Enclosure to NRC Staff Prepared White Paper
"Micro-Reactor Licensing and Deployment Considerations:
Fuel Loading and Operational Testing at a Factory"
August 2023 Draft – Released to Support ACRS Interaction

Technical, Licensing, and Policy Considerations for Factory-Fabricated Micro-Reactors

This enclosure includes various topics that are related to the licensing and deployment of factory-fabricated micro-reactors (and other technologies, as appropriate). Some of these are topics raised by developers through formal pre-application engagement with the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff and in other interactions, such as the periodic Advanced Reactor Stakeholder Meetings organized by the NRC staff. Some of the topics have previously been considered in SECY-20-0093, "Policy and Licensing Considerations Related to Micro-Reactors" (ML20129J985), and in the context of small modular reactors or non-light-water reactors, but are revisited here with the attributes of factory-fabricated micro-reactors and related deployment models in mind. The NRC staff will address design-specific issues on a case-by-case basis.

This enclosure also includes the NRC staff's near-term strategies and next steps for addressing each topic. The near-term strategies provide means for the NRC staff to address each topic under the existing regulatory framework without additional Commission direction. These are included to provide factory-fabricated micro-reactor developers and potential applicants with awareness of approaches that are available now. The next steps are focused on longer-term approaches that may involve additional Commission engagement in the future, as appropriate.

1. Considerations Related to Initial Fuel Load and Authorization to Operate at the Deployment Site for Reactors that Arrive Pre-Loaded with Fuel

<u>Deployment Model Considerations</u>

Deployment strategies that include loading fuel at a factory would result in fueled factory-fabricated modules arriving at the deployment site. However, several requirements in the Atomic Energy Act of 1954, as amended (AEA) and 10 CFR Parts 50 and 52 that are related to public notifications, the opportunity for hearing, and authorization to operate the facility under a combined license are premised on the initial loading of fuel at the deployment site. For example, AEA Section 189a.(1)(B)(i) requires the following:

> Not less than 180 days before the date scheduled for initial loading of fuel into a plant by a licensee that has been issued a combined construction permit and operating license under section 185b., the Commission shall publish in the Federal Register notice of intended operation. That notice shall provide that any

> person whose interest may be affected by operation of the plant, may within 60 days request the Commission to hold a hearing on whether the facility as constructed complies, or on completion will comply, with the acceptance criteria of the license.

In the case where a factory-fabricated module arrives at a deployment site for which a combined license has been issued, the fabricator would have loaded fuel at the factory under its license. Therefore, *at the deployment site* there would not be "initial loading of fuel into a plant by a licensee that has been issued a combined construction permit and operating license." This entry condition to the requirements in AEA Section 189a.(1)(B)(i) would not be satisfied as written, but it would be inconsistent with the law's purpose for the Commission not to publish the notice of intended operation and opportunity for the public to request a hearing on conformance with the acceptance criteria in the combined license for the deployment site. Thus, the NRC should find the closest analogue to initial fuel load by the combined license holder that also fulfills the underlying purpose of the law.[1]

In addition, AEA Section 185b. states, in part, that "[f]ollowing issuance of the combined license, the Commission shall ensure that the prescribed inspections, tests, and analyses are performed and, prior to operation of the facility, shall find that the prescribed acceptance criteria are met." The Commission has historically considered fuel loading to be part of operation, meaning that a reactor that arrives at the deployment site loaded with fuel would be "in operation" before the Commission makes the required finding that the prescribed acceptance criteria are met. As described in the options in this paper, the NRC staff proposes the use of features to preclude criticality to define fuel load under these circumstances as not being in operation and considering whether the removal of the criticality preclusion features would be the best analogue to the initial loading of fuel to which AEA Section 189a.(1)(B)(i) refers. If approved by the Commission, features to preclude criticality would avoid the situation in which a fueled reactor arrives at the deployment site and is already considered to be "in operation."

The regulations in 10 CFR Part 52, Subpart C contain requirements related to these provisions of the AEA. In relation to AEA Section 185b., the regulations in 10 CFR 52.103(g) require, in part, that "[t]he licensee shall not operate the facility until the Commission makes a finding that the acceptance criteria in the combined license are met…" In relation to AEA Section 189a.(1)(B)(i), the regulations in 10 CFR 52.103(a) require that "[t]he licensee shall notify the NRC of its scheduled date for initial loading of fuel no later than 270 days before the scheduled date and shall notify the NRC of updates to its schedule every 30 days thereafter." There are other regulations that reference initial fuel loading, discussed below. An applicant for or holder of a combined license could request exemptions from these regulations if the provisions, as written, cannot be reasonably complied with and the requested exemptions comply with the AEA. License conditions may be needed to address any regulatory gaps. If, as proposed by the NRC staff in this paper, a factory-fabricated module that includes features to preclude criticality is not in operation when loaded with fuel, then the requirements in AEA Section 185b. and 10 CFR 52.103(g) could be met as long as the module were not placed into operation (e.g., through the removal of features to preclude criticality) until after the Commission makes a

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[1] Section 189a.(1)(B)(v) of the AEA also refers to fuel load: "The Commission shall, to the maximum possible extent, render a decision on issues raised by the hearing request within 180 days of the publication of the notice provided by clause (i) or the anticipated date for initial loading of fuel into the reactor, whichever is later." The NRC staff understands "anticipated date for initial loading of fuel into the reactor" in context to refer to the scheduled date for initial loading of fuel by the COL holder that is discussed in AEA Section 189a.(1)(B)(i).

finding that the acceptance criteria in the combined license are met. This could also address reactors arriving at a deployment site for which a construction permit has been granted under 10 CFR Part 50 but not an operating license.

Near Term Strategy

If the Commission approves the NRC staff's proposed use of features to preclude criticality, then a factory-fabricated module will not be in operation during transportation to the deployment site or upon arrival. If directed by the Commission under Option 1b in this paper, the NRC staff could use the removal of these features as an analogue to initial fuel loading by the combined license holder for the purposes of the notification and opportunity for hearing in AEA Section 189a.(1)(B)(i) and as the equivalent of the commencement of operation for the purpose of AEA Section 185b. The NRC staff notes that "the removal of features to preclude criticality" and "initial loading of fuel" are both distinct actions performed by the combined license holder that put a fully constructed utilization facility in a position to sustain a nuclear chain reaction. In both cases, the utilization facility is incapable of sustaining a nuclear chain reaction (for lack of sufficient reactivity) before operation is authorized. Operation is not authorized until the Commission finds that the prescribed acceptance criteria are met, and the public is given an associated opportunity to request a hearing on whether the facility satisfies the acceptance criteria.

For fueled reactors being deployed on sites for which a construction permit has been granted under 10 CFR Part 50, the NRC staff could also use removal of features to preclude criticality as an analogue to initial fuel loading and the beginning of operation and would authorize removal of the features only after an operating license had been granted.

There are numerous other regulations including 10 CFR 50.47, 10 CFR 50.54, 10 CFR 50.55a, 10 CFR 50.71, 10 CFR 50.75, 10 CFR 50.120, 10 CFR Part 50 Appendix E, 10 CFR Part 50 Appendix J, 10 CFR 52.99, and other provisions in 10 CFR 52.103 where initial fuel load is used as a milestone. In these cases, the requirements to be implemented at fuel load are not linked to specific language in the AEA. Therefore, the NRC staff could use the "removal of features to preclude criticality" as an analogue to "initial loading of fuel" and could implement this approach through exemptions and license conditions, as appropriate, in the near term.

Next Steps

Depending on Commission direction on Option 1b in this paper, the NRC staff will consider whether additional actions are warranted related to initial fuel load and authorization to operate at the deployment site for reactors that arrive pre-loaded with fuel

2. Timeframe for Authorization to Operate at the Deployment Site

Deployment Model Considerations

Factory-fabricated micro-reactors may have significantly simpler and shorter duration construction activities at the deployment site than large light-water reactors, which typically take several years to construct. Factory-fabricated micro-reactors that are "of a self-contained" design with the nuclear and balance-of-plant systems in one or a few containers that are fully fabricated at the factory would likely require only simple construction activities at the deployment site (e.g., pouring a small concrete pad on which to place the container housing the reactor).

This type of reactor might be ready for operation within days to weeks of receipt of the construction permit or combined license for the deployment site if construction begins immediately after license issuance. Factory-fabricated micro-reactor designs that have a core module design and would require more complex construction activities at the deployment site (but still much simpler than construction activities for large light-water reactors), such as erecting a reactor building and installing power conversion equipment, might be ready for operation within a few months of the start of construction. In either case, a key aspect of factory-fabricated micro-reactor deployment models is the ability to move a factory-fabricated module from the factory to the deployment site and place it into operation as a nuclear power plant in a much shorter time than it takes to construct a large light-water reactor at the intended site of operation.

Factory-fabricated micro-reactors may be licensed at the deployment site under either 10 CFR Part 50 or 52. In both cases, a mandatory hearing would be held as part of the process for issuing the authorization to construct the reactor at the deployment site. Factory-fabricated micro-reactors would likely have standardized designs that may be described in a referenced 10 CFR Part 52 design certification or manufacturing license, which could reduce the scope of NRC review for deployment site licensing. Also, any final safety findings on final design information in a construction permit application would be incorporated in the permit in accordance with 10 CFR 50.35(b) and subject to the backfitting requirements in 10 CFR 50.109.

Under a 10 CFR Part 50 approach, an additional opportunity for hearing is required by AEA Section 189a.(1)(A) in conjunction with issuance of the facility operating license. The regulations in 10 CFR 2.309(b)(3) provide a 60-day opportunity to request a hearing (although AEA Section 189a.(1)(A) requires 30 days' notice). The potential scope of such a hearing would be the entirety of the operating license application, but the hearing scope would be reduced to the extent the operating license application references an earlier NRC license or approval providing finality on the matters resolved therein, such as a manufacturing license. Subparts C and L and Appendix B of 10 CFR Part 2 provide the rules of general applicability, procedures, and model milestones for such a hearing. If no hearing is requested, the Commission could issue the operating license immediately upon the closure of the 60-day hearing request period, provided all other requirements are met. In the case that a hearing is requested, the issuance of the license would have to wait until a Commission decision to not grant the hearing request or else the completion of the hearing if the request is granted,[2] either of which could take many months according to the procedures and model milestones in 10 CFR Part 2.

The environmental review may also affect the timeframe for deployment under the 10 CFR Part 50 licensing process. After issuing a permit to construct a nuclear power reactor with a supporting environmental impact statement (EIS), the regulations in 10 CFR 51.20(b)(2) and 51.95(b) require that the NRC staff publish a supplement to the construction permit EIS to support issuance of the operating license. The process for preparation and publication of the supplement to the EIS includes publication of a draft supplement with an additional period for public comment, and various consultations with external stakeholders. In addition, the National Environmental Policy Act (NEPA) was amended in June 2023, to include a new requirement to issue EISs within 24 months. Recent improvements in the environmental review process along

[2] AEA Section 189a.(1)(A) states, in part, that "[i]n cases where such a construction permit has been issued following the holding of such a hearing, the Commission may*, in the absence of a request therefor by any person whose interest may be affected*, issue an operating license … without a hearing, but upon thirty days' notice and publication once in the Federal Register of its intent to do so" (emphasis added).

with standardized, relatively simple reactor designs, and small reactor sites could reduce the time to complete a supplemental EIS to less than 24 months, with further process improvements planned. This is in contrast to the environmental review for a combined license, in which the environmental review is completed at the time of issuance and a supplement to the EIS is not required in connection with the 10 CFR 52.103(g) finding and the authorization to operate and would not contribute to the deployment timeframe.

Under a 10 CFR Part 52 combined license, an opportunity for hearing is required by 10 CFR 52.103(a) and AEA Section 189a.(1)(B)(i) “on whether the facility as constructed complies, or on completion will comply, with the acceptance criteria of the [combined] license.” Both 10 CFR 52.103(a) and AEA Section 189a.(1)(B)(i) require the NRC to publish a notice of intended operation providing this hearing opportunity at least 180 days before the scheduled date for initial loading of fuel (or removal of features to preclude criticality if the Commission approves Option 1b in this paper) by the combined license holder and specify a 60-day period for the opportunity to request a hearing. The potential scope of such a hearing would be limited to the inspections, tests, analyses, and acceptance criteria (ITAAC) included in the combined license.

NRC regulations in 10 CFR Part 52, Subpart C include additional timing requirements written in terms of the initial loading of fuel under a combined license:

- Under 10 CFR 52.103, the licensee shall notify the NRC of its scheduled date for initial loading of fuel no later than 270 days before the scheduled date.
- The regulations in 10 CFR 52.99 include requirements on the timing of licensee notifications of ITAAC closure and completion and NRC publication of related notices. In particular, 10 CFR 52.99(c)(3) provides that if the licensee has not provided an ITAAC closure notification under 10 CFR 52.99(c)(1) for all ITAAC by 225 days before the scheduled date for initial loading of fuel, then the licensee must provide an uncompleted ITAAC notification no later than 225 days before the scheduled initial fuel load date to describe how the licensee will complete the uncompleted ITAAC. The intent of this requirement, in part, is to ensure that information related to ITAAC closure is available to the NRC staff and the public at the time the Commission publishes the 60-day notice of opportunity for hearing in the *Federal Register* as required by AEA Section 189a.(1)(B)(i).

These regulations and the final ITAAC hearing procedures published on July 1, 2016 (Volume 81 of the *Federal Register* (FR), page 43266 (81 FR 43266)) were developed to ensure the Commission will meet the requirements of AEA Section 189a.(1)(B)(v) and 10 CFR 52.103(e), which provide that the Commission shall, to the maximum possible extent, render a decision on issues raised by the hearing request within 180 days of the publication of the notice of intended operation or the anticipated date for initial loading of fuel into the reactor, whichever is later.

For large light-water reactors, the timeframes for licensee notifications and Commission actions required by AEA Section 189 and Subpart C of 10 CFR Part 52 fit within the overall construction schedule, which is usually several years. For factory-fabricated micro-reactors that can be deployed in a matter of days to a few months, these timeframes will likely result in delays in entering the reactor into operation. If the licensee were to notify the Commission of the intended date of initial fuel load upon receipt of the combined license, that notification would start the 270-day period. By the end of that 270-day period, the Commission would normally complete the hearing, if requested and granted, and be able to determine whether it can make the 10 CFR 52.103(g) finding. Based on the 270-day notification by the licensee, the Commission would have 90 days to publish the notice of intended operation and 60-day opportunity to request a hearing required by AEA Section 189a.(1)(B)(i). Publication of this notice would start a 180-day period to normally complete the hearing process per AEA Section 189a.(1)(B)(v). However, as discussed in the NRC's ITAAC hearing procedures, the Commission has established a goal to publish the notice of intended operation 210 days before scheduled fuel load.

Notwithstanding the Commission's obligations under AEA Section 189a.(1)(B)(i), the Commission has previously established that a licensee may under certain conditions begin operation before the scheduled date of initial fuel loading submitted to the Commission under 10 CFR 52.103(a). The NRC stated at 81 FR 43273 in the publication of the final ITAAC hearing procedures that "the licensee can, consistent with 10 CFR 52.103(a), move up its scheduled fuel load date after the notice of intended operation is published. Such a contraction in the licensee's fuel load schedule would have no effect on the hearing schedule, but as a practical matter, the NRC would consider such a contraction in the licensee's schedule as part of its process for making the 10 CFR 52.103(g) finding and the adequate protection determination for interim operation." In its "Comment Summary Report – Procedures for Conducting Hearings on Whether Acceptance Criteria in Combined Licenses Are Met" (ML16167A464), Section 5, "Hearing Tracks and Schedules," Subsection G, "Contraction of Fuel Load Schedule," the NRC stated that in the absence of a hearing or if the hearing issues are resolved early in favor of the licensee, the licensee will be allowed to operate if and after the 10 CFR 52.103(g) finding is made. The NRC also stated in the Comment Summary Report that if a hearing is held and has not been completed, but the NRC staff has made the 10 CFR 52.103(g) finding and the Commission has made the adequate protection determination for interim operation, then the licensee will be allowed to enter into interim operation.

Near Term Strategy

The NRC staff intends to use the existing regulations in 10 CFR Parts 2, 50, and 52 in connection with issuing operating licenses and authorizing operation under a combined license. The NRC staff also intends to use the existing final ITAAC hearing procedures.

Several steps may be taken to potentially shorten the timeframe for deployment of a factory-fabricated micro-reactor under a combined license. A key strategy would be to publish the notice of intended operation as early as possible. The NRC could not publish this notice before combined license issuance because AEA Section 189a.(1)(B)(i)-(ii) provides that the hearing opportunity is on conformance with the acceptance criteria *in the combined license*. However, a licensee could provide the 10 CFR 52.103(a) notification of its scheduled date for initial fuel load[3] and the 10 CFR 52.99(c) ITAAC closure notifications and uncompleted ITAAC notifications for all ITAAC immediately upon receipt of the combined license. If the combined license applicant intended to do this, it should inform the NRC staff of its intention in the combined license application or by other means so that the NRC staff could make necessary arrangements to prepare the notice of opportunity for hearing. Consistent with the NRC's experience with the Vogtle ITAAC proceedings, the NRC could make publicly available the

[3] If fuel is loaded at the manufacturing facility, then the licensee would not be able to notify the NRC "of its scheduled date for initial loading of fuel" as required by 10 CFR 52.103(a) because the manufacturer would be loading fuel at its site under its license rather than the licensee for the deployment site loading fuel at the deployment site under its license. In that circumstance, the licensee could alternatively provide a schedule for the removal of the features to preclude criticality proposed by the NRC staff in this paper.

uncompleted ITAAC notification and publish the notice of intended operation within about 15 days after receipt of the uncompleted ITAAC notification.

During the 60-day opportunity to request a hearing, the licensee would presumably complete construction of the reactor, provide the appropriate notifications related to ITAAC required by 10 CFR 52.99(c), and notify the NRC of its update to the date scheduled for initial fuel load in accordance with 10 CFR 52.103(a). If the NRC staff is able to conclude that all acceptance criteria are met by the close of the 60-day period for requesting a hearing and no hearing was requested, the NRC staff would aim to make the 10 CFR 52.103(g) finding shortly thereafter, possibly within 5 days. This could result in a deployment timeframe of as little as about 80 days after combined license issuance in ideal circumstances, which may be driven primarily by the statutory requirement in AEA Section 189a.(1)(B)(i) that the notice of intended operation "provide that any person whose interest may be affected by operation of the plant, may within 60 days request the Commission to hold a hearing…." In cases where a hearing is requested, the minimum timeframe would be extended in accordance with the final ITAAC hearing procedures, i.e., the 10 CFR 52.103(g) finding might be issued (1) after the Commission's decision on the hearing request if the request is denied, (2) after a decision allowing interim operation if the hearing request is granted and the requirements for interim operation are met, or otherwise, (3) after the presiding officer has issued the decision after hearing.

If a hearing is requested, the regulations and ITAAC hearing procedures allow the licensee and the NRC staff 25 days to answer the hearing request and establish a milestone of 30 days after the answers for a Commission ruling on the hearing request. If the Commission does not grant a hearing request, this would add 55 days to the minimum deployment timeframe compared to the scenario in which no hearing request is filed (135 days total). If the Commission does grant the hearing request, then the minimum timeframe could be extended by an additional 70 to 94 days (205 to 229 days total), although interim operation may be allowed during the hearing if the Commission makes the adequate protection determination for interim operation and the NRC staff is able to make the 10 CFR 52.103(g) finding.

Under 10 CFR Part 50, the Commission would notice the opportunity for hearing in conjunction with its notice docketing the application for the operating license for the deployment site. The NRC staff would then complete its final safety evaluation during the 60-day period of the opportunity to request a hearing. This assumes that the final design, site-specific issues, technical specifications, and operational programs would have been reviewed and approved during the construction permit proceedings or other prior approvals (e.g., topical reports) and the operating license application doesn't introduce any deviations. In cases where a hearing was not requested and all other requirements are met, the deployment timeframe could be shortened to approximately 95 days, which accounts for 30 days to perform an acceptance review and docket the application, 5 days to publish the notice of opportunity for hearing, and 60 days for the opportunity to request a hearing. In cases where a hearing is requested, the timeframe will be extended in accordance with the regulations in 10 CFR Part 2. However, under the 10 CFR Part 50 process for issuing a facility operating license, the record of decision cannot be issued until both the safety and environmental reviews are completed. Characteristics of factory-fabricated micro-reactors, such as standardized designs and relatively small site footprints with limited construction activities at the deployment site, may allow for the NRC staff to complete the required supplement to the EIS in less than 24 months under the current process. In accordance with SRM-SECY-21-0001, "Rulemaking Plan —Transforming the NRC's Environmental Review Process" (ML22109A171), after completing several environmental reviews for advanced reactors, the NRC staff could further explore the idea of preparing environmental assessments to meet NEPA requirements for some categories and subcategories of license applications presently falling within the scope of 10 CFR 51.20(b), and present options to the Commission. If such a proposal was developed and approved, environmental assessments could be used both for the construction permit and the operating license at the deployment site, which would require appropriate changes to 10 CFR Part 51 or exemptions. The use of environmental assessments instead of EISs could substantially shorten the timeline such that the deployment timeframe could be dictated by the operating license contested hearing process.

Despite the differences in the requirements for opportunities for hearings for issuance of an operating license or making the 10 CFR 52.103(g) finding, the NRC staff hasn't found any reasons that either the 10 CFR Part 50 or Part 52 processes would necessarily result in significantly different deployment timeframes, except for the time needed to issue an EIS supplement to support issuance of an operating license. Ultimately, it will be up to an applicant to consider the differences in the two licensing processes and decide which one is better suited to its particular deployment model.

Next Steps

The NRC staff intends to further assess the final ITAAC hearing procedures and the requirements in 10 CFR Part 2 based on Commission direction on the options presented in this paper for features to preclude criticality, fuel loading, and operational testing; the characteristics of factory-fabricated micro-reactors; and further stakeholder input. If warranted, the NRC staff will propose an update to the final ITAAC hearing procedures for Commission consideration and consider rulemaking options, as appropriate, to ensure that hearings would not result in unnecessary delays to operation of factory-fabricated micro-reactors. The NRC staff is considering whether there are opportunities to streamline the process for preparing and publishing the supplement to the EIS if licensing under 10 CFR Part 50 is pursued by a potential applicant.

3. Replacement of Factory-Fabricated Modules at the Deployment Site

Deployment Model Considerations

Factory-fabricated micro-reactor deployment models might include periodically removing factory-fabricated modules from the deployment site at the end of their operational lives or fuel cycles and replacing them with modules of the same design.[4] This could involve shipping a factory-fabricated module away from the deployment site for refueling and refurbishment and then returning it to the deployment site or shipping a new module to the deployment site to replace the existing one. The replacement of factory-fabricated modules could result in the need for the deployment site licensee to have multiple fueled modules on site at some times to allow for transition from the operating module to the replacement module with minimal downtime.

[4] The NRC staff is aware that the designs of factory-fabricated micro-reactors could evolve over the lifetime of a single module such that when the time comes to replace a module, the replacement may be of a different design. In this case the, the fabricator would have amended its manufacturing license or obtained a new manufacturing license for the new design. Likewise, the deployment site licensee would have to amend its permits and licenses to account for the new design or obtain new permits and licenses.

The NRC staff has considered potential licensing strategies for replacement modules under the current regulatory framework. The NRC staff previously addressed licensing options for multi-module facilities in SECY-11-0079, "License Structure for Multi-module Facilities Related to Small Modular Nuclear Power Reactors," dated June 12, 2011 (ML110620459). A key difference from the small modular reactors (SMRs) considered in SECY-11-0079 is that the replacement factory-fabricated micro-reactor modules are not intended to be operated at the same time as the module it is intended to replace. Each module would operate for limited time (generally less than 10 years) and then would be removed from service and replaced. Upon removal from service, the deployment site licensee would install features to preclude criticality to take the module out of operation and store the module on site prior to decommissioning or shipment to a decommissioning facility or a refurbishment and refueling facility.

The NRC staff has identified a licensing strategy under 10 CFR Part 52 that would have the initial combined license application include one final safety analysis report to address the requirements of 10 CFR 52.79 for all factory-fabricated modules, including replacement modules, anticipated to be operated at the deployment site. The application would also specify the number of modules that could be present at the site simultaneously and operated simultaneously. The combined license application would also include any permanently installed site-specific features such as power conversion systems or structures, if applicable. Under this approach, each module would receive its own combined license; however the licenses would be issued concurrently. This would be similar to "Alternative 3: Individual Reactor Module Licenses" described in SECY-11-0079, which was the NRC staff's preferred alternative as the best approach for the licensing of multi-module power reactor facilities. As noted in SECY-11-0079:

> Consistent with NRC regulations and existing practice, a [combined license] application related to multiple modules at a single facility can undergo a single license review, safety evaluation report (SER), and hearing if a single license application is made for modules of essentially the same design. The precedent for this process comes from recent large light-water reactor [combined license] applications that have been filed under 10 CFR Part 52 for two units (e.g., Vogtle Electric Generating Plant), and many [construction permits] and [operating licenses] issued under 10 CFR Part 50…
>
> NRC regulations related to ITAAC (10 CFR 52.103(g)) adequately address the transition from construction to operation under 10 CFR Part 52 by allowing separate findings for each module. The individual license for each module would also support the transition from construction to operation under 10 CFR Part 50 by allowing the issuance of separate [operating licenses] at different times for each module (which has been the historical practice for [construction permits] issued for multiunit sites).

The EIS and hearing(s) required for combined license issuance would address all the factory-fabricated modules to be licensed to operate over the life of the deployment site. The licensee would be required to show that all ITAAC have been met and the Commission would need to issue its finding under 10 CFR 52.103(g) (and offer the associated opportunity for hearing) before the first module and every subsequent replacement module would be authorized to begin operation under its combined license. There are potential timing impacts associated with completing ITAAC and potential ITAAC hearings for each replacement module, but the license application, safety review, mandatory hearing, and opportunity for contested hearing on the combined license issuance would only be conducted once for all combined licenses for the modules anticipated to be operated at the facility. The NRC staff expects that licensees would know in advance of the need for the replacement module and therefore could plan accordingly to minimize potential impacts on plant downtime caused by the time required for ITAAC completion, a potential ITAAC hearing, and the Commission's finding required by 10 CFR 52.103(g).

For example, depending on the design of the factory-fabricated module and the installation process, the licensee could bring the replacement module onsite, provide appropriate ITAAC notifications to the NRC, and complete the ITAAC hearing process well before the currently operating module reached the end of its life or fuel cycle. If timed appropriately, this could provide time for the Commission to make the finding required by 10 CFR 52.103(g) before the licensee removed the currently operating module from service. Depending on the deployment facility and the licensee's plans for operation, the final safety analysis report may need to account for additional operating modules at the site, or conditions in the combined license for the replacement module may need to specify that the replacement module would not be placed into operation unless there were no other operating modules at the site.

Under 10 CFR Part 50, the NRC could issue one construction permit covering the construction of the facility (i.e., the factory-fabricated module and the on-site balance-of-plant) including all replacement modules anticipated to be deployed at the site. Issuance of the construction permit would involve a mandatory hearing, opportunity for contested hearing on the construction permit issuance, and an EIS that would consider all of the modules anticipated to be deployed at the site. The NRC could then issue separate operating licenses to authorize operation of each replacement module after they were installed at the deployment site. To the extent that the first module and all replacements would have the same standard design and any permanent on-site structures or features would have been approved with the issuance of the operating license for the first module at the deployment site, the NRC staff could leverage the safety review that had already been completed for subsequent operating licenses. Issuance of the operating license would require the NRC to provide an opportunity for the public to request a hearing and to publish a supplement to the final EIS for the construction permit as required by 10 CFR 51.20(b)(2) and 51.95(b). Per 10 CFR 51.95(b), the supplement to the EIS would only cover matters that differ from the final EIS for the construction permit or that reflect significant new information concerning matters discussed in that final EIS.

Under the 10 CFR Part 50 approach, the NRC staff expects that depending on the design of the factory-fabricated module, the potential downtime could be minimized by the operating license application being submitted well in advance of removing the currently operating module from service, which should allow the EIS to be supplemented and the contested hearing process to be completed. However, the process for supplementing the EIS and the opportunity for a contested hearing to support issuing the operating license for the replacement module introduce the potential for new issues to arise. This is in contrast to the approach using 10 CFR Part 52 under which the design and environmental reviews are completed and given finality upon issuance of the combined licenses.

Near Term Strategy

The NRC staff intends to use the existing regulations in 10 CFR Parts 50 and 52, as informed by SECY-11-0079 and the approach described above, for licensing replacement of factory-fabricated modules. Factory-fabricated micro-reactor developers and potential applicants will need to consider factors such as the number of modules that will be onsite at one time, the expected operational states of the onsite modules, replacement frequency, and others when deciding whether to apply for licensing of replacement modules under 10 CFR Part 50 or 52.

Next Steps

The NRC staff intends to continue to engage with stakeholders in pre-application activities and public meetings to further understand and assess planned deployment models, the potential to streamline licensing pathways, and the need for additional guidance under the current regulatory framework. The NRC staff will also consider whether other licensing approaches for multi-module sites described in SECY-11-0079, such as a single license for multiple modules or the "master facility license" alternative, could provide efficiencies for licensing replacement modules at the deployment site. The NRC staff will continue to monitor developers' and potential applicants' plans to assess the viability of proposed strategies and any potential policy issues needing further Commission engagement. If the NRC staff determines that alternative licensing strategies would require rulemaking or a policy decision, this would be addressed in a separate vote paper to seek Commission direction.

4. Autonomous Operation and Remote Operation

Deployment Model Considerations

During recent pre-application interactions with the NRC staff, significant interest has been expressed by micro-reactor developers regarding the inclusion of autonomous[5] and remote operational characteristics within their proposed designs. A remote operational model tends to center around the minimization of the numbers of both operators and other categories of facility staffing at the facility site, while an autonomous operational model would seek to eliminate reliance upon the use of operators. For the purposes of this paper, autonomous systems are considered those "…able to perform their task and achieve their functions independently (of the human operator), perform well under significant uncertainties for extended periods of time with limited or nonexistent communication, with the ability to compensate for failures, all without external intervention."[6] In the case of remote operations, the objective is to relocate staff to a centralized location and operate some reactors remotely. Such approaches differ dramatically from the current paradigm of commercial nuclear plant operations in which operators are required by 10 CFR 50.54(k) and (m) to maintain a continual onsite presence in control rooms. Furthermore, such operational approaches may entail the elimination of a main control room at the facility, thereby requiring the evaluation of requested exemptions from relevant regulations, such as 10 CFR 50.34(f)(2)(iii).[7] Thus, proposed deployment models that would involve relocating operators offsite or eliminating them entirely present a number of significant differences from traditional reactor designs and licensing paradigms.

As previously noted in SECY-20-0093, both autonomous and remote operations raise potential policy-related matters. For example, autonomous operation would entail reactivity manipulations being performed by automation rather than licensed operators, as well as potentially eliminating

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[5] As used within the context of this paper, the terms automation and autonomous have distinct meanings. Automation refers to automated processes. The term automated, in turn, is defined herein as the independent performance of tasks via the application of technology and absent continuous input from an operator. It should be noted that the proposed 10 CFR Part 53 rulemaking would define "automation" as "a device or system that accomplishes (partially or fully) a function or task."

[6] M. R. Endsley, "From here to autonomy: lessons learned from human–automation research," *Human factors*, vol. 59, no. 1, pp. 5–27, 2017.

[7] The regulations at 10 CFR 50.34(f)(2)(iii) require, in part, that applicants provide for NRC review a control room design that meets state-of-the-art human factors engineering principles.

humans as a layer of defense-in-depth. Separately, remote operations would require the NRC staff to reassess current requirements for the application of human factors engineering (HFE). Historically, operators could be expected to be able to take advantage of being co-located with the reactor facility in order to receive sensory feedback (e.g., noise, vibrations, local observation of conditions, etc.) that would serve to augment the information otherwise provided to them through the plant's instrumentation and control interfaces. Such information can be very useful in conditions of instrumentation and control failures, particularly in instances where highly automated systems are involved. Autonomous and remote operations approaches represent a shift in operator capabilities that need to be carefully considered and are areas in which the NRC staff has been working to develop the needed HFE tools as part of broader efforts at developing an HFE framework for the review of advanced reactor designs.[8, 9]

Autonomous Operations

Tasks may be fully or semi-automated, creating variations in the required degree of human oversight and control. It is important to note that increased use of automation is distinct from autonomy. Autonomy is considered to be the ability to operate with complete independence from human control, while automation refers to the machine execution of what were formerly human tasks.[10] Thus, autonomous operation may not rely on high levels of automation and could be achieved via simplicity (e.g., reliance upon inherent safety characteristics and robust passive systems) of an advanced reactor design.[11] The ability of a given design to demonstrate autonomy in its safety performance could also be a significant factor in justifying a remote operational concept for a micro-reactor facility.

The term "autonomous operation" does not have a commonly accepted definition in the nuclear industry. For example, a designer may potentially refer to a system as being "autonomous" or "fully automated" without the inclusion of any artificial intelligence while another party might assume that autonomy implies the inclusion of artificial intelligence.[12] Autonomous systems can, when the necessary capabilities are provided, potentially respond to situations beyond those explicitly programmed or anticipated in the design. Thus, autonomous systems can be capable of a certain amount of self-directed behavior and potentially act as a proxy for humans in decision-making situations.[13] Such functionality (i.e., the incorporation of artificial intelligence)

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8 See, e.g., SECY-20-0093.

9 As automation is inherently referenced to those tasks historically performed by humans, technological progress tends to gradually influence what may or may not be considered to represent advances in automation. Refer to Parasuraman, R., & Riley, V. (1997). Humans and automation: Use, misuse, disuse, abuse. Human Factors, 39(2), 230–253.

10 NUREG-2261, "Artificial Intelligence Strategic Plan Fiscal Years 2023-2027" (ML23132A305), includes a notional framework for artificial intelligence and autonomy levels in commercial nuclear activities. However, it should be noted that autonomy is generally considered within the context of artificial intelligence (e.g., machine learning) by that document. For that reason, this paper relies upon the definition of "autonomous" that was presented earlier during the discussion of "autonomous systems" in section four of this enclosure.

11 The concept of autonomy (particularly where safety performance is concerned) potentially not being achieved by complex automation but rather via simple, robust, and highly reliable safety features and characteristics runs counter to commonly used automation hierarchies that tend to instead represent autonomous operation as consisting of a very high degree of automation.

12 Gaining clarity on industry deployment concepts for autonomous operation will be an area of focus under the "Next Steps" portion of this section.

13 National Academies of Sciences, Engineering, and Medicine. 2022. *Human-AI Teaming: State-of-the-Art and Research Needs*. Washington, DC: The National Academies Press. https://doi.org/10.17226/26355.

will present new considerations for both developers and the NRC staff because it will introduce the potential for automation to potentially take actions other than those that were originally assumed during the design and licensing processes. Furthermore, the acceptability of potentially allowing AI-driven automation to control safety-significant operations (e.g., those needed to mitigate accidents) represents a presently unresolved matter and an area of ongoing discussion amongst designers, researchers, the NRC staff alike.

Remote Operations and Remote Monitoring

One currently envisioned use for micro-reactors is electrical power generation on micro-grids, including instances where the micro-reactor is the primary source of power, as well as those where it supplies the grid in parallel with other generation assets. Such uses would benefit substantially from the ability to let grid demand directly change reactor power without a human operator serving as an intermediary, such as by permitting grid control centers staffed by non-NRC-licensed individuals to directly control the electrical output of the micro-reactor facility. However, load-following operation in which a non-licensed individual modifies the power level of a nuclear reactor is precluded by the current NRC regulations. As background, AEA Section 11r. defines operators as individuals who manipulate the controls of utilization or production facilities. The AEA then mandates under Section 107 that individuals who operate “controls” must be licensed by the NRC. Notably, the AEA does not define what those “controls” consist of, thus affording the NRC the discretion to establish that definition via regulation. Both 10 CFR 50.2 and 55.4 define these “controls” (when used within the context of nuclear reactors) as consisting of apparatuses and mechanisms that directly affect the reactivity or power level of the reactor when manipulated. From the inception of operator licensing in 1956, manipulation of the controls of a utilization facility has been restricted to licensed operators under 10 CFR 50.54(i).

The NRC staff anticipates micro-reactor applicants will propose to operate (or in the case of autonomous reactors, monitor) one or more micro-reactor units from a remote location. Such cases raise the question of what technological requirements would be necessary to provide for the reliable and secure monitoring and control of one or more micro-reactor units from a remote location. For example, commercial large light-water reactor licensees have historically credited human actions for performing certain time-critical operations to meet accident analysis assumptions. Any suitable approach to remote operations would, logically, need to either provide a very high degree of assurance in the ability of operators to remotely accomplish such actions or, alternatively, eliminate reliance upon such actions for the achievement of safety functions. Two further considerations need to be taken into account here as well. First, under a remote operations approach, the absence of operators on site may potentially remove any opportunity for local, backup actions should remote operations be unsuccessful for any reason. Secondly, the viability of any remote operations approach would be predicated on the ability of developers to adequately address the needed cybersecurity considerations for remote operations. This latter point will be explored in greater depth within this section.

The remote operation of commercial nuclear power plants, has not yet been explored extensively from a practical implementation standpoint, and therefore there is a paucity of operating experience to draw from to inform future approaches to remote operations. The feasibility of an unattended light-water reactor design was studied in the early 1960s and a determination was reached that the concept was dependent on whether safety systems could be designed to a level of reliability that was high enough to preclude the need for regular maintenance, thereby necessitating a relatively simple design.[14] At present, areas of regulatory guidance needed to further develop the concept remain. Centrally, these include resolving issues regarding what design attributes would be necessary to support a safety determination for a remotely operated reactor facility. For example, the potential for loss of remote control capability may warrant requiring remotely operated reactors to meet technological criteria comparable to those proposed for “self-reliant-mitigation facilities” under 10 CFR Part 53 based upon a need to robustly demonstrate safety in the absence of any opportunity for human intervention.

Instrumentation and Control

A systematic, comprehensive assessment of potential internal and external hazards, including the potential for human-induced events, and their consequences must be performed as part of the plant’s safety analysis. Such an assessment should factor in any potential hazards stemming from the introduction of autonomous and/or remote operational instrumentation and control (I&C), which must be identified, analyzed, and appropriately addressed (e.g., prevented or mitigated) as part of the I&C design for safety. When I&C systems are relied upon to satisfy the overall nuclear power plant performance objectives, the design of the risk significant I&C systems must meet applicable regulations for safety. The I&C design, used for functions such as sensing, control, display, and monitoring of the plant, should be sufficiently reliable and robust commensurate with its safety significance as required by the regulations.

The applicable regulations, such as those under 10 CFR Part 50 or 52, for I&C are generally technology inclusive and performance-based as they primarily require the adequate demonstration of reliability (e.g., testing, surveillance, fail-safe design, and quality) and robustness (e.g., redundancy, independence, diversity, defense-in-depth, and deterministic behavior, and qualification) independent of the I&C technologies and provide licensees with flexibility to determine how to meet the established performance criteria. Similarly, NRC staff guidance that is risk-informed, performance-based, and technology-inclusive will be used to assess whether the applicant demonstrates how the specified I&C systems support the overall nuclear power plant performance objectives for a particular plant design. For example, the NRC staff has developed Design Review Guide (DRG) for I&C of Non-Light-Water Reactors (ML21011A140), which provides guidance for the NRC staff to use in reviewing the I&C portions of applications for advanced non-light-water reactors that follow Regulatory Guide (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” (ML20091L698), within the bounds of existing regulations.

Some of the biggest technical challenges faced by micro-reactor developers will be adapting or developing measurement processes to operate in significantly more cramped or inaccessible spaces limiting maintenance access. Furthermore, I&C equipment must be rugged enough to handle not only the high temperatures and high pressures in some advanced reactor designs but also the long-term effects of the coolant on the sensor interface. Corrosivity as well as high-temperature coolants will affect the operative lifespan of I&C equipment as well as contribute to measurement uncertainty. Micro-reactors that operate with thermal or fast neutron spectra may also have different sensor/actuator requirements due to a wide variety of fuels, higher-operating

[14] Rosenthal, M.W., et al., “The Feasibility of an Unattended Nuclear Power Plant,” Oak Ridge National Laboratory Report, ORNL-2985, August 1960.

temperatures, and flexible operation modes. These issues may pose challenges for I&C equipment that may need to be addressed in additional regulatory guidance.

Cybersecurity

The current power reactor cybersecurity requirements in 10 CFR 73.54 extend to 1) digital computer and communication systems and networks that are associated with safety-related, important-to-safety, security, and emergency preparedness (SSEP) functions and 2) support systems and equipment that, if compromised, could adversely impact SSEP functions. To implement the requirements, licensees identify critical digital assets (CDAs) that must be protected against cyberattacks.10 CFR 73.54 does not specifically address autonomous or remote operations; however, the performance-based nature of the regulation supports both non-autonomous, autonomous, and remote operations through application of appropriate cybersecurity considerations in a licensee's cybersecurity plan. Under 10 CFR 73.54(a), applicants for an operating license under the provisions of 10 CFR Part 50 and holders of a combined license under the provisions of 10 CFR Part 52 would be required to protect the security posture of an autonomous and/or remotely-operated system with the same level of assurance applied to non-autonomous and locally-operated digital computer and communication systems and networks.

Data communication pathways would be within the scope of digital computer and communication systems and networks required to be protected under 10 CFR 73.54(a) and would require protection against cyberattacks, up to and including the design basis threat as described in 10 CFR 73.1. Depending on the level of autonomy, micro-reactor designers would likely propose employing data connections using wired, wireless, or a combination of both pathways to communicate with critical systems and CDAs. Technical security controls in current nuclear power plant licensee cybersecurity plans prohibit the use of wireless technology for CDAs associated with safety-related and important-to-safety functions. A defensive computer security architecture that employs any type of remote access could be vulnerable to cyberattacks, such as unauthorized remote access, complete denial of service, and/or denial of authorized remote access functions. The level of autonomy, remote operations, and remote monitoring are important aspects in understanding the associated cybersecurity risks. A defensive computer security architecture that incorporates remote operation technology would have to be more complex than those of current licensees to ensure the confidentiality, integrity, and availability of digital computer and communication systems and networks associated with SSEP functions.

Near Term Strategy

To support the NRC's HFE reviews of advanced reactor license applications under 10 CFR Part 53, the NRC staff recently completed development of draft DRO-ISG-2023-03, "Development of Scalable Human Factors Engineering Review Plans" (ML22266A072, nonpublic). The draft guidance describes a method for scaling the scope and depth of HFE reviews for non-light-water reactor technologies such as micro-reactors, enabling the NRC staff to readily adjust the focus and level of NRC staff HFE review efforts considering factors such as risk insights and the unique characteristics of the design or facility operation (e.g., remote or autonomous operation). Although the guidance was developed to support reviews of applications submitted under the proposed 10 CFR Part 53, the NRC staff would intend to use the general methods described in this guidance to scale HFE reviews of micro-reactor applications submitted under 10 CFR Part 50 or 52.

For near-term license applications that include proposals for remote or autonomous operation, the NRC staff would use available guidance to assess compliance with the applicable 10 CFR Part 50 or 52 regulations and applicant requests for exemptions, as needed.

Remote Operations

Appendix A to 10 CFR Part 50 contains the general design criteria (GDC), which establish the minimum requirements for the principal design criteria (PDC) for water-cooled nuclear power plants “similar in design and location to plants for which construction permits have been issued by the Commission.” Appendix A also establishes that the GDC are considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units. GDC 19, “Control room,” requires (for water-cooled reactors), in part, that a control room 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, including loss-of-coolant accidents. Regulatory Guide (RG) 1.232, Revision 0, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” describes the NRC’s guidance on how the GDC in Appendix A, “General Design Criteria for Nuclear Power Plants,” of 10 CFR Part 50 may be adapted for non-light-water reactor designs. For nuclear reactors for which the GDC are requirements, the NRC staff would need to determine whether proposed PDCs for remote operation meet GDC 19. As noted in RG 1.232, the GDC in 10 CFR Part 50, Appendix A are not regulatory requirements for non-light-water reactor designs but provide guidance in establishing the PDC for non-light-water reactor designs.

As noted above, 10 CFR 50.34(f)(2)(iii) requires, in part, a control room design that reflects state-of-the-art human factor principles. For applications that include proposals for remote operation of a nuclear reactor, the NRC staff would intend to apply this requirement to any facility from which a nuclear reactor can be operated remotely. In addition, the NRC staff would need to determine whether providing a facility from which the reactor can be operated remotely would preclude the need for an on-site control capability beyond that which would otherwise be provided under the 10 CFR Part 50, Appendix A, GDC 19 requirement for equipment at appropriate locations outside the control room with (1) a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

The requirements in 10 CFR 50.54(m) specify minimum licensed operator staffing. The requirements are stated largely in terms of operators being required to be “present at the facility” or “on-site.” As written, the requirements do not address a model in which operation of a nuclear reactor would be performed from a location other than on site. For applications that include proposals for remote operation of a nuclear reactor, the applicant would need to request an exemption from requirements in 10 CFR 50.54(m), unless the deployment model included maintaining on-site licensed operator staffing that meets the current staffing requirements.[15] In 2005, the NRC staff published NUREG-1791, “Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR

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[15] Staffing operators both on site and at a remote operations facility would not be an economical long-term strategy but is an option that licensees might exercise if it is deemed a viable pathway toward developing a performance-based case for requesting an exemption from on-site staffing requirements at some point following initial plant start-up.

50.54(m).”[16] The NRC staff developed the guidance to support reviews of staffing for advanced reactors in which the concept of operations differed from that of the large light-water reactors upon which the current staffing regulations are based. The NRC staff would intend to evaluate staffing exemptions requests associated with applications proposing remote operations using NUREG-1791 as the guidance addresses not only potential reductions in staffing but also the possibility of operation from remote facilities and portable devices.

The NRC’s requirements in 10 CFR Part 26 are applicable to, among others, licensed operators at light-water reactor nuclear power plants currently licensed under 10 CFR Part 50 or 52. However, 10 CFR 26.4(a), limits applicability to “persons who are granted unescorted access to nuclear power reactor protected areas” and who, in the case of operators, perform the duties of “operating or onsite directing of the operation of systems and components that a risk-informed evaluation process has shown to be significant to public health and safety.” This leads to potential gaps in the coverage of these regulatory requirements. For example, a remotely located supervisory operator could potentially direct the operations of local personnel from a remote location; under such an arrangement, the supervisory operator would be neither “operating” nor “onsite directing,” and thus could bypass fitness-for-duty requirements. In developing these requirements, the NRC did not consider the possibility of remote operations where the reactor may be controlled by individuals who are not at the reactor site and therefore may not require unescorted access to a nuclear power reactor’s protected areas. Therefore, the NRC staff would need to address the limitations of the current 10 CFR Part 26 applicability language for license applications proposing remote operation of a nuclear reactor facility. The NRC staff would intend to address the limitations in a manner consistent with the proposed amendments to 10 CFR Part 26 in the 10 CFR Part 53 draft proposed rule and the associated draft guidance in DG-5078, “Fatigue Management for Nuclear Power Plant Personnel at Commercial Nuclear Plants Licensed Under 10 CFR Part 53” (ML22264A109, nonpublic).

Autonomous Operations

The requirements in 10 CFR 50.54(i), (j), (k), and (m) address onsite operator staffing and having a licensed operator at the controls during facility operation, require that only licensed operators may manipulate controls, and require that apparatus and mechanisms, other than controls, that may affect reactivity or power level be manipulated only with the knowledge and consent of an operator or senior operator. License applications submitted under 10 CFR Part 50 or 52 for autonomous operations of nuclear reactors that, for example, would not include licensed operator staffing in a control room or would permit a non-licensed grid operator to change plant output (i.e., load following) would need to include requests for exemptions from these requirements, where relevant. Beyond this, additional regulatory implications will also need to be addressed, such as requirements for a control room and a remote shutdown capability. The NRC does not currently have guidance specific to the evaluation of such requests. Additionally, as noted previously, the AEA places certain mandates upon the NRC to license those individuals who will operate the controls of utilization facilities. However, as

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[16] More recently the NRC staff developed draft interim staff guidance to augment NUREG-1791 to support NRC staff review of advanced reactor staffing plans under 10 CFR Part 53 (i.e., DRO-ISG-2023-02, “ISG Augmenting NUREG-1791, ‘Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m),’ for Licensing Plants under Part 53” (ML22266A068, nonpublic)).

described in the following section, "Next Steps," the NRC staff has initiated research to more fully understand the extent to which reactor vendors intend to include autonomous operations in proposed concepts of operations for future applications and where there may be matters of importance or urgency for development of additional licensing guidance.

Cybersecurity

Control commands needed for remote operation require bidirectional data communication. Bidirectional data flow would create digital architectures that vary from those used by plants in the current operating fleet, which only allow unidirectional data flow from high levels of security to lower levels. Fully understanding the architecture will be key in providing insights about the attack surface and potential attack vectors. Without a clear understanding of the architecture and how cybersecurity controls and safety protocols would be implemented to ensure the security of a data communication pathway to the micro-reactor, it is difficult to determine risk, establish confidence, and credit important human actions needed to mitigate an event from compromise. Additionally, it is likely that protection against disruption or malicious control for micro-reactors will rely heavily on properly implemented DCSA including technical security controls – such as cryptography – rather than extensive use of physical security as is currently used in operating nuclear power plants. Furthermore, the level of autonomy and the capabilities of the autonomous technologies that would be used to replace humans for remote operations would be equally important to understand.

The regulations at 10 CFR 73.55 require, in part, that cybersecurity requirements must be implemented "before fuel is allowed onsite (protected area)." The portable design of a factory-fabricated micro-reactor introduces a challenge to current cybersecurity regulations and guidance written for 10 CFR Part 50 and 52 plants. The NRC staff would need to understand the security design elements that would be used to provide assurance that systems and networks are protected from cyberattacks during fabrication, transit, testing, and startup phases of the micro-reactor's product development lifecycle.

Next Steps

The NRC staff plans to further develop its understanding of the industry deployment models for factory-fabricated micro-reactors with respect to industry plans for remote and autonomous operations, identify any gaps in the existing HFE review needed to address the deployment models, and develop the technical bases for any new guidance that may be needed.

Regarding cybersecurity, the NRC staff has proposed as part of the 10 CFR Part 53 rulemaking, a new risk-informed, performance-based, technology-neutral cybersecurity requirements that apply a graded approach to advanced reactors for protection of digital computers, communication systems, and networks within the scope of the new requirement.

5. Transportation of Fueled Factory-Fabricated Modules

Deployment Model Considerations for Transportation

Factory-fabricated micro-reactor developers (and potentially developers of floating nuclear power plants that use reactors with higher power levels) envision transporting fueled factory-fabricated modules from a factory to the deployment site for operation and later removing fueled modules from the deployment site at the end of operation. As previously discussed in this paper, the use of features to preclude criticality would be needed in order to consider that the factory-fabricated module would not be in operation when loaded with fuel.

Shipment of a factory-fabricated module containing fuel would be subject to the transportation requirements in 10 CFR Part 71. Shipment would also be subject to the requirements applicable to the licenses held by the fabricator for possession of the modules, special nuclear material, and any other radioactive material contained in the modules. As stated in this paper, the NRC staff assumes that the fabricator will obtain a manufacturing license under 10 CFR Part 52 for the factory-fabricated modules and that the manufacturing license would authorize possession of the modules at the factory. A license issued pursuant to 10 CFR Part 70 would also be required for possession of the special nuclear material in the fuel loaded in the module and a byproduct material license issued pursuant to 10 CFR Part 30 might also be required if the module had been operated for testing and contained fission products.

Under a 10 CFR Part 52 manufacturing license, the licensee would be subject to the regulations in 10 CFR 52.153, 52.157, 52.167, in addition to 10 CFR Part 71 requirements. These requirements provide that the reactor is transported to a deployment site for which the licensee accepting the shipment has the proper licenses (e.g., a construction permit under 10 CFR Part 50 or a combined license under 10 CFR Part 52) and provide reasonable assurance that the reactor can be installed and operated at the deployment site. In addition, the regulations in 10 CFR 52.157(f)(26)(iv) require that the application for a manufacturing license include a description of the proposed procedures for shipment of the reactor. These procedures, which may be different from the operating procedures submitted for package approval under 10 CFR Part 71, govern preparation of the reactor for shipping, performing the shipment, and verification of the reactor's condition upon receipt at the site to minimize the potential that the reactor arrives at its destination and is unable to operate within the parameters of the license at the deployment site.

The requirements in 10 CFR 70.20a and 10 CFR 70.20b contain provisions for general licensees to possess special nuclear material and irradiated fuel during transport and storage incident to transport. In addition, 10 CFR Parts 70, 50, and 52 contain requirements for physical protection of fissile material in transport. The general license requirements in 10 CFR 70.20a and 70.20b also contain physical protection requirements to follow the appropriate sections in 10 CFR Part 73, for material transport and storage incident to transport. However, the regulations for physical protection of special nuclear material in transport do not explicitly include requirements for special nuclear material that is loaded in a utilization facility during transport. As discussed in the main body of this paper, the NRC staff will consider whether additional Commission engagement is needed related to physical security requirements for factory-fabricated micro-reactors, including for transportation.

<u>Packaging and Transportation</u>

The regulations in 10 CFR Part 71 contain requirements for Type B, Type B fissile (Type BF), and Type A fissile (Type AF) material packages and their transportation. The regulations that apply to the package will depend on the contents. Transportation packages for factory-fabricated modules approved under 10 CFR Part 71, may consist of the module itself or the module plus an additional overpack or other materials, as needed, to meet the packaging requirements. Packages for transporting a module from the factory to the deployment site could be either a Type AF or Type BF package, as defined in 10 CFR Part 71. Selection of the appropriate package would depend on the enrichment history of the uranium in the loaded fuel and whether the module was operated at the factory for testing, which would determine whether there is greater than a Type A quantity of radioactive material in the package.

In addition to the requirements in 10 CFR Part 71, the packaging and transport of licensed material are also subject to other requirements in 10 CFR and to the regulations of other agencies. A factory-fabricated module, whether shipped prior to operation or after operation, would be subject to the requirements in 10 CFR Parts 20 and 21, regardless of the contents or licensing pathways chosen by the Commission in response to this paper. In addition, the regulations in 10 CFR Parts 30, 50, 52, and 70 could apply, depending on the contents and parts in 10 CFR under which the reactor is licensed.

The NRC also co-regulates transportation with the Department of Transportation (DOT). The DOT regulates shipment of all classes of hazardous material, including radioactive materials. The shipper, and carrier are subject to the DOT regulations and depending on the components of the package, may be subject to more than one hazard class under DOT regulations. DOT regulations authorize shipment of fissile material and radioactive material in some NRC-approved packages under 49 CFR 173.415 and 173.416, respectively. However, if the NRC package approval options of either 71,41(c) or an exemption are utilized, the shipper must obtain a special permit issued by the DOT, as these package approvals are not automatically authorized in DOT regulations.

Front End Transport

As discussed below, 10 CFR Part 71 is adequate for approving a factory-fabricated module for shipment. A package used to transport a fueled factory-fabricated module must be evaluated against the package performance requirements in 10 CFR Part 71 for the type of package that the module fabricator or licensee proposes to use to ship the module. A fueled factory-fabricated module that has not been operated for testing at a factory and contains commercial grade uranium enriched to less than 20 weight percent in the uranium-235 isotope would be classified as a Type AF package. However, if the package contains low-enriched fuel that includes reprocessed or downblended uranium, then the package would likely be classified as a Type BF package, depending on the quantity of impurities in the fuel and the A_1 or A_2 value[17] for the mixture. Transport of a module that has been operated for testing at a factory could call for a Type AF or Type BF package depending on the quantity of radionuclides generated during testing and the radionuclides initially present, e.g., if the fuel was reprocessed or downblended. Based on the expected power level and duration of testing and the time between the completion of testing and shipment, the applicant should determine the quantity of radionuclides that would be present at the intended time of shipment, using Appendix A to 10 CFR Part 71, and determine whether the contents constitute a Type A or Type B quantity of material. The regulations in 10 CFR 71.31 require the applicant for a package approval to provide a description of the contents and determine, based on the proposed contents, whether the package would be a Type AF or Type BF package.

Both Type AF and Type BF packages would be subject to the tests and conditions for normal conditions of transport and hypothetical accident conditions and required to maintain criticality safety in accordance with 10 CFR 71.55 and 71.59 and show that the dose rates for normal conditions of transport remain below the criteria in 10 CFR 71.47. However, there would be no

17 Appendix A, "Determination of A_1 and A_2," to 10 CFR Part 71 contains the A_1 and A_2 values for individual radioisotopes and the method for calculating the aggregate A_1 and A_2 value for a mixture of radionuclides.

containment or dose rate criteria after hypothetical accident conditions for the Type AF package. For Type BF packages, the package designer would be required to show that the package meets the hypothetical accident condition dose rate and containment criteria in 10 CFR 71.51 in addition to maintaining criticality safety and meeting the dose rate criteria in 10 CFR 71.47.

Shipment of Factory-Fabricated Modules Between NRC-Licensed Sites or Back End Transport

The point at which a factory-fabricated module would transition from a Type AF package to a Type BF package depends heavily on the module design and power level and duration of operation. The NRC staff estimates that after half a day of full power operation or equivalent, the module would likely contain greater than a Type A quantity of radionuclides. Presuming the factory-fabricated module is operated at full power for more than a single day, the applicable regulatory requirements would very likely be those for a Type BF package.

Near Term Strategy

The NRC and the DOT co-regulate transportation of radioactive material, with the NRC regulating transportation for its licensees and issuing certificates of compliance for both fissile (Type AF and Type BF) and nonfissile (Type B) packages.[18] The DOT regulations in 49 CFR 173.416 and 173.417 authorize, among other things, shipment of any Type B or fissile material package approved by the NRC. The NRC staff intends to use the existing regulatory framework (primarily 10 CFR Part 71) to review transportation of commercial fueled factory-fabricated modules in the near term.

The regulations in 10 CFR Part 71 contain performance-based requirements for packaging and transportation of radioactive material. The NRC regulations have been harmonized with the International Atomic Energy Agency standards in the 2009 Edition of TS-R-1, ‘‘Regulations for the Safe Transport of Radioactive Material,” to facilitate international transport.[19] However, under the current regulatory framework, the NRC may approve alternate standards for packages that may not meet all of the packaging requirements in 10 CFR Part 71. Specifically, 10 CFR 71.41(c) allows for environmental and test conditions different from those in 10 CFR 71.71 and 71.73 if the shipper’s controls provide safety of the shipment equivalent to that provided by meeting the regulations. Further, 10 CFR 71.41(d) provides for special package authorization if the application demonstrates that compliance with the regulations is impracticable and the safety standards established by the regulations have been met through alternative means. Finally, an applicant may request an exemption as specified in 10 CFR 71.12. Each of these alternatives has limitations, as described in more detail below.

The requirements in 10 CFR 71.41(c) provide for alternate environmental and test conditions for a package that, when subjected to the environmental conditions required by the regulations, in conjunction with one or more of the tests for normal conditions of transport or hypothetical accident conditions, cannot meet the post-test criteria. Use of the alternate test criteria in 10 CFR 71.41(c) has several limitations. An applicant for package approval cannot eliminate the test but rather can reduce the severity of the test (e.g., the applicant can use a 20-foot drop

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18 Type B packages contain a quantity of radioactive material greater than a Type A quantity. The NRC defines a Type A quantity of material in 10 CFR 71.4.

19 The NRC is in the process of harmonizing 10 CFR Part 71 with the International Atomic Energy Agency safety standards in Specific Safety Requirements No. 6 (SSR-6) (2018 Edition). See proposed rule dated September 12, 2022, (87 FR 55708) “Harmonization of Transportation Safety Requirements with IAEA Standards” (RIN 3150-AJ85; NRC-2016-0179).

instead of 30-foot drop but cannot substitute a different test) so that the package can meet the post-test criteria. In addition to the alternative environmental conditions or test criteria, the applicant must submit additional controls that the shipper can exercise to provide an equivalent level of safety for the shipment. Because the regulations in 10 CFR 71.41(c) do not offer alternate post-test criteria, the applicant would still need to meet the regulatory limits for dose rate, containment, and criticality safety. Differing post-test criteria can only be approved through exemption.

After its experience with issuing the exemption for the Trojan reactor vessel (see ML20155E053 and SECY-98-0231, "Authorization of the Trojan Reactor Vessel Package for One-Time Shipment for Disposal," dated October 2, 1998), the NRC noticed a need for a provision for a special package authorization for one-time shipment of large components that do not meet the criteria for shipment as low specific activity packages or surface contaminated objects.[20] In the 2002 proposed rulemaking ((67 FR 21390), Issue No. 12, "Special Package Authorizations"), the NRC added the special package authorization option in 10 CFR 71.41(d) for limited circumstances involving large packages for which it is not practical to fabricate an authorized packaging. In particular, the NRC limited this alternate approval method to, among other things, one-time shipments of large components "for which compliance with the other provisions of these regulations [i.e., 10 CFR Part 71] is impracticable." The final rule (69 FR 3742, dated January 26, 2004) states that the "special package authorizations that will apply only in limited circumstances and only to one-time shipments of large components." In order to meet the intent of the rule, a different application, and an NRC review resulting in a different approval would be needed for each shipment for each factory-fabricated module, likely making this option cost-prohibitive.

If neither 10 CFR 71.41(c) nor 10 CFR 71.41(d) can be used, licensees can request an exemption from the regulations pursuant to 10 CFR 71.12. Through exemption, licensees can provide alternate environmental conditions and tests *and* alternate post-test criteria. The exemption request must contain sufficient technical information for the NRC staff to determine that the request is authorized by law and will not endanger life or property or the common defense and security. The exemption request should be accompanied by an environmental report because the categorical exclusion in 10 CFR 51.22(c)(13) for "package designs for packages to be used for the transportation of licensed materials" would not apply. In addition, each licensee making a shipment needs to request a separate exemption, because an exemption cannot be made generically applicable to multiple licensees. Further, the DOT's regulations do not specifically authorize NRC-issued exemptions as a package approval; therefore, each licensee would need a DOT-issued special permit for its shipment.

<u>Next Steps</u>

The NRC staff will continue to engage with factory-fabricated micro-reactor developers and potential package applicants to discuss their plans for package approval. As appropriate, depending on the package approval plans for factory-fabricated modules, the NRC staff will evaluate the need for future Commission papers, rulemaking, and guidance, including for security. The NRC staff plans to continue to engage with individual developers through pre-application activities and other stakeholders via the periodic advanced reactor stakeholder meetings.

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[20] For the definitions of low specific activity and surface-contaminated object, see 49 CFR 173.403, "Definitions." For the exemption from most of the requirements in 10 CFR Part 71 for low-specific-activity packages and surface-contaminated objects, see 10 CFR 71.14(b)(3).

6. Storage of Fuel After Irradiation in a Power Reactor

Deployment Model Considerations

Fuel for factory-fabricated micro-reactors (and advanced reactors in general) may have much higher temperature limits than fuel historically used in light-water reactors, and there could be advanced reactor designs for which the spent fuel would not need to be water cooled immediately after withdrawal from the reactor, like existing zirconium-clad, light-water reactor fuel. For advanced reactors, depending on the time duration between withdrawal of the fuel from the reactor (or the final reactor shutdown) and placement into a dry storage facility, different regulations may apply to the storage of the reactor fuel or the fueled factory-fabricated module.[21] Spent fuel must be cooled for at least 1 year prior to being stored in an independent spent fuel storage installation (ISFSI) regulated under 10 CFR Part 72. Absent exemptions from the scope defined in 10 CFR 72.2(a) and other related requirements in 10 CFR Part 72, advanced reactor fuel that has not been cooled for at least 1 year would be required to be licensed under 10 CFR Part 50 or 52 for near-term storage as part of the reactor facility, similar to a spent fuel pool for a light-water reactor.

As provided in 10 CFR 72.2(a)(1), ISFSIs licensed under 10 CFR Part 72 are limited to the receipt, transfer, packaging, and possession of "[p]ower reactor spent fuel to be stored in a complex that is designed and constructed specifically for storage of power reactor spent fuel aged for at least one year, other radioactive materials associated with spent fuel storage, and power reactor-related GTCC [greater than class C] waste in a solid form in an independent spent fuel storage installation (ISFSI)." Similarly, the definition of spent fuel in 10 CFR 72.3 includes criteria that the fuel has been withdrawn from a nuclear power reactor following irradiation and has undergone at least one year's decay since being used as a source of energy in a power reactor. Therefore, the NRC could not license an ISFSI under 10 CFR Part 72 in which the spent fuel was decayed for less than a year.

The regulatory history of 10 CFR Part 72 provides insight into the basis for this one-year decay period. In the proposed rulemaking for 10 CFR Part 72 dated October 6, 1978 (43 FR 46309), the NRC stated:

> The storage of spent fuels under water is only necessary for those fuels which have not undergone sufficient aging since their discharge from a reactor to make cooling by some other means feasible.
>
> The proposed rule is applicable only to "aged" fuel, with more than one year's decay since reactor shutdown. Aged spent fuel, having lost the short-lived radionuclides by decay, need not have a high degree of protection from weather extremes, tornadoes, or tornado generated missiles.

At the time, water cooling of commercial light-water reactor spent fuel was deemed necessary prior to placement in an ISFSI to ensure that the fuel would not overheat. Storage of spent fuel

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[21] The NRC's definition of spent fuel in 10 CFR 71.4 is the same as that currently found in 10 CFR 72.3 for ISFSIs. The transportation and packaging requirements in 10 CFR Part 71 do not use the term spent fuel, such that the only location this term exists in the transportation regulations is in the definitions of spent fuel in 10 CFR 71.4. Because there are no package approval standards or transportation requirements in 10 CFR Part 71 stating that the fuel must be cooled for a year prior to transport, the NRC can approve package designs for shipment of fuel cooled for less than 1 year.

in an ISFSI was not limited to water pool installations. Further, in the final rulemaking dated November 12, 1980 (45 FR 74694) the NRC stated:

> The long-lived radionuclides present in spent fuel are proportional to burnup; but within the limits of expected burnups, this is not a significant factor for spent fuel aged more than one year.
>
> The one-year decay stipulation has been retained as this is a basis for the requirements of Part 72, i.e., the presumption is made that no short-lived radionuclides are present and the levels of volatile radioactive materials are very substantially reduced.
>
> Inasmuch as the definition of spent fuel eligible for storage in an ISFSI [Section 72.3] specifies that the fuel must have undergone at least a year's decay since its irradiation in a power reactor, any facility for temporary storage of fuel irradiated in a power reactor which has not undergone a year's decay would be licensed under Part 50 rather than Part 72.

As stated in 10 CFR 50.1, the purpose of 10 CFR Part 50 is to "provide for the licensing of production and utilization facilities," so the reference to "licensed under 10 CFR Part 50" in the 1980 final rulemaking should be understood in that context. Since that 1980 rule, the NRC has established 10 CFR Part 52 as an available path to license nuclear power reactors.

<u>Near Term Strategy</u>

Dry cask storage designs for use at a generally licensed ISFSI or an ISFSI with a specific license for storage of spent fuel that has been cooled for at least a year would be approved under 10 CFR Part 72. However, any near-term spent fuel storage of fuel that has been cooled for less than 1 year would be licensed under either 10 CFR Part 50 or 52, unless an exemption from the minimum cool time requirements of 1 year were granted to allow issuance of an ISFSI license under 10 CFR Part 72.

<u>Next Steps</u>

The NRC staff intends to engage with stakeholders as they further develop their strategies for handling and storage of irradiated and spent fuel generated in factory-fabricated micro-reactors.

7. Decommissioning Process and Decommission Funding Assurance

<u>Deployment Model Considerations</u>

Factory-fabricated micro-reactor deployment models might involve transporting a factory-fabricated module away from the deployment site to a facility at a different location for decommissioning at the end of its life or for refurbishment and refueling before re-deployment. A decommissioning facility might be used to dismantle the factory-fabricated module to recover reusable parts and prepare the waste and spent fuel for transfer, or it might also include an independent spent fuel storage facility. A refurbishment and refueling facility might be used to defuel the factory-fabricated module, perform maintenance, refuel the module, and possibly operate the module for testing before re-deployment.

In the following discussion, the NRC staff assumes that the decommissioning of factory-fabricated micro-reactors would involve independent decommissioning of the factory-fabricated module and the deployment site.[22] This would require physically transferring the module from the deployment site to a decommissioning facility with the appropriate license(s) to receive the module and its contents. The deployment site licensee would be responsible for decommissioning the remaining onsite structures and meeting the relevant criteria for license termination and release of the site. The decommissioning facility licensee would be responsible for decommissioning the module. Under this scenario, the deployment site licensee might be the same entity as the decommissioning facility licensee or there might be different licensees. In either case, a factory-fabricated module would need to be covered by separate licenses appropriate for the activities to be conducted at the deployment site and at the decommissioning facility. If the module were transferred to a refurbishment and refueling facility instead of a decommissioning facility, then the refurbishment and refueling facility licensee would be responsible for the module and its redeployment and the deployment site licensee would be responsible for the deployment site decommissioning.

A decommissioning facility or a refurbishment and refueling facility might require several NRC licenses depending on the activities to be conducted at the facility. A license issued pursuant to 10 CFR Part 30 would be required to receive, possess, and transfer the byproduct material created by operation of the reactor at the deployment site. A license issued pursuant to 10 CFR Part 70 would be needed for receipt, possession, and transfer of the special nuclear material in the form of the irradiated or spent fuel removed from the factory-fabricated module and any fresh fuel needed for refueling. A license issued pursuant to 10 CFR Part 72 would be required if the facility were also to serve as the storage location for spent fuel, power reactor-related Greater than Class C waste, and other radioactive materials associated with spent fuel storage. The facility would also need an operating license issued pursuant to 10 CFR Part 50 or a combined license issued pursuant to 10 CFR Part 52 to receive, possess, use the reactor in order to operate it for testing after refurbishment and refueling, and for reactor and facility decommissioning. Also, a utilization facility license would be required to receive and possess the factory-fabricated module at a decommissioning facility or a refurbishment and refueling facility even if it is not operated for testing.

At the end of the life of a module or its fuel cycle at the deployment site, the deployment site licensee would remove it from service and install features to preclude criticality that would render the module not in operation as proposed by the NRC staff in this paper. For transportation to either a decommissioning facility or a refurbishment and refueling facility, the deployment site licensee would be required to prepare the module for transport in accordance with the requirements in 10 CFR Part 71, the package approval, and other applicable regulations. Among other requirements, the transportation package would have to meet the criticality safety requirements in 10 CFR 71.55 and 71.59, which could potentially be satisfied through installation of the features to preclude criticality. Once the factory-fabricated module is removed from the deployment site and transferred to the decommissioning facility or refurbishment and refueling facility, the module would no longer be the responsibility of the deployment site licensee for the purpose of decommissioning the deployment site. The deployment site licensee would be responsible for completing decommissioning of the deployment site consistent with applicable regulations of 10 CFR Part 20, Subpart E, as guided by NUREG-1757, "Consolidated Decommissioning Guidance." The deployment site license for that reactor could be terminated upon meeting the decommissioning requirements for the

[22] Factory-fabricated micro-reactors could also be decommissioned at the deployment site following the traditional approach used for large light-water reactors.

remainder of the deployment site structures and systems unique to that reactor. If shared structures, systems, and components were to be used in connection with a replacement factory-fueled module, those structures, systems, and components important to safety or otherwise would need to be transferred to the new license for the replacement module.

The deployment site licensee would need to establish decommissioning funding assurance that considered the cost of removing the module from the site and decommissioning it elsewhere, in addition to the cost of decommissioning onsite structures in order to permanently terminate the license and meet the requirements in 10 CFR 20, subpart E. As required by 10 CFR 50.33(k)(1), reactor applicants for an operating license or a combined license must provide a report "as described in [10 CFR] 50.75, indicating how reasonable assurance will be provided that funds will be available to decommission the facility." For power reactor licensees, "reasonable assurance consists of a series of steps as provided in paragraphs (b), (c), (e), and (f) of this section." The regulations at 10 CFR 50.75(c) establish the minimum amounts of funding required to demonstrate reasonable assurance of funds for decommissioning by reactor type and thermal power level for pressurized water reactors and boiling water reactors. However, most current designs for factory-fabricated micro-reactors use non-light-water reactor technology that may involve significantly different decommissioning considerations and strategies compared to pressurized or boiling water reactors. Further, 10 CFR 50.75(c) requires that a power level of at least 1200 megawatts thermal be used in calculating the minimum amounts, which is roughly 500 to 100 times the power level of factory-fabricated micro-reactor designs. The formulas result in required funding assurance in excess of $75 million (January 1986 dollars) for each reactor, which is not compatible with deployment models. Reliance on use of the minimum formula amount for decommissioning during operations as reflected in 10 CFR 50.75(c) may need to be revisited as discussed in "Near Term Strategy" below.

In order to ensure adequate funding, a decommissioning cost estimate would need to consider and account for all activities and waste disposal costs associated with decommissioning the deployment site and decommissioning the factory-fabricated module at a decommissioning facility. It is possible that the decommissioning cost estimate for a module could be a predetermined estimate provided by a decommissioning facility licensee or a refurbishment and refueling facility licensee authorized to acquire such a module (which could be the original fabricator of the module). It is also possible that the deployment site licensee would be the decommissioning facility licensee and a cost estimate would account for the various component costs to dismantle and dispose of the module and store or transfer the spent fuel. In either case, a preliminary decommissioning plan submitted with the deployment site license application would need to describe how the decommissioning funds would be accounted for between decommissioning the factory-fabricated module and the deployment site decommissioning activities. Later, decommissioning plans would be required to be submitted in the form of a Post-Shutdown Decommissioning Activities Report, a site-specific Decommissioning Cost Estimate, a License Termination Plan, and a final status survey in accordance with 10 CFR 50.82 or 52.110.

Near Term Strategy

The NRC staff intends to use the existing regulatory framework to review applications for licenses related to decommissioning and refurbishment and refueling activities for factory-fabricated micro-reactors. If the Commission approves the use of features to preclude criticality, then the NRC staff will consider this in its review of applications for decommissioning and refurbishment and refueling facilities. The NRC staff intends to also consider the approaches for licensing multi-module sites in SECY-11-0079 in relation to decommissioning and refurbishment and refueling facilities that may require 10 CFR Part 50 or 52 licenses.

With respect to decommissioning funding assurance, the NRC staff intends to consider specific exemptions under 10 CFR 50.12 from the requirements in 10 CFR 50.75(c) which were not established for non-light-water reactors. The NRC staff may consider site-specific decommissioning cost estimates in the review of applications for operating licenses and combined licenses for factory-fabricated micro-reactors.[23] The NRC staff notes that the use of site-specific decommissioning cost estimates is included in the proposed draft regulations in 10 CFR Part 53.

Next Steps

The NRC staff will continue to engage stakeholders on considerations related to decommissioning and refurbishment and refueling of factory-fabricated micro-reactors to better understand the range of options under consideration. If the NRC staff identifies issues that involve policy decisions or potential rulemaking, the NRC staff will seek Commission direction through an additional options paper.

8. Siting in Densely-Populated Areas

The NRC has a longstanding policy of siting nuclear power reactors away from densely populated centers and preferring areas of low population density. As discussed in SECY-20-0045, “Population-Related Siting Considerations for Advanced Reactors” (ML19143A194), dated May 8, 2020, the attributes of advanced reactors, including micro-reactors, are expected to provide a reduced likelihood of accidents and to result in a smaller and slower release of radioactive material in the unlikely event of an accident. These attributes of advanced reactors, if demonstrated, may support siting them closer to population centers than large light-water reactors typically have been. As such, in SRM-SECY-20-0045, “Staff Requirements – SECY-20-0045 – Population-Related Siting Considerations for Advanced Reactors” (ML22194A885), dated July 13, 2022, the Commission approved the NRC staff’s proposal to revise the population-related siting guidance in Regulatory Guide (RG) 4.7, “General Site Suitability Criteria for Nuclear Power Stations,” Revision 3 (ML12188A053), issued March 2014, to provide technology-inclusive, risk-informed, and performance-based criteria to assess certain population-related issues in siting advanced reactors.

The NRC staff is updating RG 4.7 to include the alternative population-related criteria approved by the Commission in SRM-SECY-20-0045. The guidance will state that, instead of locating a reactor in an area where the population density does not exceed 500 persons per square mile (ppsm) out to 20 miles from the reactor, an applicant can demonstrate compliance with 10 CFR 100.21(h) by siting a nuclear reactor in a location where the population density does not exceed 500 ppsm out to a distance equal to twice the distance at which a hypothetical individual could receive a calculated TEDE of 1 rem over a period of 1 month from the release of radionuclides following postulated accidents.

While the NRC is revising its population-related siting guidance to include alternate means of compliance with 10 CFR 100.21(h), the regulations in 10 CFR 100.21 remain unchanged. This includes the provision in 10 CFR 100.21(b), which requires that “[t]he population center

[23] The NRC staff described application of this approach to small modular nuclear reactors in SECY-11-0181, “Decommissioning Funding Assurance for Small Modular Nuclear Reactors” (ML112620358), dated December 22, 2011.

distance, as defined in § 100.3, must be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone." In 10 CFR 100.3, "population center distance" is defined as "the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents."

Further, as discussed in RG 4.7, a densely populated center is considered to contain more than about 25,000 residents and the boundary of the population center is determined based on consideration of population distribution rather than political boundaries. As such, current Commission policy and regulations would preclude siting a commercial power reactor, no matter the size or type of reactor, within a densely populated center. The allowable distance from the reactor to a densely populated center of approximately 25,000 residents would be no closer than 1.33 times the radius of the low population zone (LPZ). In accordance with 10 CFR 50.34(a)(1)(ii)(D)(2), 10 CFR 52.17(a)(1)(ix)(B), and 10 CFR 52.79(a)(1)(vi)(B), the LPZ is required to be of such a size that an individual located on its outer boundary during the course of the postulated accident would not receive a radiation dose in excess of 25 rem TEDE. The size of the LPZ depends on atmospheric dispersion characteristics and population characteristics of the site, as well as aspects of plant design. The NRC staff notes that 10 CFR 100.21 is not applicable to research and test reactors. However, testing reactors are subject to 10 CFR 100.11(a)(3) which requires a population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. There are research reactors currently sited within population centers greater than 25,000 residents, and the NRC staff anticipates license applications for additional research reactors to be sited in densely populated areas.

<u>Deployment Model Considerations</u>

Some micro-reactor license applicants may seek to site reactors at locations that are inconsistent with the current Commission policy and the regulations in 10 CFR 100.21(b). Such deployment scenarios are being considered for several reasons including replacing existing coal plants or providing process heat for heating or industrial applications, or to provide power to remote communities or smaller grids with relatively small but concentrated populations that would be close to a reactor site.

<u>Near Term Strategy</u>

The NRC staff will continue its effort to revise RG 4.7 and will review license applications in accordance with current Commission policy that allows alternative population-related siting criteria but precludes siting a commercial power reactor, no matter the size or type of reactor, within a populated center of 25,000 residents or more.

<u>Next Steps</u>

The NRC staff will continue to engage with reactor developers and prospective license applicants as it revises the guidance in RG 4.7. The NRC staff will inform the Commission if it becomes aware of any license applicants who intend to seek exemption from 10 CFR 100.21(b) and will raise associated policy issues to the Commission accordingly.

9. Commercial Maritime Applications

Deployment Model Considerations

The NRC staff is aware of growing interest in commercial maritime applications of micro-reactors and other reactor technologies for stationary power production, marine vessel propulsion, production of decarbonized fuels, and other uses. Stationary reactors might be located in ports or other coastal locations or further out from the shore in domestic waters. Reactors used for commercial maritime vessel propulsion might be operated solely within U.S waters or internationally, especially in the shipping industry. Reactors used for decarbonized fuel production or other chemical processing or industrial applications would typically be stationary power reactors in that they would be moored or anchored at a fixed site while in operation, but they might be moved between several locations during their lifetime.

The various maritime applications envisioned by developers give rise to numerous potential legal, regulatory, and policy issues. For example, reactors located in coastal waters may have different environmental considerations than land-based reactors. Such environmental considerations could potentially involve the Coastal Zone Management Act, Magnuson-Stevens Fishery Conservation and Management Act, and Marine Mammal Protection Act. In addition, siting in the marine environment may require different approaches to analyses of external hazards, dose modeling, and others. Deployment models might also include scenarios where larger advanced reactors or light-water reactors are fabricated and potentially fueled in a factory before being transported to the maritime deployment site, which could potentially give rise to additional considerations beyond those examined for micro-reactors in this paper. Also, the current legal and regulatory framework may present challenges (such as regulatory jurisdiction considerations, international licensing, and domestic licensing of operating reactors without fixed sites) for deployment models involving nuclear propulsion of commercial maritime vessels, including those where a U.S. flag vessel would operate in international or foreign waters, or a foreign flag vessel would operate in domestic waters.

Near Term Strategy

The NRC staff plans to assess the existing regulatory framework and its applicability to the licensing of stationary floating nuclear power plants, which might use factory-fabricated micro-reactor designs or larger advanced reactor or light-water reactor designs.

Next Steps

The NRC staff will monitor developments related to commercial maritime applications and assess the need for future Commission direction and coordination with other Federal agencies related to deployment of commercial maritime reactors. The NRC staff will also continue to communicate periodically with DOE staff on maritime reactor activities through DOE's Maritime Nuclear Application Group.

10. Commercial Space Applications

Deployment Model Considerations

The NRC staff is aware that developers are considering space applications of factory-fabricated micro-reactors. Government agencies such as the National Aeronautics and Space

Administration (NASA) and US DOE are encouraging development of the technology, primarily for government projects. The NRC staff is not aware of any fully commercial ventures that plan to use micro-reactors for space applications, whether that be for power generation for space vehicles, extraterrestrial installations, or propulsion systems.

In the case of a fully commercial space application of a factory-fabricated micro-reactor, the NRC's established regulatory jurisdiction and licensing authority, subject to the authorities of other agencies such as the DOT, would cover the related terrestrial activities prior to launch activities.[24] Upon initiation of launch activities, the Federal Aviation Administration's (FAA) Office of Commercial Space Transportation (a part of the DOT) would have authority.[25] Terrestrial activities under NRC's regulatory jurisdiction would include licensing and oversight of the manufacture, construction, potential operation for testing or technology demonstration, transportation, and storage of utilization facilities intended to be deployed at extraterrestrial locations and in space. The NRC's authority under the AEA and 10 CFR Parts 50 and 52 for domestic licensing and regulation of utilization facilities does not extend to the operation of reactors outside the borders of the United States.

<u>Near Term Strategy</u>

If developers engage the NRC staff on terrestrial activities related to commercial space applications of micro-reactors, the NRC staff intends to apply the established regulatory framework, as informed by this paper and any resultant Commission direction for factory-fabricated micro-reactors.

<u>Next Steps</u>

The NRC staff will monitor developments related to commercial space applications, including those involving Government and commercial partnerships, and assess the need for future Commission direction.

The NRC staff will continue to engage with other Government Agencies on matters of regulatory jurisdiction, licensing, and safety of launches and applications of space nuclear systems as appropriate through ongoing interagency activities and the INSRB.

11. Commercial Mobile Micro-Reactors

<u>Deployment Model Considerations</u>

The NRC staff uses the term "mobile micro-reactor" to refer to a micro-reactor that is intended to be operated at more than one location on an as-needed, where-needed basis without a preapproved specific site license. The U.S. Department of Defense (DoD) Strategic Capabilities Office Project Pele aims to develop and test a demonstration unit for a mobile micro-reactor for

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24 Letter from Samuel J. Collins to Robert D'Ausilio (May 6,1998) (ML20013J130, nonpublic).

25 The Interagency Nuclear Safety Review Board (INSRB) issued a trial use "playbook" titled, "Non-binding Guidance for INSRB and Its Counterparts," dated January 20, 2023, that includes, "Appendix E: Defining a US Government Launch versus a Commercial Launch and DOT Authority." The appendix discusses the responsibilities and authorities of Federal Aviation Administration's Office of Commercial Space Transportation (a part of the DOT) as they relate to launches of space nuclear systems and states that, "the definition of what is a commercial launch, from the DOT/FAA licensing perspective, is whether or not the launch or reentry event is commercially conducted."

military applications. NRC licensing is not required for this reactor because it is authorized under AEA Section 91b., but DoD is seeking NRC approval of a transportation package under 10 CFR Part 71. The DoD engaged with multiple vendors who developed proposed designs for this program. Some micro-reactor developers have indicated that they may eventually deploy mobile commercial factory-fabricated micro-reactors in this way, such as for disaster relief applications. However, this deployment model does not appear to be a near-term focus for developers.

The NRC has historically issued licenses for land-based reactors at fixed sites. Issuing separate licenses for each site as the need arises would not support the rapid deployment needed for disaster relief because of the time needed for the licensing process (safety and environmental reviews, hearings, etc.). To support such rapid deployments, the NRC would need to issue a license approving potential sites ahead of time (e.g., a license that would address safety and environmental issues for all potential operating sites within the United States). However, it would be difficult under the current regulatory framework to license commercial mobile micro-reactors for all potential operating sites. The NRC's approach to safety and environmental reviews presumes that a reactor will operate at a single site. Some of the specific technical requirements that would need to be satisfied in advance of deployment of a mobile micro-reactor are:

(1) The regulations in 10 CFR Part 100 establish approval requirements for proposed sites for power and testing reactors subject to 10 CFR Part 50 or 52. The regulations at 10 CFR Part 100 specify that the requirements are for "stationary" reactors, so the NRC staff would need to evaluate regulatory applicability (e.g., Could mobile reactors be considered "stationary" at each site of operation?) and how mobile micro-reactor applicants would meet the underlying purpose of the 10 CFR Part 100 requirements without knowing specific deployment sites ahead of time.[26]

(2) NEPA requires Federal agencies to evaluate the impacts of proposed federal actions on the human environment. The regulations in 10 CFR Part 51 form the basis for the NRC's NEPA compliance and direct the NRC staff in how to perform environmental reviews. As a federal agency, the NRC must assess the environmental effects of proposed actions prior to making decisions. Therefore, movement of a reactor to a site that had not been previously permitted or licensed would not be allowed under the current regulatory framework. The NRC has never attempted to perform an environmental review that would potentially cover deployment at innumerable, unspecified sites within the United States. The NRC staff would need to further evaluate the feasibility of performing such a review and possible appropriate ways to meet the NEPA requirements, perhaps through bounding site parameters, etc.

Transportation requirements for fueled micro-reactors are discussed in Section 5 of this enclosure. The transportation considerations for mobile micro-reactors would be the same as those described in Section 5 for factory-fabricated modules loaded with fuel that had been operated for some time (such as operational testing at a factory or operation at a deployment site). Transportation package certifications are issued in accordance with the requirements in 10 CFR Part 71 and can be used to support an unlimited number of transportation events without additional licensing. However, if the Commission does not approve the NRC staff's proposal in this paper for the use of features to preclude criticality, the reactor would be considered to be in

[26] For 10 CFR Part 52 design certifications, the NRC staff bases its safety review of the standard design on postulated site parameters that would bound a number of sites. However, applicants for particular sites perform site investigations and analyses to establish that their sites are appropriately bounded, and the NRC staff reviews this information for each site-specific application.

operation when loaded with fuel and could not be transported between various sites for as-needed, where-needed power production.

Near Term Strategy

The NRC staff does not intend to take actions in the near term to address changes to the regulatory framework that may be needed to support mobile micro-reactor licensing.

Next Steps

The NRC staff will monitor developments in the commercial sector related to deployment models and the demand for commercial mobile micro-reactors. If developers place additional focus on commercial mobile micro-reactor deployment, the NRC staff will assess the need for changes in the regulatory framework and Commission direction. Depending on the interest in applications for commercial mobile micro-reactors for particular uses (such as disaster relief), the NRC staff will consider the need to engage other federal agencies and the need for development of a new regulatory framework.