactors, it is likely that their training and requalification could be substantially different (and less extensive) than that contemplated in Part 55 and NUREG-1021" (NEI, 2019a, p. B-3). Examples of the types of changes that may be appropriate for advanced reactor operator licensing include accommodating wide variations in both reactor technologies and concepts of operations, not adopting all existing simulator facility requirements, and allowances for varying licensing examination scope on a facility-specific basis. Accordingly, it is appropriate to consider a revised approach to operator licensing that can flexibly address a wide variety of advanced reactor designs in a manner that is both efficient and consistent with maintaining safe plant operations.

The Shift Technical Advisor Position

In addition to licensed operators, the control room staffing at power reactors also includes the Shift Technical Advisor (STA) position. In contrast with licensed operators, the requirements for STA staffing are primarily rooted in Commission policy, and not regulation or statute. As addressing the role of the STA for advanced reactors requires an understanding of the basis behind, and underlying purpose of, this position, an overview of the relevant history and policy associated with the STA position is provided below.

In response to the March 1979 accident at Three Mile Island (TMI), the NRC identified in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," a recommendation for each nuclear power plant to have on duty a dedicated STA whose function was to provide engineering and accident assessment advice to the shift supervisor in the event of abnormal or accident conditions. The NUREG identified STA qualifications as including a bachelor's degree in engineering (or an equivalent), as well as specific training in plant response to transients and accidents. The recommendation for an STA position was subsequently incorporated into NUREG-0737, "Clarification of TMI Action Plan Requirements," in 1980. The recommendations of NUREG-0737 and NUREG-0578 were later followed by plant-specific Confirmatory Orders. Notably, NUREG-0737 included the following clarification concerning the STA position:

> The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue. (NRC, 1980, p. 3-3)

On October 28, 1985, 50 FR 43621, "Commission Policy Statement on Engineering Expertise on Shift," final, was issued, which states:

> Concurrently, the NRC and industry embarked on a longer-term effort aimed at upgrading staffing levels and the training and qualifications of operating staffs, improving the man-machine interface, and increasing capabilities for responding to emergencies. At the time the STA requirement was imposed, it was intended that the use of the dedicated STA would be an interim measure only until these longer-term goals were achieved. These long-term initiatives collectively result in an improvement in the capabilities and qualifications of the shift crew and their ability to diagnose and respond to accidents. These initiatives include shift staffing increases, training and qualification program improvements, hardware modifications, emphasis on human factors considerations, procedural upgrades, and development of extensive emergency response organizations to augment on-shift capabilities during abnormal conditions.

On August 15, 1989, 54 FR 33639 was issued, which contained a policy statement on education for SROs and shift supervisors, stating that:

> The Commission believes that the safety of commercial power reactors is enhanced by having on each shift team of NRC licensed professionals that combine technical and academic knowledge with plant-specific training and substantial hands on operating experience. The Commission's position is predicated on the fact that, even though reactor licensees try to anticipate and address in training programs and procedures all conceivable situations which could arise during normal and off normal operation, there will always be the potential for situations to arise which are not covered through training or procedures. The Commission is persuaded that there is a need for some individuals on each shift who have an innate understanding of systems level performance of a nuclear power plant. The types of knowledge that are needed are scientific and engineering fundamentals and the basic scientific principles that govern the behavior of electrical, mechanical and other engineered systems. This is precisely the type of knowledge that academic institutions develop and convey well and that form the basis of an academic degree program in a technical discipline.

The 1989 policy statement also reaffirmed the previous statement on expertise on shift, stating that:

> It is important to have engineering and accident assessment expertise available to the operating crew at all nuclear power plants. The STA has proven to be a worthwhile addition to the operating staff by providing an independent engineering and accident assessment capability, and we support continuation of this position. (54 FR 33639; August 15, 1989)

As described above, the STA position was originally established as a short-term action following the TMI accident to improve the ability of the on-shift operating crew to recognize, diagnose and effectively respond to plant transients and abnormal conditions. This was associated with longer-term actions to improve the qualifications of shift managers and senior operators in addition to upgrading the man-machine interfaces (i.e., the human-system interfaces) in the main control room. However, both the 1985 and 1989 Commission policy statements subsequently supported continuation of the STA position to provide engineering and accident assessment capability, as well as for the enhancement of plant safety.

In summary, the NRC staffs' current interpretation of the STA Commission policy is that, on each shift, there should be at least one person on duty who has a degree in physical science, engineering, or engineering technology (or who has a Professional Engineering (PE) license). The function of this person is to provide independent engineering expertise, accident assessment, and technical advice to the main control room operators. At present, the staff has concluded that a regulatory framework for advanced reactors that eliminates the STA position at a facility would be a departure from existing Commission policy, as well as from longstanding agency and industry practice.

Training

In addition to operator licensing requirements, the training requirements for the personnel at nuclear power plants are also established by regulation. It should be noted that portions of these regulations have statutory bases as well, with the Nuclear Waste Policy Act of 1982, as amended (NWPA), being of significance. Section 306 of the NWPA (42 U.S.C. 10226) states, in part, the following:

> The Nuclear Regulatory Commission is authorized and directed to promulgate regulations, or other appropriate Commission regulatory guidance, for the training and qualifications of civilian nuclear power plant[8] operators, supervisors, technicians and other appropriate operating personnel. Such regulations or guidance shall establish simulator training requirements for applicants for civilian nuclear powerplant operator licenses and for operator requalification programs; requirements governing NRC administration of requalification examinations; requirements for operating tests at civilian nuclear powerplant simulators, and instructional requirements for civilian nuclear power plant licensee personnel training programs…

10 CFR 50.120, "Training and qualification of nuclear power plant personnel," describes, in part, categories of nuclear power plant personnel that the NRC requires to be included within licensee training programs, as shown in the following excerpt:

> (b) Requirements.

[8] 42 U.S.C 10101, "Definitions," states, "The term "civilian nuclear power reactor" means a civilian nuclear powerplant required to be licensed under section 2133 or 2134(b) of this title," where "title" is Title 42 and sections 2133 and 2134 correspond with AEA sections 103 and 104, respectively. Section 2133 of Title 42 explains that these include production and utilization facilities for industrial or commercial purposes.

> (1)(i) Each nuclear power plant operating license applicant, by 18 months prior to fuel load, and each holder of an operating license shall establish, implement, and maintain a training program that meets the requirements of paragraphs (b)(2) and (b)(3) of this section…
>
> (2) The training program must be derived from a systems approach to training as defined in 10 CFR 55.4, and must provide for the training and qualification of the following categories of nuclear power plant personnel:
>
> > (i) Non-licensed operator.
> >
> > (ii) Shift supervisor.
> >
> > (iii) Shift technical advisor.
> >
> > (iv) Instrument and control technician.
> >
> > (v) Electrical maintenance personnel.
> >
> > (vi) Mechanical maintenance personnel.
> >
> > (vii) Radiological protection technician.
> >
> > (viii) Chemistry technician.
> >
> > (ix) Engineering support personnel.

It should be noted that other sections of NRC's regulations cover other aspects of training requirements, such as that specified for licensed operators under 10 CFR Part 55. For example, licensed operator requalification training is addressed separately under 10 CFR 55.59, "Requalification."

As indicated in the preceding excerpt, 10 CFR 50.120(b)(2) requires that the training and qualification programs for the listed categories of personnel be based upon a systems approach to training (SAT). The recognition of SAT-based training programs as constituting an acceptable means of meeting NRC regulations can also be seen in 10 CFR 55.59, as shown in the excerpt below:

> (c) Requalification program requirements. A facility licensee shall have a requalification program reviewed and approved by the Commission and shall, upon request consistent with the Commission's inspection program needs, submit to the Commission a copy of its comprehensive requalification written examinations or annual operating tests. The requalification program must meet the requirements of paragraphs (c) (1) through (7) of this section. In lieu of paragraphs (c) (2), (3), and (4) of this section, the Commission may approve a program developed by using a systems approach to training.

In order to fully appreciate the value of SAT-based training, it is informative to understand the NRC staff's original reasoning for incorporating SAT requirements into the regulations covering personnel training at nuclear power plants. On January 7, 1992 (57 FR 537), the NRC published a proposed rule on the "Training and Qualification of Nuclear Power Plant Personnel," which included, in part, the following discussion regarding SAT-based training:

> The SAT was selected because it has the following characteristics:
>
> (1) Training content and design are derived from job performance requirements;
>
> (2) Training is evaluated and revised in terms of the job performance requirements and observed results on the job;
>
> (3) Trainee success in training can predict satisfactory on-the-job performance; and
>
> (4) Training and associated programs can be readily audited because they involve clearly delineated process steps and documentation. (57 FR 538; January 7, 1992)

Furthermore, the proposed rule also provided the following explanation of the adaptability of SAT and its suitability for nuclear power plant training:

> The SAT process is a generic process, and its application is not limited to a certain subject matter or to specific licensee personnel. Training programs based on job performance requirements have been successfully used by the military for over 20 years, and by the nuclear industry for much of the past decade. Furthermore, the Commission has recognized the appropriateness of using this approach to training in its requirements for operator licensing prescribed in § 55.31(a)(4), and for operator requalification prescribed in § 55.59(c). (57 FR 538; January 7, 1992)

The SAT process itself is a training methodology that is defined within 10 CFR 55.4, "Definitions," as being comprised of the following programmatic elements:

> *Systems approach to training* means a training program that includes the following five elements:
>
> > (1) Systematic analysis of the jobs to be performed.
> >
> > (2) Learning objectives derived from the analysis which describe desired performance after training.
> >
> > (3) Training design and implementation based on the learning objectives.
> >
> > (4) Evaluation of trainee mastery of the objectives during training.
> >
> > (5) Evaluation and revision of the training based on the performance of trained personnel in the job setting.

Based upon the preceding discussion, it is evident that SAT-based training programs form an important element of personnel training models within current regulations. Given the central nature of this training approach to the existing regulatory framework, as well as the inherently adaptable nature of the SAT-based training methodology, it is appropriate that SAT-based training should be considered for similar inclusion in a regulatory framework for advanced reactors as well.

<u>In summary, with regard to the training requirements for the personnel at nuclear power plants, the following highlights serve to emphasize some key points:</u>

- The NWPA directs the NRC to establish regulations for the training and qualifications of nuclear power plant operators, supervisors, technicians and other operating personnel.
  - The NWPA also directs the NRC to establish requirements for simulator training, requalification examinations, operating tests, and instructional requirements.
- Existing regulations prescribe training and qualification requirements for specific categories of personnel at nuclear power plants.
  - As these existing personnel categories are associated with a regulatory framework that centers on large LWRs, modifications to these categories should be considered for future regulations covering training at advanced reactors.
- The SAT plays a central role in nuclear training and qualification programs; it is identified as either a requirement or as an acceptable approach to training within regulations.
- The SAT process is generic in nature and can be adapted to any reactor technology, including those associated with essentially any foreseeable advanced reactor designs.

Human Factors Engineering

The application of HFE in the design of nuclear power plant control rooms is required under existing regulations. Specifically, 10CFR 50.34(f)(2)(iii), directs applicants under Part 52 to "[p]rovide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts." This post-TMI requirement is the primary regulation[9] that explicitly requires the incorporation of HFE, and its scope focuses on the facility control room.

The current human factors engineering review model utilized by the NRC is provided in NUREG-0711, "Human Factors Engineering Review Model." This NUREG is used by the NRC staff to review the HFE programs of construction permit, operating license, standard design certification, and combined license applicants. The intent of this review process is to verify that an applicant's HFE program is derived from practices and guidelines that the NRC staff has found to be acceptable. Acceptable HFE programs, in turn, are the those which incorporate the following twelve elements (NRC, 2012a, p. iii):

[9] In 10 CFR Part 52, Section 52.47, "Contents of applications; technical information," refers to the section 50.34(f)(2) requirements. Specifically, 52.47(a)(8) requires that applications contain "[t]he information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v)." Section 52.79(a)(17) includes a similar requirement.

- HFE Program Management
- Operating Experience Review
- Functional Requirements Analysis and Function Allocation
- Task Analysis
- Staffing and Qualifications
- Treatment of Important Human Actions
- Human-System Interface Design
- Procedure Development
- Training Program Development
- Human Factors Verification and Validation
- Design Implementation
- Human Performance Monitoring

Within the NRC's human factors engineering review model, the concept of operations can play a key role. NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," notes that "[a] concept of operations covers all facets of personnel interaction with a complex system; therefore, it provides a good organizational framework within which to cluster and integrate a wide variety of issues" (NRC, 2008, p.11). As advanced reactors are expected to be accompanied by innovative technologies, it will be important for the NRC staff to have an integrated view of how such facilities will be operated. This will help to prevent the NRC staff from making inappropriate assumptions about operations that are based upon past experiences with large LWRs. In the case of advanced reactors, it is probable that some such traditional assumptions (e.g., having a centralized control room, operator actions being necessary for safety, static degrees of automation, etc.) may no longer apply. Thus, a clear understanding of the relevant concept of operations, as well as its implications for HFE, can be an asset in supporting appropriate, risk-informed, performance-based decisions by the staff during the review of advanced reactor applications.

NUREG/CR-6947 goes on to indicate that "[a]nother reason for choosing concept of operations as an organizational framework is that it plays a significant role in the NRC's review of the human factors aspects of NPPs [Nuclear Power Plants], as per NUREG-0711" (NRC, 2008, p.11). For its part, NUREG-0711 defines the concept of operations as follows:

> A concept of operations (ConOps) defines the goals and expectations for the new system from the perspective of users and other stakeholders and defines the high-level considerations to address as the detailed design evolves. An HFE-focused ConOps addresses the following six dimensions (NRC, 2012a, p. 113):
>
> - Plant Goals (or Missions)
> - Agents' Roles and Responsibilities ["agents" refers to the person or automation (or any combination thereof) that are responsible for completing a plant function]
> - Staffing, Qualifications, and Training
> - Management of Normal Operations

- Management of Off-normal Conditions and Emergencies
- Management of Maintenance and Modifications

Under the current regulatory framework, 10 CFR 50.34(a)(6) requires that applications for a construction permits include "[a] preliminary plan for the applicant's organization, training of personnel, and conduct of operations." Although the plan for the "conduct of operations" and the "concept of operations" are not synonymous, this existing requirement for including such operations-related information in applications provides both a useful reference point and an evolutionary step towards future approaches.

In summary, the NRC staff's human factors engineering reviews currently center on human actions and on the interactions between those humans and the tools they use; this focus is typically on the human-system interfaces located within control rooms. While an understanding of the factors listed above is necessary to support all human factors reviews, moving forward towards a risk-informed, performance-based, and technology-inclusive framework should include taking a fresh look at how human factors engineering reviews can be implemented most effectively for advanced reactors. This warrants the consideration of new approaches, such as the application of scalable human factors engineering review processes, as well as thinking beyond the confines of traditional control rooms. It has also been demonstrated that the concept of operations can serve as a valuable tool in gaining the kind of design understanding necessary to conduct appropriate human factors engineering reviews. As will be further discussed in subsequent sections of this paper, both the concept of operations and the human factors engineering content of applications are anticipated to be important considerations during the review of operations aspects of applications for advanced reactors.

The Evolving Concept of the "Control Room"

Some advanced reactor facilities may wish to not utilize traditional control rooms in their designs. The NRC staff noted within SECY-20-0093 that "…micro-reactors that do not rely on operators to perform safety-related actions, and that can rely on autonomous operation, may have facility designs that do not include a control room from which individuals would be able to operate the facility" (NRC, 2020d, Enc. 1 p. 7). Separately, NEI has made the following observation regarding the potential future for control rooms at microreactor facilities:

> Due to advances in technology, a traditional control room may not be necessary. For micro-reactors that demonstrate the safety of the reactor can be assured without the need for operator action, and if an individual is unable to compromise the safety of the reactor through the manipulations of the controls, then there would be no need for requirements relating to the control room or for an operator-initiated shutdown. Therefore, requirements for control rooms, such as 10 CFR Part 50 Appendix A, General Design Criteria 19, may not be applicable to micro-reactors. The use of portable monitoring devices may also allow responsible personnel to monitor plant parameters and maintain operational control from either outside the control room or offsite during normal operations. (NEI, 2019a, p. B-3)

As referenced above, 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," includes Criterion 19 - Control room, which contains, in part, the following requirement:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions…

It is possible, for example, that control capabilities for those functions associated with reactivity and safety systems could be distributed throughout a facility or, perhaps, even controlled by means of portable devices. Irrespective of the location from, or means by which, such control of important plant functions is implemented, a need will still exist to ensure that such actions are facilitated in a manner that is consistent with safe, reliable facility operations in which the risk posed by human errors is minimized to the maximum extent practical. While written within the context of a different set of control room-related regulations, NEI, in its white paper on "Micro-Reactor Regulatory Issues," suggested that "…since some micro-reactors may not have a control room, it may be beneficial to modify several regulations to refer to 'reactor controls' rather than the 'control room'" (NEI, 2019a, p. B-4).

It is important to note that, at present, existing regulations only link human factors engineering requirements to traditional control room locations. Specifically, where human factors engineering is concerned, the primary regulation of interest is 10 CFR 50.34(f)(2)(iii), which requires that applicants "[p]rovide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts."

Based upon these considerations, three interrelated issues can be identified:

- Requirements addressing matters associated with control rooms will need to be revisited in Part 53 with an understanding that the functions involved may become decentralized in an advanced reactor facility.
- In order to continue contributing to the safe operation of reactor facilities, human factors engineering requirements will essentially need to be able to "follow" important functions if they are relocated outside of a traditional control room, thereby ensuring that the appropriate standards will be incorporated both where and when they are relevant.
- It may also be necessary to account for the potential emergence of functions that have no precedent within traditional control rooms as well. Specifically, such functions could be associated with human actions deemed important to the safe operation and maintenance of an advanced reactor facility.

Additional Organizational Considerations

In contemplating a hypothetical advanced reactor design that is operated without humans (i.e., a fully autonomous plant), it should be noted that the existing regulatory framework also assigns certain responsibilities of licensee organizations to their licensed operators that are beyond those covered in the portions of 10 CFR already discussed. As an example of this, the requirements of 10 CFR 50.54(x) and (y) are summarized below:

> (x) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.
>
> (y) Licensee action permitted by paragraph (x) of this section shall be approved, as a minimum, by a licensed senior operator… prior to taking the action.

This example makes it clear that the hypothetical absence of licensed operators at an advanced reactor facility would require resolution of how, in part, the intent of this requirement (i.e., authorizing an emergency-related departure from license conditions for the protection of the public) would still be accomplished such that a reasonable assurance of public health and safety could be demonstrated. However, beyond this example, there remain a number of other licensed operator administrative responsibilities that are both important to safety and derived from regulatory requirements; such responsibilities would need to be addressed as well (NRC, 2020d). Examples of such administrative responsibilities include compliance with technical specifications, operability determinations, NRC notifications, emergency declarations, risk-assessment, maintenance oversight, and radiological release limit compliance.

Summary

As has been observed within this section, the efficiency and effectiveness of NRC licensing activities related to staffing, operator licensing, training, human factors engineering, and control room operations are expected to be impacted by changes associated with advanced reactors. <u>The following points summarize important considerations identified within this section:</u>

- General Staffing Considerations
  - Current staffing requirements under 10 CFR 50.54(m) are prescriptive in nature.
  - NUREG-1791 facilitates staffing-related regulatory exemptions but is not designed to account for facilities without any licensed operators.
  - Staffing models that rely upon facility-specific staffing analyses that are flexible (instead of prescriptive staffing requirements) may be better suited to advanced reactors.
  - Criteria are needed in a new regulatory framework that can support safety determinations on staffing models that do not utilize any licensed operators.
  - Regardless of the number of licensed operators, each licensee must provide for the accomplishment of administrative and emergency preparedness functions with implications for safety.
- Operator Licensing Considerations
  - The AEA requires the NRC to prescribe uniform conditions for operator licensing.
  - License examinations are currently approved and administered by the NRC staff.
  - Current power reactor operator licensing examinations do not vary in scope based upon the specific facility in question and require a simulator facility.
    - The RTR operator examination process applies a graded approach.

- Shift Technical Advisor Considerations
    - The current interpretation of the STA policy is that operating crews need to include one person with a degree in either a physical science, engineering, or engineering technology (or a PE license).
- Training Considerations
    - The NWPA directs the NRC to establish regulations for the training and qualifications of nuclear power plant operators, supervisors, technicians and other operating personnel.
    - The NWPA also directs the NRC to establish requirements for simulator training, requalification examinations, operating tests, and instructional requirements.
    - Existing regulations prescribe training and qualification requirements for specific categories of personnel at nuclear power plants.
        - Modifications to these categories need to be considered in developing future regulations covering training at advanced reactors.
    - SAT plays a central role in nuclear training and qualification programs; it is identified alternately as a requirement or as an acceptable approach to training within regulations.
        - SAT is generic and can be adapted to foreseeable advanced reactor designs.
- Content of Applications Considerations
    - Advanced reactor applications may need to contain a broader scope of HFE information.
    - A ConOps can assist HFE reviewers in conducting appropriate reviews.
- Human Factors Engineering Considerations
    - A risk-informed, performance-based, and technology-inclusive framework requires scalable HFE reviews.
    - Current HFE reviews tend to focus on control room human-system interfaces.
    - Advanced reactors may require HFE review beyond traditional control rooms.
        - Regulations need to adapt to functions moving outside of control rooms.

Through an understanding of the nature of these interrelated issues, it is possible to create a risk-informed, performance-based, and technology-inclusive framework that approaches these considerations in an integrated manner. With that being said, the focus of this paper will now shift to applying such an approach to establishing objectives that can appropriately inform the content of Part 53.

Objectives for the Part 53 Rule: An Integrated Approach to Humans and Systems

Building upon the aforementioned considerations, specific objectives for the Part 53 rule will now be presented here. Instead of treating the role of humans in the operational safety of advanced reactors (e.g., staffing, training, operator licensing and human factors engineering) as discrete review areas, the following objectives employ an integrated approach to humans and systems:

- The rule should recognize that staffing, training, operator licensing, and human factors are interrelated areas that affect one another, and that diverse advanced reactor technologies necessitate integrating the review of these areas under a flexible approach *(e.g., an advanced reactor's specific human-system interface design influences what staffing is needed, the depth of the human factors review, and operator licensing needs)*.
- The rule should account for variations in accident consequence severity among the different advanced reactor technologies.
    - The rule should also facilitate an informed licensing process for any combination of design features *(e.g., automation, modular operations, load following, etc.)*.
- The rule should require a human factors engineering program that is adequate to ensure that humans (e.g., operators, supervisors, technicians, and other appropriate personnel) can understand the status of the plant, take actions necessary to ensure safety, and perform other important technical and administrative functions with safety implications.
    - Those human roles that are associated with the management and availability of plant-specific safety functions should also be taken into account when considering the applicability of HFE requirements.
- The rule should ensure that the operator licensing process accomplishes the following:
    - complies with applicable statutory requirements *(i.e., AEA and NWPA)*;
    - conforms with accepted testing standards;
    - facilitates consistent and reliable licensing decisions by the NRC;
    - makes efficient use of NRC and vendor and/or facility licensee resources; and
    - provides assurance that operators can manage plant-specific safety functions.
- The rule should account for designs that do not utilize traditional control rooms.
    - This should include accounting for the potential of an applicant demonstrating that a main control room is unnecessary *(e.g., a design might complete all control functions using a mobile tablet or from various field locations)*.
- The rule should allow for consideration of innovative features intended to make new designs safer, while also accounting for uncertainties associated with new approaches.
- The rule should, in a non-prescriptive manner, require staffing levels needed to support safe operation and allow for the possibility of demonstrating that no humans are necessary; however, the rule should also *prescribe minimal requirements that must be met to not use licensed operators at all*.
- The rule should ensure that advanced reactor defense-in-depth approaches do not rely exclusively upon a single operational feature or rely excessively upon human actions.

- The rule should account for the possibility of load-following where the load changes themselves are controlled externally from a grid control center; however, the rule should also require that load-following requests must be capable of being refused if unsafe conditions would result (regardless of whether a human or automation is in control).
- The rule should require that sufficient information be submitted as part of applications to facilitate the NRC staff's licensing reviews of the areas outlined within these goals. *Examples* of such information may include the following:
    - the concept of operations for the design;
    - functional requirements analyses showing what features, systems, and human actions are relied upon to demonstrate safety during postulated events;
    - a function allocation describing the assignment of safety functions to personnel, automatic systems, or combinations of both.
    - the staffing plan, with supporting human factors engineering-based analyses;
    - a proposed SAT-based training program for relevant personnel categories; and
    - a description of the program for evaluating and applying operating experience.

Potential Approaches to Advanced Reactor Licensing Reviews

This section further discusses potential ways in which the objectives discussed in the previous section could be implemented. It should be recognized that some aspects of the approaches discussed within this section would be reflected by the requirements of Part 53, while other aspects would be captured within associated regulatory guidance and review documents. The specific topics addressed within this section include scalable human factors engineering reviews, evaluating proposals for staffing facilities without using licensed operators, and an operator licensing process that is suitable to advanced reactors. The subsequent section will go on to further consider potential changes in the contents of applications for advanced reactors.

Scalable Human Factors Engineering Reviews

As noted earlier, advanced reactor designers may find that the extensive HFE analyses, design activities, and validation elements described in NUREG-0711 may be unnecessary or unwarranted for designs that have reduced or minimal reliance on human action. In a related manner, it would not be desirable to expend NRC staff resources in the review of certain HFE-related analyses should there be little safety value associated with them for a given application. In light of such considerations, current efforts are underway to develop a scalable approach to performing technical reviews in the HFE area for advanced reactor license applications. For example, SECY-20-0093 describes a key component of these ongoing efforts:

> The staff has initiated work under contract with Brookhaven National Laboratory (BNL) to develop a method for scaling the scope and depth of HFE reviews for non-LWR technologies such as micro-reactors. The objective of this effort is to enable the staff to readily adjust the focus and level of staff HFE review efforts considering factors such as risk insights and the unique characteristics of the design or facility operation (including remote/autonomous operation). (NRC, 2020d, Enc. 1 p. 8)

The NRC staff anticipates that BNL's development work on this scalable HFE process will be complete by June 2021. In the interim, the NRC staff also has the ability to adjust the scope of a NUREG-0711 review on a case-by-case basis should a given license application warrant a reduction in the scope of an HFE area technical review.

Evaluating Proposals for Staffing Facilities Without the Need for Licensed Operators

For an advanced reactor design to be able to justify not using licensed operators, it must be able to demonstrate that adequate protection of the public health and safety will exist in the absence of any operator action for preventing or mitigating accidents.

<u>The following are examples of criteria that could potentially be used for assessing the acceptability of an advanced reactor design operating *without using any* licensed operators:</u>

1. <u>The accident analysis for the design should demonstrate that radiological consequence criteria will be met without reliance on human actions for event mitigation, defense-in-depth, or safe shutdown</u>. It is expected that such radiological consequence criteria will be established as part of Part 53 rulemaking; these criteria may be similar to those

discussed in NEI 18-04.[10] Furthermore, in conjunction with the revised, graded approach to advanced reactor operator licensing that is discussed within this paper, *only licensed operators should be credited for defense-in-depth actions.*

2. The safety of the design should be reliant upon *inherent safety characteristics*. In order to establish adequate protection of the public health and safety in the absence of credited human actions, the absolute reliability of inherent safety characteristics (as discussed earlier in this paper) would likely be essential as part of such a design.
3. The design may be either fully autonomous or have a sufficient degree of autonomy to support safety without human action.
    a. Consistent with the discussion of automation earlier in this paper, an autonomous design may still require human presence and action for startup. In such an instance, it may be appropriate to require an applicant to do *one* of the following:
        i. utilize a licensed operator to conduct the reactor startup, or
        ii. demonstrate that the safety case bounds all postulated errors by a non-licensed operator during conduct of a reactor startup. This extra layer of analysis is appropriate since the abilities of a non-licensed startup operator would not provide the NRC staff with the same degree of assurance as that of a licensed operator.
4. If load-following will be incorporated into the design, then the autonomous control system should be capable of refusing demands from the grid operator when they could challenge the safe and reliable operation of the plant, or when precluded by the plant equipment conditions.
5. License conditions should be established for the facility by which those administrative responsibilities with safety implications that would otherwise have been allocated to licensed operators are reassigned appropriately (such as to a designated licensee manager). As noted earlier in this paper, such administrative and emergency preparedness responsibilities include (but are not limited to) authorizing

[10] NEI 18-04 discusses a risk-informed approach for various postulated events and their associated radiological consequences. In considering the approach outlined in this document, it should be recalled that risk is considered to be the product of consequence and frequency. A simplified version of selected elements of that approach will be discussed here. Specifically, NEI 18-04 describes a set of frequency–consequence criteria, the evaluation correlation of which is referred to as an "F-C Target." These F-C Target values are selected in such a manner that, as the frequency of a given event decreases, the corresponding risk does not increase (NEI, 2019b, pp. 7-9). The F-C Target values for different categories of events are associated with varying radiological consequences as follows (frequencies are omitted from this list for simplicity):

- The F-C Target for high-frequency Anticipated Operational Occurrences is based on a risk profile defined by an annual exposure limit of 100 mrem.
- The F-C Target for Design Basis Events is based on a risk profile involving dose consequences that range from 1 to 25 rem at the Exclusion Area Boundary for the 30-day period following the onset of the release.
- The F-C Target for Beyond Design Basis Events is based on a risk profile involving dose consequences that range from 25 to 750 rem; the basis described for this is that of ensuring that the Quantitative Health Objective for early health effects is not exceeded. (NEI, 2019b, pp. 7-9)

While not intended to provide an in-depth discussion of this aspect of NEI 18-04, this brief summary provides some familiarization with an approach that has been taken for addressing risk and radiological consequences at non-LWR facilities.

emergency-related departures from license conditions for the protection of the public, compliance with technical specifications, operability determinations, maintenance and configuration control, making required notifications to the NRC, and emergency declarations.

6. For the STA position, the staff would need to engage with the Commission on a proposed departure from Commission policy should an advanced reactor applicant propose a staffing plan that does not include on-shift engineering expertise (either by a dedicated STA or a dual-role qualified operator). A key consideration for any such staffing proposal would likely be the applicant's ability to demonstrate that the result of staffing-related analyses remain adequate in the absence of the on-shift engineering expertise provided by an STA.

Potential variations on, and alternatives to, the criteria outlined above may include the following:

- Allowing non-licensed operators to be credited for defense-in-depth related actions.
- Not requiring licensed operators for certain classes of advanced reactor facilities (such as, for example, microreactors).
- Requiring an STA anytime that a licensed operator is present (e.g., during startups).
- Including a criterion that would require advanced reactors to be rated below a certain thermal power level (e.g., 10 MW) in order for not using licensed operators to be an option (as a means reducing associated source terms for such facilities).
- Setting more restrictive accident dose limits for plants where licensed operators are not present.
- Only allowing autonomous facility operations during certain modes of operation (e.g., requiring licensed operators staffing for facility startup and the conduct of certain tests).

It should also be noted that it is possible that certain *non-power* reactors (which require licensed operators under current regulations) might also satisfy any such criteria to not use licensed operators; this would represent a potential area of conflict that may need to be addressed in the future.

Applying a Scalable Approach to Operator Licensing Requirements

The NRC staff is currently in the early stages of development work on a revised approach to the operator licensing process for advanced reactors. This effort builds upon the existing operator licensing processes and principles that are outlined earlier within this paper. Although preliminary, some of these early concepts will be discussed here. The following subsection highlights some basic elements of an operator licensing framework for advanced reactors.

In order to comply with Section 306 of the NWPA (as discussed earlier in this paper), the NRC will need to continue to accomplish, in part, the following for those advanced reactors that are classified as civilian nuclear power plants (and, therefore, subject to NWPA statutory requirements in question):

1. Establish regulations or other Commission guidance for the training and qualifications of civilian nuclear power plant operators, supervisors, technicians, and other appropriate

operating personnel, and the instructional requirements for personnel training programs.[11]

   a. A technology-neutral approach to item (1) above might be achieved, in part, by requiring that advanced reactor training programs:
      i. Be derived from a SAT
      ii. Provide for the training and qualification of the following categories of nuclear power plant personnel:
         1. non-licensed operators,
         2. applicants for operators' licenses,
         3. licensed operators,
         4. supervisors,
         5. technicians, and
         6. other appropriate plant personnel as applicable.

2. The regulations or other Commission guidance must establish the following:[11]
   a. simulator training requirements for applicants for operator licenses and for operator requalification programs;
      i. A suitable approach to item (a) above might be achieved, in part, by requiring that advanced reactor facilities require simulators…
         1. …if they will be using licensed operators as part of their staffing model.
         2. …that can model the expected plant response to personnel input, as well as for normal and abnormal conditions.
         3. …that are capable of modeling those events where operators perform actions associated with the manipulation of plant power level and reactivity, as well as any other activities that personnel perform that require an NRC license.
   b. requirements governing NRC administration of operator requalification exams;
   c. requirements for operating tests at the powerplant simulator (i.e., that exists for the purpose of (a) above, which is training operator license applicants and licensed operators).

A process that advanced reactor vendors and licensees could use to develop an operator licensing examination for their sites could be implemented, for example, by way of either supplementing NUREG-1021 or via the issuance of a Regulatory Guide. The basic components of this process could consist of the following:

- A job task analysis would be an essential first step to identify the knowledge, skills, and abilities (KSAs) that are related to the purpose for which operators would be licensed (i.e., protection of public health from harmful radiation and the management of plant-specific safety functions). The vendor or licensee would perform this analysis and

[11] It should be noted that, although there is a clear distinction between regulations and guidance, the wording presented here is reproduced directly from that of the indicated statute.

the NRC staff would review the product. These actions would establish what is known as the “content domain” and the amount of it that is testable.

- The training and evaluation methods to be used would be selected using a Systems Approach to Training (SAT) process. This would serve to ensure that the training and evaluation settings remained appropriate for the job the operator is being licensed to perform. In the case where it is necessary to issue operator licenses, the vendor or licensee would propose a means of testing a representative sample of the important KSAs and the methods to be used (e.g., written exam, JPMs, or simulator test), including recommendations on details such as how many written questions, JPMs, or simulator events would be included in the initial licensing examination.
- Initial licensing examination pass/fail criteria (e.g., the cut score for a written exam) and the basis for those criteria would be proposed by the vendor or licensee and reviewed by the staff.
- The vendor or licensee would then pilot the proposed initial licensing examination and make changes as needed.
- The examination would then be reviewed, approved, and administered by the NRC staff.
- Following documentation of the examination results, operator licenses would then be issued by the appropriate NRC licensing authority.

Potential variations on, and alternatives to, the general process outlined above include the following:

- There may be a practical limit to the number of different technologies that NRC examiners can remain trained and proficient on. In light of the potential for a number of different advanced reactor technologies to seek to license operators, the NRC will need to determine how to establish the requisite examiner technical expertise for reviewing and administering such license examinations.
    - One potential option would be to permit vendors and licensees to <u>administer their own operator licensing examinations</u>, with the NRC retaining its authority to either issue or deny operator licenses based upon the examination results.
        - Under such a model, the NRC staff could still approve the examination prior to administration and inspect the examination administration process.
- The NRC could, in theory, not require licensed operators for certain classes of advanced reactors.
- Another option would be to simply require advanced reactors to implement operator licensing examination programs under the NUREG-1021 process for power reactors. However, this would be burdensome for small facilities and would not address the related considerations noted earlier in this paper.

Potential Implications for Contents of License Applications

The focus of this section will be to further discuss examples of areas in which, based upon the considerations discussed thus far, the contents of advanced reactor applications may differ from those associated with large LWRs. These areas will include concepts of operations, staffing analyses, and HFE programs.

Concept of Operations

Against the backdrop of SMR development, NUREG/CR-7126, "Human-Performance Issues Related to the Design and Operation of Small Modular Reactors," provides the following explanation of the contribution made by the concept of operations (ConOps) to the overall development and review process:

> A ConOps document plays an important role in the NRC's review of the HFE aspects of NPPs. [The] ConOps [is defined as] as a "…description of how the design, systems, and operational characteristics of a plant, such as an advanced reactor, relate to a licensee's or applicant's organizational structure, staffing, and management framework." Although ConOps documents are employed by design organizations for developing systems, the NRC uses them as an information source for reasonably assuring that the intended human-system integration can properly support safe operations (NRC, 2012b, p. 6)

In Sandia National Laboratory's 2020 report on "Human Factors Considerations for Automating Microreactors," the foundational role of the concept of operations for advanced reactor designs was also discussed. In particular, the overall import and central nature of the Concept of Operations was illustrated as follows:

> The first step of designing a new system after operational needs have been identified is to develop a concept of operations. A Concept of Operations (ConOps) is an enterprise-level living document that helps define a conceptual view of a new system, particularly from the perspective of the user/operator. The ConOps indicates assumptions and intent of the system and can also act as a justification for why the system should exist (the need), how it meets stakeholder needs, the requirements that ensure those needs are met, as well as lifecycle information of the entire system. (p. 36)

There is currently no regulation that requires applicants to provide a docketed concept of operations as part of a licensing application. Until now, this has not been necessary for the operating fleet, largely because the ConOps is already well understood by the NRC staff based upon previous experience with large LWRs. However, new designs will likely conceive of radically different concepts of operations for which the staff may have little or no prior understanding. Therefore, there may be a need to explicitly make the concept of operations a part of the content of applications under the proposed Part 53 rule.

A description of the concept of operations that addresses the points above (and potentially other relevant topics) will help the NRC staff in the following ways:

1. It will reduce the opportunity for confusion. Without a concept of operations, the NRC staff reviewers may have to rely upon their knowledge of the concept of operations at large LWR plants, which may be inappropriate and lead to poor assumptions being made during reviews.
2. It will help the NRC staff to understand and confirm to what extent a design relies on the humans for safe operation.
3. It can be used to help determine the appropriate scope of the staff review.
4. It may reduce the need for Requests for Additional Information, audits, and other clarifying questions.

Staffing Analyses

For advanced reactors, it may be appropriate to allow applicants to propose their own alternative staffing models in lieu of taking a prescriptive approach under the regulation. This would entail, at a minimum, a license application including an HFE-based staffing analysis of sufficient scope and depth to allow for the NRC staff to make an adequate assessment of the acceptability of the proposed staffing levels. In adopting such an approach, the following issues would also need to be considered:

- A simulation facility may need to be required in order to facilitate the requisite HFE program elements necessary to support the staffing analysis. It is possible that such a simulator facility may not need to be of the "full scope" type that is required for existing power reactors; however, it would still need to be of sufficient scope to accommodate the requisite HFE analyses. *Separately, a simulator may also be necessary to support licensed operator training and examination programs, as discussed earlier.*
- The development and justification of alternative staffing models by advanced reactor applicants could be informed by the existing process of NUREG-1791 that was outlined earlier in this paper.
- It may be appropriate for the Part 53 rule to provide a prescriptive staffing model as an <u>option</u> for applicants that prefer not to conduct the staffing analyses needed to support an alternative, flexible staffing model.
- A well-defined process does not currently exist for applicants to propose the elimination of the STA position. However, as noted earlier, the following are some factors that, considered in aggregate, could potentially support such a proposal:
    - Licensed operator qualifications and training
    - Control room human-system interfaces;
    - Design features;
    - Limited reliance on human actions for safety;
    - Automated capabilities that supply additional defense-in-depth and reduce crew workload.

Human Factors Engineering Program

Advanced reactor applications are likely to need to contain specific information that is expected to be of an HFE programmatic nature. For example, certain aspects of an application, such as design of control room human-system interfaces or proposals for alternative staffing models, would be expected to be informed and guided by acceptable HFE principles. However, at present, the primary regulation[12] that explicitly requires the application of HFE is the post-TMI requirement of 50.34(f)(2)(iii), which (as noted earlier in this paper), directs applicants to "[p]rovide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts." This present regulatory framework for HFE creates two significant issues where advanced reactors are concerned:

1. The wording of the present regulation specifically limits its scope to the design of the control room. For advanced reactors, it is possible that a traditional control room will not be part of the facility. For example, important control functions (e.g., withdrawing control rods for a startup) might occur from a cabinet that is not located within what would generally be regarded as a control room. As a separate example, it is possible that an operator may have a mobile device that provides indications and control functions. Thus, for advanced reactors, it would be more appropriate to require HFE principles to be applied based upon "where HFE is needed," versus simply designating a location within the plant where such requirements apply.
2. HFE programmatic information should also be included within advanced reactor applications to the extent that such information is necessary to support all relevant portions of the application. Under an integrated approach to humans and systems (discussed earlier within this paper), this would, however, still constitute a more flexible approach than that which is currently used for large LWRs

Based upon these considerations, Part 53 should require advanced reactor applications to address the incorporation of state-of-the-art human factor principles more broadly than what is presently required under existing regulations. As previously identified in the discussion of proposed objectives for the Part 53 rule, an advanced reactor human factors engineering program should be adequate to ensure that humans (e.g., operators, supervisors, technicians, and other appropriate personnel) cannot only operate the plant, but also perform the full range of tasks necessary to ensure the continued availability of plant-specific safety functions in an effective manner. In some instances, it may be relevant to apply HFE considerations to, for example, those maintenance and testing activities that are related to plant safety functions. This scope of potential activities should be taken into account when considering the applicability of HFE requirements.

[12] It should be noted that 10CFR 52.47, "Contents of applications; technical information," does contain a pointer to the 50.34(f)(2)(iii) requirement as well. Specifically, 52.47(a)(8) requires that applications contain "[t]he information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v)."

Conclusion

This paper has considered possible approaches for a regulatory framework that addresses advanced reactor human-system considerations in a manner that is risk-informed, performance-based, and technology-inclusive. The NRC's existing human-system regulatory framework for power reactors is primarily based on operating experience with traditional, large LWRs, resulting in a system of regulations and regulatory guidance that may not be well aligned with certain key attributes that advanced reactors are expected to possess. In response to this, a new framework for better addressing some of these aspects of advanced reactor human-system considerations has been proposed. It is intended that, ultimately, the material in this paper will be used to facilitate further development of the Part 53 rule for advanced reactors, as well as the associated regulatory guidance.

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