U.S. House Committee on Energy and Commerce
Subcommittee on Energy
"Nuclear Permitting Reform: Legislation to Advance Efficient Licensing."
June 9, 2026
Documents for the Record

1. March 2023, A report from the Nuclear Innovation Alliance titled "Improving the Effectiveness and Efficiency of the Advisory Committee on Reactor Safeguards," submitted by the Majority.
2. February 19, 2021, A letter from Nuscale Power, addressed to Margaret Doane of the U.S. Nuclear Regulatory Commission, submitted by the Majority.
3. June 9, 2008, A letter and piece of draft legislation from NRC Commissioner Dale Klein, addressed to Speaker of the House Nancy Pelosi, Submitted by the Majority.

Improving the Effectiveness and Efficiency of the Advisory Committee on Reactor Safeguards

March 2023

NUCLEAR INNOVATION ALLIANCE

Improving the Effectiveness and Efficiency of the Advisory Committee on Reactor Safeguards

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The contents of this report are based on interviews with over thirty-five individuals, including current and several former members of the ACRS, current and former ACRS staff, several former NRC Commissioners, former NRC Staff, and members of the ACRS stakeholder community.

Disclaimer:

The views expressed herein are those of the authors and the Nuclear Innovation Alliance.

March 2023

This report is available online at: https://nuclearinnovationalliance.org/resources

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I. Executive Summary

The Nuclear Innovation Alliance's ("NIA's") previous report, *Promoting Efficient NRC Advanced Reactor Licensing Reviews to Enable Rapid Decarbonization*,[1] suggested that "the Commission should systematically evaluate the [Advisory Committee on Reactor Safeguards] ACRS review process and how this can be appropriately aligned with the expectations that Congress set out for the Commission under [Nuclear Energy Innovation and Modernization Act ("NEIMA")]."[2] In light of this suggestion, NIA undertook its own extensive review of the ACRS to determine how it should better align with Congressional expectations under NEIMA without diminishing the significant role the ACRS has in the review and resolution of key technical issues associated with nuclear power plant regulation. Based on this review, the authors produced four main recommendations accompanied by specific proposed solutions. These recommendations broadly align with ACRS' own suggestions for the self-transformation presented to the Commission in 2019.[3] A brief description of each overarching recommendation and the takeaways from that recommendation are described briefly below.

1. The first overarching recommendation is to "Re-focus the Scope and Depth of ACRS Reviews."

In accordance with this recommendation, the ACRS should:

- focus on safety-significant matters and assist the NRC in meeting its statutory mandate to determine "that there is reasonable assurance"[4] "that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public."[5] (pg. 14)
- increase training, focus the scope of reviews, and use an action plan to further prioritize matters needing review. (pg. 15)
- consolidate duplicative Full Committee and Subcommittee meetings. (pg. 18)
- provide dates in the schedule for placeholder meetings. (pg. 18)

The Commission should:

- direct the ACRS to focus on novel and safety-significant issues in its reviews, and potentially refer specific matters to the ACRS with novel technical issues prior to review. (pg. 15)

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[1] Alex Gilbert, *Promoting Efficient NRC Advanced Reactor Licensing Reviews to Enable Rapid Decarbonization*, Nuclear Innovation Alliance (2021), *available at* https://nuclearinnovationalliance.org/licensingdurationsforclimatemitigation.

[2] *Id.* at 18.

[3] *See e.g.*, Letter from Peter C. Riccardella, Chairman of NRC Advisory Committee on Reactor Safeguards to Kristine Svinicki, NRC Chairman (Oct. 17, 2019), available at https://www.nrc.gov/docs/ML1929/ML19290F956.pdf (hereinafter "ACRS Transformation Letter"); ACRS, *Commission Meeting with the Advisory Committee on Reactor Safeguard (ACRS)* (Dec. 6, 2019), available at https://www.nrc.gov/reading-rm/doc-collections/commission/slides/2019/20191206/staff-20191206.pdf.

[4] AEA at § 185(b).

[5] AEA at § 182(a).

- establish timelines and milestones for ACRS reviews. (pg. 16)
- have OGC available to assist the ACRS in understanding the agency's statutory mandate. (pg. 16)
- communicate topics of interest to the NRC Staff in advance of meetings. (pg. 17)
- establish a hard deadline for the NRC Staff to provide documents in advance of the meetings and allow the meeting date to slip if the NRC Staff fails to meet its deadline. (pg. 17)
- exercise greater discipline on itself to limit the demands it places on the Staff to what is essential to ensuring adequate protection. (pg. 17)

The NRC Staff should:

- improve its preparation for engagements with the ACRS to better optimize the review of topics. (pg. 16)
- review its own practices in engaging with the ACRS, identify best practices that lead to efficient and effective ACRS reviews, and promote those best practices. (pg. 16)
- provide the ACRS with documents sufficiently in advance of ACRS meetings to allow for a fulsome review. (pg. 20)
- communicate the portions of the review that have the greatest potential safety significance. (pg. 20)
- engender a culture where the NRC Staff can feel empowered to raise concerns that the Committee is raising issues that are not safety significant. (pg. 20)

Finally, Congress should:

- revise the ACRS' statutory mandate in the Atomic Energy Act to emphasize that the ACRS should review only novel and safety-significant issues, and remove the requirement that the ACRS review all construction permit and operating license and renewal applications. (pg. 16)

2. The second overarching recommendation is to "Improve ACRS Operations and Management."

In accordance with that recommendation, the ACRS should:

- keep itself to approximately ten members. (pg. 21)
- diversify the background of ACRS members (drawing from former industry members, academics, and former national lab personnel or consultants). (pg. 22)
- relax experience requirements in certain areas of new state-of-the-art technology (e.g. artificial intelligence). (pg. 22)
- adhere to term limits. (pg. 22)
- not allow a single member to dominate the conversation for a particular subject area. (pg. 24)

The Commission should:

- implement the above suggestions for the ACRS to the extent that the ACRS cannot do so. (pg. 23)
- hire a consultant with expertise in organizational effectiveness to evaluate the manner in which ACRS members engage with the NRC Staff and licensees and suggest options for training and best practices on public and peer engagement. (pg. 25)
- incorporate components that screen for individuals who are independent yet collaborative and collegial when selecting ACRS members. (pg. 24)

The Executive Director for Operations in coordination with the ACRS should:

- not allow the ACRS to criticize, badger, or undermine individuals who are unable to answer ACRS questions on the spot. (pg. 24)
- request that ACRS members be able to set forth a brief explanation for why they are asking a question and tie it back to regulation (i.e., what is the member trying to understand and what is the safety concern). (pg. 24)

The ACRS Chairman should:

- ensure that debate among ACRS members is constructive, collegial, and within the ambit of its statutory purpose. (pg. 24)
- ensure that the views of individual ACRS members do not unduly chill or influence the views of the NRC Staff. (pg. 24)
- provide and maintain a safe space for respectful disagreement. (pg. 24)

3. The third overarching recommendation is "Reduce the Cost of ACRS Reviews."

In accordance with that recommendation, Congress should:

- amend the Atomic Energy Act to provide that all costs associated with ACRS reviews, including the cost of ACRS time be excluded from the fee recovery requirement. (pg. 25)
- amend the Atomic Energy Act to provide that all NRC Staff time used to prepare for ACRS meetings should not be billed to licensees and should also be excluded from fee recovery. (pg. 25)

4. The fourth overarching recommendation is to "Adjust Management of the ACRS."

In accordance with that recommendation, the Commission should:

- be more involved in the screening and selection of individual candidates who possess the knowledge, skills, and abilities necessary to address keys topics before the NRC. (pg. 26)
- be more involved in selecting or identifying the ACRS Chair and engage with the ACRS Chair on a regular basis. (pg. 26)
- provide the ACRS information on topics requiring ACRS review, particularly those that are novel or have significant safety implications. (pg. 26)

- discuss budgeting and prioritization with the ACRS. (pg. 27)
- set much of the agenda for the semi-annual meetings with the ACRS. (pg. 27)
- revamp the way that it interacts with the ACRS in meetings by eliminating meetings solely focused on repeating written material in paper filings. (pg. 27)
- ensure the ACRS Executive Director position is always filled by a seasoned executive who has technical credibility and sufficient weight and standing within the Commission to push back against the NRC Executive Director of Operations as well as the ACRS Chairman and members, and who has the experience needed to garner respect, as well as the savvy needed to deal with various disparate personalities. (pg. 27)
- perform a budget review of the ACRS staffing needs to ensure the Executive Director's organization is appropriately staffed to ensure it can meet the anticipated bow-wave of new reactor reviews. (pg. 27)

The authors' hope is that these recommendations (or some semblance thereof) will be implemented to position ACRS and NRC to successfully enable safe deployment of advanced nuclear energy.

II. Introduction

The Nuclear Innovation Alliance's ("NIA's") previous report, *Promoting Efficient NRC Advanced Reactor Licensing Reviews to Enable Rapid Decarbonization*,[6] suggested that "the Commission should systematically evaluate the [Advisory Committee on Reactor Safeguards] ACRS review process and how this can be appropriately aligned with the expectations that Congress set out for the Commission under [Nuclear Energy Innovation and Modernization Act ("NEIMA")]."[7] In light of this suggestion, NIA undertook its own extensive review of the ACRS to determine how it could better align with Congressional expectations under NEIMA without diminishing the significant role the ACRS has in the review and resolution of key technical issues associated with nuclear power plant regulation. The authors reviewed documentation relevant to the Committee, including statutory authority, requirements by NRC rule, and some ACRS meeting transcripts. The authors also interviewed over 35 individuals, including current and several former members of the ACRS, current and former ACRS staff, several former NRC Commissioners, former NRC Staff, and members of the ACRS stakeholder community. These interviews, which comprised over 100 hours of interview time, demonstrated a broad array of opinions regarding the ACRS, with several common themes described below.

Interviewees overwhelmingly indicated that the ACRS has the potential to be a valuable part of the NRC review process, but there was also uniform agreement that the ACRS and its processes need to be improved and modernized. A majority of interviewees indicated that the scope of ACRS reviews must be narrowed and focused on safety-related issues, the length of reviews must be reduced, and inefficiencies must be eliminated, particularly in light of the potential influx of a large number of advanced reactor reviews. Interviewees also indicated that the ACRS and the NRC Staff should work together to optimize the review process, control or reduce process costs for applicants, and maintain a positive working relationship between all stakeholders. In terms of ACRS membership, interviewees indicated that the ACRS should pursue greater diversity of members' viewpoints and experience. Interviewees also suggested that the Commission should be more involved with and set priorities for ACRS reviews. Finally, interviewees suggested the Executive Director of the ACRS[8] (a senior NRC Staff member who serves as the bridge between ACRS and the NRC Staff) is important to implementing some of the recommendations in this paper.

This paper provides a brief background of the ACRS and provides NIA's recommendations to enhance the value of the ACRS based on NIA's independent assessment of interviewee suggestions and NIA's review of ACRS' history. These recommendations can enable the ACRS and the NRC to implement numerous improvements to reduce the scope and length of ACRS reviews; increase focus on safety-relevant issues; gain greater efficiencies and optimize

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[6] Alex Gilbert, *Promoting Efficient NRC Advanced Reactor Licensing Reviews to Enable Rapid Decarbonization*, Nuclear Innovation Alliance (2021), *available at* https://nuclearinnovationalliance.org/licensingdurationsforclimatemitigation.

[7] *Id.* at 18.

[8] The Executive Director of ACRS coordinates technical, management, and administrative support of the statutory Advisory Committee on Reactor Safeguards (ACRS); provides overall program and management direction for ACRS administrative and technical support and associated resource management; and maintains liaison with the Commission, NRC staff, and others to provide for the conduct of ACRS in a manner responsive to the needs of the Commission.

interactions with the NRC Staff; reduce applicant, licensee, and NRC costs; optimize Committee membership and diversity; and improve Commission involvement in the Committee. These recommendations will enable the review of advanced reactors to focus on safety-related issues and to support the clean energy transition, improve energy security, and enable rapid decarbonization. These recommendations are also in line with ACRS' own suggestions for the ACRS transformation presented by the ACRS to the Commission in October and December 2019 to prioritize reviews on issues related to public health and safety, emphasizing risk significance and agency transformation priorities; stay abreast of staff transformation initiatives and continue to contribute to those initiatives; and improve operational efficiency.[9]

III. Background on the Advisory Committee on Reactor Safeguards

A. History

The Advisory Committee on Reactor Safeguards ("ACRS") was first organized in June 1947 when the Atomic Energy Commission ("AEC") established a blue-ribbon advisory group (the Reactor Safeguards Committee) "to evaluate the technical health and safety aspects of reactor hazards."[10] In 1950, the AEC created a second advisory group, the Industrial Committee on Reactor Location Problems, to "evaluate the scientific and environmental aspects of reactor locations."[11] In 1953, these two committees were combined to form the ACRS. All of these developments occurred in the infancy of the nuclear industry: the first nuclear reactor produced electricity in 1951 and the first commercial reactor did not come online until 1957.

Improved safety was a consideration in the first revision of the Atomic Energy Act of 1954, leading to the development of the ACRS as a statutory committee authorized to provide oversight on safety and report directly to the Commission under the Price Anderson Act of 1957.[12] Price-Anderson mandated that the ACRS review each power reactor or test facility application and that the ACRS reports be made public.[13] Prior to this statutory mandate, the AEC's advisory committees existed on an ad hoc basis.

With the enactment of the Energy Reorganization Act of 1974, the licensing functions of the AEC were transferred intact to the NRC. Since then, the ACRS has continued in the same advisory role to the NRC, with its responsibilities changing with the needs of the Commission. The ACRS consists of up to fifteen technical experts (as of this publication, the ACRS has ten members, with the NRC currently seeking to fill vacant positions[14]) who serve on a part-time

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[9] *See e.g.*, Letter from Peter C. Riccardella, Chairman of NRC Advisory Committee on Reactor Safeguards to Kristine Svinicki, NRC Chairman (Oct. 17, 2019), available at https://www.nrc.gov/docs/ML1929/ML19290F956.pdf (hereinafter "ACRS Transformation Letter"); ACRS, *Commission Meeting with the Advisory Committee on Reactor Safeguard (ACRS)* (Dec. 6, 2019), available at https://www.nrc.gov/reading-rm/doc-collections/commission/slides/2019/20191206/staff-20191206.pdf.

[10] *See* NRC, *ACRS History*, available at https://www.nrc.gov/about-nrc/regulatory/advisory/acrs/history.html.

[11] *Id.*

[12] P. Samantha, *NRC Regulatory History of Non-Light Water Reactors (1950-2019)*, at 2-2 (June 2019).

[13] *See infra* at 5.

[14] See NRC Notice, Seeks Qualified Candidates for Appointment to the Advisory Committee on Reactor Safeguards https://www.federalregister.gov/documents/2022/05/20/2022-10841/seeks-qualified-candidates-for-appointment-to-the-advisory-committee-on-reactor-safeguards.

basis[15] and meet as a Full Committee approximately ten times a year. They deliberate as a collective body, and the Full Committee provides advice to the Commission in the form of letters, reports, white papers, and memoranda – all of which may contain recommendations for the Commission's consideration.

In addition to the Full Committee, as of this writing, the ACRS has the following subcommittees:

- Planning and Procedures Subcommittee (which meets each month the Full Committee meets)
- Digital Instrumentation and Controls ("I&C")
- Metallurgy and Reactor Fuels
- Accident Analysis
- Regulatory Rulemaking, Policies, and Practices
- Plant Operations and Fire Protection
- Radiation Protection and Nuclear Materials
- Probabilistic Risk Assessment ("PRA")
- Fuels, Materials, and Structures
- Future Plant Designs
- Design Centered Licensing
- Application/Design-Specific Subcommittees
    - SHINE
    - Kairos
    - NuScale
    - BWRX-300

There are approximately 10 Full Committee and 50 Subcommittee meetings every year.[16]

Today, the ACRS has four primary purposes:

1. to review and report on safety studies and reactor facility license and license renewal applications;
2. to advise the Commission on the hazards of proposed and existing production and utilization facilities and the adequacy of proposed safety standards;
3. to initiate reviews of specific generic matters or nuclear facility safety-related items; and
4. to provide advice in the areas of health physics and radiation protection.[17]

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[15] The ACRS is subject the Federal Advisory Committee Act ("FACA") and is composed of Special Government Employees who are not fulltime Federal government employees.

[16] *ACRS Charter*, available at https://www.nrc.gov/docs/ML2033/ML20337A117.pdf.

[17] NRC, *Advisory Committee on Reactor Safeguards*, available at https://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html.

The ACRS performs these functions through its meetings, publications (primarily through the use of letter reports[18]), and interactions with NRC Staff, Commissioners, licensees, applicants, and external stakeholders.

The cost of the review activities of the ACRS is subsumed into the overall Part 171 fee structure that the utilities and others pay in their yearly general license fees. Thus, the ACRS itself is not off the fee base. However, the NRC Staff cost associated with preparing for and attending ACRS meetings is a cost charged directly to applicants for a specific license review under Part 170.

B. Authorizing Statute

As noted previously, the ACRS is statutorily mandated. Section 29 of the Atomic Energy Act of 1954 mandates that:

> There is hereby established an Advisory Committee on Reactor Safeguards consisting of a maximum of fifteen members appointed by the Commission for terms of four years each. The Committee shall review safety studies and facility license applications referred to it and shall make reports thereon, shall advise the Commission with regard to the hazards of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards, and shall perform such other duties as the Commission may request. One member shall be designated by the Committee as its Chairman. The members of the Committee shall receive a per diem compensation for each day spent in meetings or conferences, or other work of the Committee, and all members shall receive their necessary traveling or other expenses while engaged in the work of the Committee. . . [19]

The ACRS' broad scope of optional and required reviews is also set forth in the Act:

> The Advisory Committee on Reactor Safeguards shall review each application under section 103 or section 104b. for a construction permit or an operating license for a facility, any application under section 104c. for a construction permit or an operating license for a testing facility, any application under section 104a. or c. specifically referred to it by the Commission, and any application for an amendment to a construction permit or an amendment to an operating license under section 103 or 104a., b., or c. specifically referred to it by the Commission, and shall submit a report thereon which shall be made part of the record of the application and available to the public except to the extent that security classification prevents disclosure.[20]

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[18] NRC, *Advisory Committee on Reactor Safeguards (ACRS) Letter Reports*, available at https://www.nrc.gov/reading-rm/doc-collections/acrs/letters/index.html.

[19] See Atomic Energy Act of 1954, as Amended, available at https://www.nrc.gov/docs/ML1327/ML13274A489.pdf#page=23.

[20] Atomic Energy Act of 1954, as Amended, available at https://www.nrc.gov/docs/ML1327/ML13274A489.pdf#page=23 Page 150.

The ACRS is further governed by NRC regulations, particularly those set forth in 10 CFR Part 7, the ACRS Charter[21] (required under the Federal Advisory Committee Act to specify the Committee's mission or charge, specific duties, and general operational characteristics), and the ACRS Bylaws[22] (describing the procedures to be used by the ACRS in performing its duties, and the responsibilities of the members).

IV. Recommendation 1: Re-focus the Scope and Depth of ACRS Reviews.

A. Problem Statement

There is no doubt that the ACRS works hard to provide the Commission with an independent analysis of the matters under its purview. However, many interviewees indicated that changes in the NRC and its scope of work over the decades along with a static statutory mandate has increased ACRS engagement in matters where the ACRS provides less overall value. Therefore, the ACRS needs modernization.

When the ACRS was first created in 1947 and statutorily mandated in 1957, the AEC was tasked with licensing the United States' first commercial nuclear power reactors. The AEC was a new Federal agency, responsible for both promoting and regulating nuclear power, and its Staff had limited experience with evaluating nuclear technology. The first reactors often included a variety of different reactor technologies (e.g., pressurized water reactors, boiling water reactors, and sodium fast reactors), and there were significant differences between subsequent reactors of the same type based on rapid advances in the state of the art and siting-based modifications. In light of the circumstances, the ACRS provided significant value reviewing each application for a construction permit and operating license for new nuclear power plants and providing an independent analysis of each design and site—separate and apart from the Staff of the AEC. This made sense, given the state of industry and AEC Staff knowledge at the time: virtually everything was novel; without appreciable experience, there was little or no understanding of what was considered safety significant; and therefore a broad range of topics warranted review from independent experts. The ACRS' statutory framework reflects this reality. With the lack of experience at the Atomic Energy Commission and subsequently the NRC in deploying and regulating light water nuclear reactors, the ACRS played a useful role in raising fundamental issues related to reactor design, operation, and siting to support the licensing of these first reactors.

The industry and the NRC have changed drastically over the course of sixty-five years since this statutory framework was first established. The industry has since gained tremendous experience licensing and safely operating light water reactors, with over 100 reactors licensed, most reactors in an extended period of operation under a new license, and some reactors pursuing Subsequent License Renewal (SLR) or entering their second period of extended operation. The NRC Staff, likewise, has significant experience and technical knowledge of light water reactors that did not exist when the ACRS was first created. At this point, numerous members of the NRC Staff and the industry have decades of experience operating, maintaining, or regulating light water reactors, and this level of knowledge and familiarity with these designs is functionally equivalent

[21] *ACRS Charter*, available at https://www.nrc.gov/docs/ML2033/ML20337A117.pdf.

[22] *ACRS Bylaws*, available at https://www.nrc.gov/docs/ML2121/ML21217A060.pdf.

(or may exceed) the experience of some members of the ACRS. For this reason, it is timely to consider a different role for the ACRS as it no longer serves the same role as its predecessor did in the 1950s-60s during the dawn of the nuclear age.

Numerous interviewees indicated challenges arising from this evolution. For one, the ACRS is still required to provide an outside assessment of every application for a construction permit or operating license, irrespective of standardization or design maturity, including some license renewals for facilities that have operated safely for many decades.[23] Some interviewees were concerned that, as more advanced and new reactors pursue licensing while, in parallel, existing reactors pursue subsequent license renewal, the requirement of the ACRS to review every application could quickly fill the ACRS calendar, making it more difficult for advanced and new reactors to obtain a slot on the meeting schedule and delaying completion of mandated licensing reviews. Other interviewees thought that the statutory requirements would continue to drive the ACRS to focus too heavily on areas with little or no added benefit, including repetitive reviews of light water reactor technologies or advanced reactor technologies that have already undergone NRC and ACRS review (i.e., nth of a kind reactors).

Finally, a number of interviewees noted that some individual ACRS members have pursued questioning on matters of personal and professional curiosity that were not intrinsic to a safety concern and that consumed precious time that could have been better spent on matters of potential safety significance. Given the fact that NRC, applicant, and licensee resources are finite, and the NRC is a fee-based agency, ACRS leadership and members should exercise discipline and avoid lines of inquiry that are not germane to the safety decision so as to avoid burdening the NRC Staff, licensee or applicant with unnecessary costs and delays. More importantly, there is a public interest in the NRC and the ACRS using their time to focus on significant safety issues.

On the other hand, several interviewees indicated that the ACRS is at its best when reviewing novel regulatory issues and that the ACRS has served a beneficial role in helping the NRC Staff resolve these issues. One example is the ACRS review of reduced emergency planning zones commensurate with reduced risk from advanced reactors, which provided additional support for the NRC's Staff determination. It is NIA's understanding that the ACRS also worked closely with the NRC Staff to develop the agency's current risk-informed regulations[24] to the benefit of the NRC Staff. Some interviewees also indicated that the ACRS can be valuable to the Commission in reinforcing expectations to modernize and evolve regulatory perspectives. NIA also understands that the ACRS has authored useful generic papers regarding advanced reactor technology, including a short paper on the technology of pebble bed reactors and the challenges associated with the deployment of this technology. These papers can enable technical knowledge management and knowledge transfer to NRC Staff.

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[23] We understand that the ACRS has stated that they need to review every topical report written by advanced reactor applicants; however, we do not agree that such review is required particularly to the extent that Topical Reports are building on known technology or methods.

[24] See "Risk-informed regulation" definition on the NRC website, https://www nrc.gov/reading-rm/basic-ref/glossary/risk-informed-regulation html.

These observations are generally consistent with the ACRS' own analysis of Commissioner input on where ACRS engagement is most effective, as set forth in the ACRS Transformation Letter.[25] In that ACRS Transformation Letter, the ACRS stated that its engagement is most effective in:

- "Risk-Informed Decision Making – Several commissioners opined that the most important role the ACRS can play is to continue its firm support of and advice regarding risk-informed decision making."[26]

- "Digital I&C – This was identified as an important topic, and one commissioner stated that NRC and the nuclear industry are far behind where they should be on this topic." [27]

- "Research Reviews – Research reviews were identified by several commissioners as an important area for continued ACRS involvement."[28]

- "New Technologies and Reactor Types –ACRS is a highly competent group of dedicated experts from outside the agency, encompassing a broad range of disciplines, who dig deeply into the matters subject to the Committee's review. The ACRS members' independent technical assessments assist the NRC staff in making high-quality regulatory evaluations."[29]

In sum, interviewees generally agreed that the ACRS should optimize the use of its finite resources by focusing on safety-significant[30] issues, commensurate with risk, and novel technologies. However, some interviewees disagree that ACRS engagement has been effective in digital I&C and instead believe the ACRS has contributed to the failure of the agency and industry to make progress on digital I&C deployment.[31] In addition, some interviewees suggested that the ACRS could reduce the length of reviews and remove inefficiencies with a few relatively modest modifications as elaborated below.

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[25] *See supra* at n.4.

[26] ACRS Transformation Letter at 3.

[27] ACRS Transformation Letter at 3. [28] ACRS Transformation Letter at 3.

[28] ACRS Transformation Letter at 3.

[29] ACRS Transformation Letter at 3.

[30] See NRC definition of "safety-significant" https://www.nrc.gov/reading-rm/basic-ref/glossary/safety-significant.html. While the definition of safety significant is broad, it is worth noting that some technologies undergoing licensing by the NRC have been around for decades and, as such, are not novel per se; they just have not been used in commercial applications. For example, air-cooled condensers are widely used in industrial applications but not in nuclear power generation, and they have no safety function. Unless there is a nexus to safety, the ACRS should not invest significant time reviewing these components.

[31] Based on these interviews, the ACRS does not possess a state-of-the-art understanding of the status of digital I&C in nuclear, energy and process industries. In Section IV.B.1.a, NIA recommends steps that should be taken to improve ACRS's awareness of state-of-the-art approaches.

The following sets forth proposed solutions to address these issues.

B. Proposed Solutions

There are several ways to implement greater ACRS focus on safety-significant issues and novel technologies or novel approaches to regulatory issues. The following sections explore implementation pathways for various stakeholders including the ACRS itself, the Commission, the NRC Staff, and Congress. Ideally, all stakeholders would pursue improvements in concert with one another.

1. Solution 1. Focus ACRS review on safety-significant issues and novel technology or approaches to regulatory issues.

a. The ACRS

Recent ACRS Chairmen have been moving to streamline the level of activities that are reviewed by the Committee, as discussed further in subsequent sections. While this is positive, further changes would improve the operational effectiveness of the ACRS. As an initial matter, the ACRS could improve training for new members on the requirements of the Atomic Energy Act. NIA understands from some interviews that ACRS training is focused on requirements relating to managing their time, avoiding conflicts of interest and other administrative matters, but that no formal training is conducted on the roles and responsibilities of ACRS members in serving the Commission. Since the ACRS exists specifically to assist the Commission to meet the requirements of the Act, the members should receive appropriate training on how the ACRS helps the Commission perform its duties. At a minimum, ACRS members should receive training on the requirement that licensees demonstrate a reasonable assurance of adequate protection and what that means. The ACRS should also implement training on the NRC's regulations and how they are implemented, including the rulemaking and licensing review process so that ACRS members better understand how the Agency operates as a whole. As part of that training, ACRS members should have an opportunity to tour operating nuclear power plants and non-power reactors to gain a better practical understanding of how these facilities are operated, consistent with the ACRS-EDO MOU[32], which already contemplates the possibility that the ACRS might visit licensee facilities.[33] A greater understanding of these topics should help the ACRS focus on the safety-significant issues where it is most needed and avoid areas which do not add value in helping the Commission meet its mission under the Act or involve areas outside of the legal or regulatory mandate of the ACRS. Because of its broad benefits, training costs should not be borne by applicants or licensees.

Interviewees that served on the ACRS also indicated that the ACRS Chairman and subcommittee chairs could more proactively reign in members who pursue time-consuming questions with no safety significance. Indeed, the "Chairman of the Committee is empowered to conduct the

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[32] *See ACRS-EDO Memorandum of Understanding* (2018 Revision), released in response to a FOIA request and available at ML19018A064 (hereinafter ACRS-EDO MOU).

[33] Interviewees also suggested that the ACRS could benefit from informal briefings from the NRC Staff on technical issues or new technologies, a topic that we address in more detail in Solution 2.

meeting in a manner that, in his/her judgment, will facilitate the orderly conduct of business."[34] The ACRS already screens out issues of low safety significance, like "requests for power uprates less than 7 percent, requests for plants to operate in the expanded power to flow domain, and some license renewal applications,"[35] and the ACRS has also established criteria for in-depth reviews.[36] The ACRS is attempting to move to a new reactor design certification application review process that focuses on key, risk-significant issues that are cross-cutting over the application.[37] The ACRS in the past has also used an action plan, NUREG-0286, to focus its review on key issues. The ACRS could formalize a screening or prioritization process, perhaps by updating NUREG-0286[38] every year, and applying it to all ACRS reviews to eliminate previously reviewed or repetitive issues, or those of low safety significance. For example, the ACRS could screen out reviews in cases with no offsite consequences that would impact public health and safety, nth of a kind reactor reviews, or in cases such as uprates with no new issues of potential safety significance.

b. The Commission

The Commission should direct the ACRS to focus on novel and safety-significant issues in its required reviews, either informally through a change in the ACRS Charter, or the NRC rules applying to the ACRS. As it stands, the ACRS does not perform an in-depth review of every aspect of every license application: such a review would be impracticable, if not impossible. It would be logical for the Commission to direct the ACRS to focus specifically on those matters most likely to impact the Commission's required finding "that there is reasonable assurance"[39] "that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public."[40]

Re-reviewing technology or methods[41] that have already been reviewed by the ACRS during previous licensing reviews or focusing on matters of little or no safety significance ultimately does not assist the Commission in making the findings that it is required to make under the Atomic Energy Act, and the Commission should ask the ACRS to focus on new matters that are relevant to the findings required by law. The Commission can also specifically refer anticipated

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[34] 84 Fed. Reg. 27662.

[35] ACRS Transformation Letter at 3.

[36] ACRS Transformation Letter at 3. ("The following criteria will be used to set priorities for our in-depth reviews: - Does the issue affect public health and safety? - Does the issue relate to one of the four agency transformation initiatives (i.e., risk-informed decision making; 10 CFR 50.59 flexibility; licensing of non-[light water reactors ("LWRs")]; or digital I&C safety design principles)? - Does the issue involve new methods or technologies, or is it a routine matter that we have reviewed numerous times before, and for which the staff processes are mature and technically advanced? - Is the activity directed by the Commission? - Other criteria that future staff transformation activities may identify.")

[37] *Id.*

[38] We heard in our interviews that prior members of the ACRS found this NUREG to be helpful.

[39] AEA at § 185(b).

[40] AEA at § 182(a).

[41] The ACRS should be directed to consider prior reviews as final. In other words, if they have previously reviewed a safety methodology in a Topical Report, they should not re-open those topics in an application. If that cannot be done, then the ACRS should not be reviewing the Topical Report in the first place.

matters (including novel technical issues) to the ACRS for review before they arise in an application. Such matters could be flagged by NRC Staff as they encounter novel technical issues during pre-application engagement with developers and applicants.

In addition to narrowing the scope of ACRS reviews, the Commission could also establish timelines and milestones for ACRS reviews, similar to those found in 10 CFR Part 2 for adjudicatory proceedings.

Finally, the Commission should require the ACRS to have its own assigned counsel from the NRC's Office of General Counsel at the table for Full Committee meetings, similar to the role that the NRC General Counsel plays at the meetings of the Commission itself. This Counsel would be able to provide guidance on legal and regulatory interpretation and could assist the ACRS Chairman in ensuring that the ACRS is focused on matters within its statutory and Commission-instructed mandate to provide reasonable assurance of adequate protection. Several interviewees indicated that ACRS members have, from time to time, made legal or regulatory interpretations on matters for which they lack this expertise or where such interpretation is outside their scope. Having a legal counsel at the table could help to avoid situations where the ACRS is opining on areas outside of their expertise or legal mandate. The Commission has been well served by having OGC represented at the table during its Commission meetings, and ACRS could similarly benefit.

c. The NRC Staff

The NRC Staff should improve its preparation for engagements with the ACRS to better optimize the review of topics. For example, the NRC Staff should proactively identify novel aspects of design prior to ACRS review for specific consideration. The NRC Staff should also identify and rank systems in proposed designs based on the potential to impact safety. The NRC Staff should review its own practices in engaging with the ACRS, identify best practices that lead to efficient and effective ACRS reviews, and promote those best practices.

d. Congress

Finally, Congress should revise the ACRS' statutory mandate in the Atomic Energy Act to emphasize that the ACRS should review only novel and safety-significant issues at the direction of the Commission and remove the requirement that the ACRS review all construction permit and operating license and renewal applications. By focusing on novel elements of an application, the ACRS' unique capabilities could be better leveraged. This could include, for example, novel designs (e.g., NuScale's original passive[42] LWR design application), novel issues (a significant modification in a design or the use of that design), novel technologies (e.g., reactors with molten salt coolants), first-of-a-kind reactors or facilities, novel regulatory

[42] NuScale's recent Standard Design Approval application presents an opportunity for the ACRS to demonstrate a streamlined approach. The ACRS should limit its review to those aspects of the design that are new or that have the potential to be of such significance they undo the agency's prior reasonable assurance finding.

activities (e.g., the first subsequent license renewal[43]) or applications with unique site-specific aging issues (e.g., alkali-silica reactions at Seabrook).

There is likewise no need for the ACRS to conduct reviews of the applications for the siting of every new reactor at the construction permit phase, or for license applications that have had large portions previously reviewed and approved (e.g., siting of standardized reactor designs at existing nuclear power plant sites). Nor should the ACRS focus heavily on reviewing the aspects of reactor technologies that are well known and well understood, even if they are being incorporated into a new reactor design. (e.g., light-water reactors, radiation protections, atmospheric transport). Revising the ACRS' statutory mandate to eliminate the requirement to perform duplicative reviews of routine matters makes sense and would enable the ACRS to focus on the safety issues where it can have the most impact.[44]

2. Solution 2. Reduce the length of and improve timing of ACRS reviews, eliminate repeat reviews, and reduce time spent on NRC Staff preparation.

Interviewee opinions varied on the length and timing of ACRS reviews. Some interviewees stated that the NRC Staff would frequently issue documents after the agreed-upon deadline, compressing the time for ACRS members to review them and formulate a collective opinion on their adequacy. The ACRS-EDO Memorandum of Understanding requires the NRC Staff to provide documents to the ACRS four weeks in advance of meetings,[45] but the NRC Staff frequently does not meet that deadline. Other interviewees indicated that the ACRS meeting schedule drove delays due to limited schedule openings and the need to set the overall review schedule months in advance. If the NRC Staff were to miss a meeting, it could cause a cascading delay since the busy ACRS calendar might result in a meeting delay of several months.

The authors also heard scheduling complaints about the prevalence of repetitive Full and Subcommittee reviews. The Subcommittee is supposed to be "comprised of three to six members with the relevant expertise"[46] including the "ACRS members who are most cognizant of the technical details of issues" under review.[47] These topics are then put up again for "later consideration by the full membership during Full Committee meetings."[48] In practice, however, many Subcommittee meetings include the majority of the Committee. According to ACRS meeting transcripts, of seven of the Subcommittee meetings in August 2022, September 2022, and October 2022, two had nine members in attendance, four had eight members in attendance,

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[43] In any event, SLRs should only be focused on longstanding challenging issues that could potentially be exacerbated by further aging. Additionally, if ACRS activities remain the same, someone that volunteers to be the first SLR should have reduced or waived fees to participate in this activity.

[44] Further, while we are not suggesting the elimination of the ACRS, we would note that every one of NRC's peer nuclear regulators effectively reviews new reactor designs without having the additional equivalent of the ACRS.

[45] *See* ACRS-EDO MOU.

[46] *See* Federal Advisory Committee Act ("FACA") Database Entry – Advisory Committee on Reactor Safeguards, available at https://www facadatabase.gov/FACA/apex/FACAPublicCommittee?id=a10t0000001gzx0AAA (hereinafter FACA Database Entry - ACRS).

[47] ACRS-EDO MOU.

[48] *See* FACA Database Entry – ACRS.

and one had seven members in attendance.[49] In comparison, the October 2022 Full Committee meeting had eight members in attendance. It seems completely unnecessary for the ACRS to pull NRC Staff members and licensees into Subcommittee meetings in addition to Full Committee meetings when the level of attendance is the same at both meetings, particularly when there is no procedural benefit as the ACRS "Subcommittee meetings are conducted under the same FACA procedures as the Full Committee meetings."[50]

In addition to scheduling issues, some interviewees observed that the ACRS members sometimes asked questions on issues resolved months prior during the NRC Staff review, unnecessarily bringing up issues that the parties believed were already resolved. Many interviewees also indicated that, in their opinions, the NRC Staff has, at times, spent an inordinate amount of time and effort preparing for ACRS meetings, resulting in significant costs and potentially more delays. One interviewee stated that the majority of the time, the NRC Staff tends to defer to ACRS determinations rather than stand their ground on their own analysis in their Safety Evaluation Reports (SERs). This in turn extends the review period, duplicates efforts, and ties up valuable NRC Staff resources in addition to increasing costs as the NRC Staff modifies their Safety Evaluation. Having this interaction occur at the end of NRC Staff review is viewed as an unnecessary delay in the overall process.

Regardless of the ultimate cause, interviewees clearly agreed that the NRC Staff and ACRS could improve coordination to reduce the possibility for applicant costs and delays. Below are some suggestions of improvements that could be implemented by the NRC Staff and ACRS. The Commission could also require the NRC Staff and ACRS to implement these suggestions.

a. The ACRS

First, as described above, the ACRS should eliminate duplicative Subcommittee and Full Committee meetings. Where the same number of members attend each, and the same procedures apply, there is no reason for both sets of meetings to occur. In these cases, the ACRS should consolidate its review into only a Full Committee meeting (with the depth of a Subcommittee meeting). Alternatively, the Subcommittee could perform a fulsome review and only refer key items up to the full committee, with the Full Committee adopting the remainder of the Subcommittee review. These suggestions, in addition to modifying the ACRS' scope of review to focus on novel matters of potential safety significance, would enable the ACRS to free up meeting slots on its schedule in order to implement the next suggestion (see Section 2.b).

The ACRS should modify its schedule to include informal "placeholder" meetings. (Of note, the authors would not recommend these meetings unless all ACRS costs (both direct costs and those associated with all Staff preparation time) were moved off-fee, as recommended later in this

[49] According to ACRS meeting transcripts, the August 16 meeting on Plant Operations and Fire Protection and the October 18 meeting on Part 53 each had 9 members in attendance. The September 23 meetings on Digital Instrumentation & Control and Regulatory Policies & Practices, the October 17 meeting on Kairos, and the October 20 meeting on Accident Analysis: Thermal-Hydraulics each had 8 members in attendance. The October 4 meeting on Metallurgy and Reactor Fuel had 7 members in attendance. *See generally*, 2022 ACRS Meeting Schedule and Related Documents, available at https://www nrc.gov/reading-rm/doc-collections/acrs/agenda/2022/index html.

[50] *See* FACA Database Entry – ACRS.

report). These meetings would serve two purposes. During the “placeholder” meetings, the NRC Staff could give a series of short, one-hour updates on developing topics and in-progress reviews (with a focus on novel matters of potential safety significance). This would give the ACRS an established and scheduled advanced opportunity to gain familiarity with topics and ask early questions or give early feedback, instead of leaving ACRS reviews until near the end of an NRC Staff review when it is most likely to cause delays.[51] However these meetings should not be commensurate with an in-depth review. Placeholder meetings could also serve as a timeslot for makeup meetings if another meeting were to be delayed, since rescheduling a placeholder meeting would not impact any specific review schedules. Of note, while not entirely the same, the implementation of placeholder meetings is consistent with the ACRS-EDO MOU which contemplates that the ACRS will “convene informal meetings with the staff for the purpose of preparation for subcommittee and Full Committee meetings, or to receive updates on various technical matters that would facilitate the planning of the ACRS meetings.”[52] It is also consistent with the ACRS Transformation Letter, which contemplates that the ACRS “will arrange for periodic updates” on NRC transformation initiatives.[53] Nor would this be the first placeholder meeting on the ACRS schedule, as the ACRS has previously held such meetings, including an open dialogue with the NRC Staff on placeholders from 10 CFR Part 50 and Part 52 reviews.[54]

Interviewees indicated that ACRS meetings proceed more smoothly when the NRC Staff and the ACRS communicate about topics of interest prior to the meeting. This allows the NRC Staff to focus its preparation efforts, to ensure that the proper Staff members are in attendance, and to prepare responses to questions. While the ACRS and NRC Staff already coordinate in advance on meeting topics,[55] the authors recommend that the ACRS issue a written notice to the NRC Staff, more than one week in advance of the ACRS meeting, with more detailed topics of special interest that the NRC Staff should be prepared to actively discuss. An example topic of interest from recent meetings might be: “The use of Quantitative Health Objectives (“QHOs”) in Part 53, including the basis, justification, and need for including QHOs in the rule.” The authors also recommend that the ACRS and the NRC Staff agree as a general matter that some questions may be best addressed in writing after a meeting is complete. Interviewees voiced competing

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[51] One example of the benefit of early engagement and familiarization is the NRC Staff’s recent engagement with the ACRS on licensing fusion systems under the NRC’s Part 30 framework. The ACRS criticized the NRC Staff’s recommended approach and submitted a letter to the Commission advocating for a different framework. However, it is apparent that the ACRS did not review the public record to familiarize itself with material presented at NRC public meetings over the two-year period that led to the NRC Staff’s development of the proposed framework. Early informal engagement from the NRC Staff setting forth that history would have allowed ACRS members an opportunity to better familiarize themselves with the NRC Staff’s proposed framework and to develop a clearer understanding of the basis of the NRC Staff’s present analysis. The introduction of scheduled meetings specifically left open for early engagement on developing topics would provide the NRC Staff with a necessary forum going forward to better address potential misunderstandings early on.

[52] *See* ACRS-EDO MOU.

[53] ACRS Transformation Letter at 4.

[54] As an example, on September 20, 2019, the ACRS called a meeting with the NRC Staff to discuss lessons learned from Part 50 and Part 52 reviews.

[55] “The Program Office technical contact and the ACRS staff contact should work together to prepare the meeting agenda.” ACRS-EDO MOU.

opinions as to whether it is acceptable for the NRC Staff to provide supplemental responses to the ACRS after a meeting (although it is explicitly allowed in the ACRS-EDO MOU[56]) and having a more robust formal policy in place to allow supplementary responses might temper NRC Staff tendencies to have all the answers up front and over-prepare for ACRS meetings with numerous dry runs. However, licensees should not be required to provide supplemental answers after ACRS meetings.

Providing discussion topics in advance of a meeting would require the NRC Staff to meet its deadlines and provide documents well in advance of the meeting date. The ACRS should establish a hard deadline for the NRC Staff to provide documents in advance of the meetings and allow the meeting date to slip if the NRC Staff fails to meet its deadline. If the NRC Staff is failing to meet deadlines and provide documents well in advance of meetings, allowing the meetings to slip will provide a public record of those issues for the Commission's further consideration.

Finally, multiple interviewees indicated that the ACRS places unreasonable demands on the NRC Staff to be prepared to respond to questions on topics that are not safety-significant, is insufficiently respectful of the NRC Staff's time, and sometimes even treats the NRC Staff disrespectfully. In response, the NRC Staff go to extraordinary lengths to prepare for ACRS meetings. The ACRS should recognize that NRC Staff time is a valuable resource. The ACRS should exercise greater discipline on itself to limit the demands it places on the Staff to what is essential to ensuring adequate protection.

b. The NRC Staff

Interviewees indicated that the NRC Staff frequently fails to provide the ACRS with documents sufficiently in advance of ACRS meetings to allow for a fulsome review. The NRC Staff should exercise better time management and provide the ACRS with documents for review at least four weeks in advance of ACRS meetings, as a hard deadline., consistent with the ACRS-EDO Memorandum of Understanding. The NRC Staff should also communicate the portions of the review that have the greatest potential safety significance. Having sufficient time for the review should allow the ACRS to provide topics of discussion in advance of meetings and should allow the NRC Staff to narrowly tailor its preparation for those meetings. The NRC should engender a culture where the NRC Staff can feel empowered to raise with the ACRS that the Committee is raising issues that are not safety significant.

V. Recommendation 2: Improve ACRS Operations and Management.

A. Problem Statement

The ACRS generally has a collegial atmosphere, and many interviewees focused on the importance of recruiting highly qualified and technically balanced members for the ACRS. As a general matter, most interviewees believe that the current membership of the ACRS is well

[56] "ACRS subcommittee members may ask for additional information about the documents under review that were not supplied prior to the meetings. The Program Office technical contact and other program office staff supporting the ACRS meeting should endeavor to provide the additional information, if it is reasonably available after the subcommittee meeting, to the ACRS staff contact." ACRS-EDO MOU.

informed and conscientious. However, interviewees agreed that there could be improvements in the diversity of viewpoints on the ACRS, and some cautioned against increasing the membership of the ACRS.

The ACRS has generally improved in maintaining respectful and collegial relations with the NRC Staff and licensees over the last twenty years. However, there are two exceptions to this progress. Certain ACRS members hold such strong views on their technical topics of interest that the NRC Staff is overly deferential when addressing those specific topics, knowing that any relevant regulatory initiatives will be subject to vigorous challenge from the ACRS. Certain ACRS members have recently engaged in unprofessional behavior like reading newspapers during review meetings.

B. Proposed Solutions

1. Solution 1. Keep the number of ACRS members manageable, hire the best possible members, and ensure a diversity of viewpoints.

a. The ACRS and the Commission

Both the ACRS and the Commission share the burden when it comes to hiring the best possible members for the ACRS and optimizing its numbers. As a result, the suggestions here are directed at both organizations. While the Commission could engage in rulemaking to place greater specificity on how the ACRS performs hiring and staffing, the Commission is more likely to implement these suggestions on an informal basis as is the ACRS. However, the authors strongly recommend that the Commission consider these recommendations, particularly as to term limits and the diversity of viewpoints, and push for implementation even if it is informal.

First, based on interviews, including suggestions from a former ACRS Chairman, the authors suggest that the ACRS keep itself to approximately ten members. Some interviewees indicated that having fifteen ACRS members was too many, and it hindered both collegiality and efficiency of the ACRS. As one interviewee phrased it, it takes more time to write a letter with fifteen different opinions as input. In addition, one former ACRS Chairman felt that having nine or fewer members was the most effective approach to fostering collaboration.

That former ACRS Chairman also suggested that the ACRS should be composed, with equal representation, of former industry members, academics, and former national lab personnel or consultants. Indeed, multiple interviewees with experience serving on the ACRS found that it was helpful to have participation from individuals with utility experience, and most interviewees thought that a diversity of work history and educational backgrounds would benefit the ACRS. This is consistent with the ACRS' own Membership Balance Plan which emphasizes the importance of diverse points of view.[57] That said, it would help to have more individuals with experience running nuclear power plants in the United States.

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[57] *See* ACRS, *Membership Balance Plan* at 3 (Dec. 2020), available at: https://gsa-geo.my.salesforce.com/sfc/p/#t0000000Gyj0/a/t0000001F3Xu/eFQkKnXVMnaf6p.zZ1Jnw8LZkrKsl7KgUKlGAR_jGus

In addition, while the current ACRS has a variety of backgrounds, current ACRS members are required to have decades of experience in their narrow fields of technical expertise. This can limit the diversity of viewpoints within the ACRS, particularly for new and innovative technologies. The Commission should consider redefining these experience requirements in the interest of balancing technical expertise with applied experience in nuclear power generation and knowledge of integrated plant operations. Additionally, a limited number of ACRS members should be recruited from the industry to provide front-line engineering and operations experience in areas of cutting-edge technology that are subject to rapid development and change and for which technical expertise may become quickly outdated (for example, recent advancements in fusion technology, digital instrumentation and controls, computing advances, and advanced reactor designs). Numerous interviewees opined that it does not benefit the ACRS to rely on decades-old experience in these areas of rapid development as the prior experience may not be fully relevant to state-of-the-art practices. Moreover, if the ACRS is going to effectively drive transformation and modernization of the NRC and its regulatory regimes, it must include members who break with outdated views and practices held over from large light-water reactor designs and operations. The ACRS should also recruit experts from outside the nuclear industry to provide perspective on rapidly developing areas of technology like digital instrumentation and control, automation in lieu of human operation, and artificial intelligence.

Further, in order to improve the ACRS' performance, there should be greater adherence to term limits on ACRS members. Moreover, term limits should be only two, or in rare circumstances three, consecutive four-year terms (i.e., eight years on the Committee with twelve years in only rare circumstances). Such a limit would be only slightly more limiting than the ACRS Membership Balance Plan, which already states that "absent unusual circumstances, [members] do not serve more than three, four year terms," and "members are reappointed in excess of this period only if there is a compelling continuing need for their expertise."[58] Indeed, the ACRS could simply revise the Membership Balance Plan to state that "absent unusual circumstances, [members] do not serve more than ***two***, four year terms," with an allowance to extend that period in compelling circumstances. If the Commission is unable or unwilling to implement a term limit for ACRS members, whether informally or by rule, Congress could modify the Act to implement such a limit.

From a practical perspective, interviewees made the additional following suggestions for optimizing ACRS membership. While the ACRS may be undertaking some of these actions, the Committee (or to the extent necessary, the Commission) should continue to expand these actions to help the ACRS be more effective:

- enhance efforts to maintain an ongoing bench of individuals who indicate an interest in serving on the ACRS so that the selection process time can be reduced;

[58] ACRS, *Membership Balance Plan* at 3 (Dec. 2020), available at https://gsa-geo.my.salesforce.com/sfc/p/#t0000000Gyj0/a/t0000001F3Xu/eFQkKnXVMnaf6p.zZ1Jnw8LZkrKsl7KgUKlGAR_jGus. Of the current members, two members are serving their fourth term, and two other members are serving their third term. As such, 36% of the current members of the ACRS are serving on their third or greater term.

- engage with the Institute of Nuclear Power Operations (INPO)[59] to identify potential former Chief Nuclear Officer or senior nuclear executives who could serve on the ACRS;
- advertise through organizations like ANS, ASME, or IEEE, in order to improve awareness of open positions on the ACRS;
- recruit members with specific knowledge of upcoming issues (i.e., molten salt, fusion, and individuals with reactor design experience, etc.);
- use consultants to supplement the expertise of the ACRS in new and unique areas where the ACRS may not have sufficient expertise;
- use rotational assignees from Department of Energy National Laboratories, the U.S. Navy, or NASA; and
- have ACRS support staff include rotations of NRC Staff from the offices of Nuclear Material Safety and Safeguards or Nuclear Reactor Regulations or NRC Regional Field offices.

As already noted, according to interviews and the Membership Balance Plan, the ACRS may already have taken some of these steps, like maintaining a pool of applicants, using consultants, publishing solicitations in trade and professional society publications, and recruiting members in specific areas of expertise. However, interviewees indicated that it is still worth pursuing these steps to the maximum extent possible. The ACRS should also do what it can to improve the work-life balance of members in order to make the position more appealing to more applicants, for example, by following meeting schedules and not extending meetings well into the evening hours. Curtailing the field of ACRS reviews to those that add the most value (as recommended above) would also alleviate these burdens and improve time demands for members.

As a final note regarding the diversity of views, the majority of current members (six out of ten as of this publication) have a background at or ties to the Massachusetts Institute of Technology. The ACRS should obtain greater diversity by recruiting members from other universities, as there are a number of high caliber nuclear institutions that could serve as potential sources of new members.

2. Solution 2: Maintain a collegial atmosphere within the ACRS, improve relations with the NRC Staff, applicants and licensees.

a. The ACRS and the Commission

While former NRC Staff and licensees told numerous "war stories" about unprofessional and aggressive verbal behavior by some ACRS members—although a small number over the past 20+ years—the authors also received meaningful feedback that ACRS has generally improved in maintaining respectful and collegial relations with the NRC Staff and licensees over the last ten years.[60]

[59] INPO is an independent organization that was established in 1979 in response to the Three Mile Island accident. INPO has provided additional oversight to nuclear power plant facilities, as well as self-directed industry initiatives to improve operation, maintenance, and analysis of nuclear plants since the ACRS was formed.

[60] The authors heard repeated comments that in the past, some ACRS members, particularly those with an academic background, treated their ACRS role akin to the role they would play challenging PhD candidates during a

While the authors' research and discussions demonstrate that the demeanor of ACRS meetings has substantially improved since the late 1990s, interviewees shared several recent examples where ACRS members have engaged in unprofessional behavior such as reading newspapers during meetings, or disrespectful behavior toward their colleagues, applicants, or NRC staff.

In light of these comments, the authors would suggest that the Commission hire a consultant with expertise in organizational effectiveness to evaluate the manner in which ACRS members engage with the NRC Staff and licensees and suggest options for training and best practices on public and peer engagement. Further, it seems apparent that the evaluative process for selecting ACRS members should also incorporate components that screen for individuals who are independent yet collaborative and professional.

Finally, the ACRS should avoid circumstances where it defers solely to the expertise of one member. Instead, the ACRS should develop its opinions as a whole, and should be empowered by the ACRS Chair to push back when one member's entrenched views dominate a meeting.

b. The Executive Director for Operations ("EDO") in Coordination with the ACRS

As discussed previously, the EDO and the ACRS have a Memorandum of Understanding intended to establish a process to facilitate effective planning and engagement between the NRC Staff and the ACRS and the ACRS staff. The following matters should be addressed in that MOU and enforced:

- There should be a shared understanding between the ACRS and the EDO that NRC Staff preparations do not need to be at the level of a dissertation defense and that it is acceptable that the NRC Staff provide subsequent responses when they do not have an immediate answer to an ACRS member. The ACRS should not be permitted to criticize, badger, or undermine individuals who are unable to answer ACRS questions on the spot.

- There should be a shared understanding between the EDO and the ACRS that ACRS members should be able to set forth a brief explanation for why they are asking a question and tie it back to regulation (i.e., what is the member trying to understand and what is the safety concern). ACRS members should not ask questions only to "find" the presenter's "level of ignorance" or to pursue matters of personal interest unrelated to the objectives of the Commission or the ACRS regulatory mission. The ACRS Chairman should ensure that debate among ACRS members is constructive, collegial, and within the ambit of its statutory purpose.

- There should be a shared understanding between the EDO and the Chairman of the ACRS on how to ensure that the views of individual ACRS members do not unduly chill or influence the views of the NRC Staff. Rather, the EDO and the ACRS Chair should provide and maintain a safe space for respectful disagreement.

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dissertation defense, seeking to test the full range of knowledge of the individual (NRC Staff or licensee) before them. This would help to explain the reticence that some NRC Staff have in meeting with the ACRS and the resultant excessive and expensive (for the licensee) preparations for presentations before the Committee.

VI. Recommendation 3: Reduce the Cost of ACRS Reviews.

A. Problem Statement

Many interviewees, including several former ACRS members, stated a belief that the ACRS performs an important function by providing an independent review of the NRC Staff's work products. Some licensees and applicants, however, saw this as a problem as the licensee or applicant is responsible for paying for two reviews: one by the NRC Staff and a confirmatory review by the ACRS that covers the same content as the NRC Staff analysis. Further, it was observed that many NRC reviewers have become fairly adept at anticipating questions the ACRS might ask, and these issues are often already reflected in the ACRS presentation; yet the ACRS feels obligated to ask more questions. This issue is further compounded by what many saw as a tendency by the NRC Staff to excessively overprepare for ACRS meetings (sometimes with multiple dry runs per meeting), at least partially in response to the ACRS' sometimes unreasonable demands. The authors also heard complaints that ACRS members are sometimes unprepared for meetings and will go over or discuss information already spelled out in detail in the NRC Staff review documents. That said, as described above, the authors heard competing complaints from former ACRS members about the NRC Staff consistently providing documents behind schedule with minimal time for review prior to the meetings. The ACRS members should have the documents in sufficient time to prepare for scheduled meetings, but the ACRS Chairman also needs to hold individual members accountable for being fully prepared to avoid wasteful review of materials covered in work-product and background materials provided in advance by NRC Staff to the ACRS.

Recommendation 1 already addressed inefficiencies associated with NRC Staff and ACRS interactions, but the costs associated with ACRS reviews require independent recommendations.

B. Proposed Solution

1. Solution: Reallocate costs associated with ACRS reviews.

a. Congress

Congress should consider amending the Atomic Energy Act to provide that all costs associated with ACRS reviews, including the cost of ACRS time (which we understand is approximately $5-5.5 million[61]), be excluded from the fee recovery requirement. In addition, NRC Staff time used to prepare for ACRS meetings should not be billed to licensees and should also be excluded from fee recovery. This would help resolve some of the applicant and licensee concerns relating to the overall ACRS review process. It is of course still imperative that ACRS and NRC Staff make these reviews as efficient as possible in accordance with the other recommendations in this paper, as the public has an interest in fiscal responsibility as well as timely deployment of climate solutions.

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[61] *See* FACA Database Entry – ACRS.

VII. Recommendation 4: Adjust Management of the ACRS.

A. Problem Statement

Over the last several decades, Commissions have shown varying levels of interest in selecting the ACRS members and the Executive Director of the ACRS. Based on the authors interviews and observations, the ACRS and the Commission would benefit from more involved Commission attention to the evaluation and selection of ACRS members and the Executive Director of the ACRS. This would engage the Commission more deeply in the work product of the ACRS and enhance ACRS understanding of the value it provides to the Commission. Many interviewees commented on the oversight and management structure of the ACRS. Some interviewees opined that the Commissioners should exercise greater oversight of the ACRS. For example, some interviewees indicated that the Commission, as a whole, could better focus the topics of ACRS review, while other interviewees raised concerns about the undue influence and coercion of a prior NRC Chairman attempting to direct ACRS views without input from the rest of the Commission. Ultimately, the Commission, as a whole, should take a greater interest, and have more involvement, in the matters put before the ACRS. Of note, some interviewees indicated that they thought Commission meetings with the ACRS were merely a "dog and pony show" without sufficient material substance and depth.

Several interviewees indicated that the Executive Director of the ACRS staff is a critical role for the success of the institution as a bridge between the ACRS and the NRC Staff. The Executive Director should keep the ACRS informed of NRC Staff reactions and priorities and the "pulse" of the agency and industry. The authors also heard that the Executive Director is key in establishing a good ACRS staff.

A. Proposed Solutions

1. Solution 1: Increase Commission engagement with the ACRS.

a. The Commission

The authors suggest that the Commission increase its engagement with the ACRS including by being more proactive, directional, and cognizant of ACRS priorities to ensure review activities are appropriately focused. As an initial matter, the Commissioners need to take ACRS membership, the Chairmanship and the executive leadership of the ACRS staff more seriously. They should be more involved in the screening and selection of individual candidates who possess the knowledge, skills and abilities (KSAs) necessary to address key topics before the NRC. The Commission's involvement should start earlier in the process of ACRS member selection rather than waiting until an up-or-down vote on the NRC Staff recommendation. Also, the Commissioners should vote on each individual ACRS nominee individually rather than a slate of nominees. The Commission should also be more involved in selecting or identifying the Chair and should engage with the Chair on a regular basis. An ACRS Chair should have the ability to facilitate engagement with all stakeholders and impose discipline while reducing conflicts among individual ACRS members. The Chair should also be a strong leader capable of effective decision-making.

The Commission as a whole should also provide the ACRS information on topics requiring ACRS review, particularly those that are novel or have significant safety implications. If the ACRS were to utilize an action plan, like NUREG-0286 as suggested previously, the Commission should review, amend, and approve that document, while providing recommended areas for the ACRS to review, including areas to not review. Given the bow wave of licensing activities that the Commission will see from subsequent license renewals and potential new reactor orders, having a candid discussion about budgeting and prioritization is vital for both the Commission and the ACRS.

Also, the Commission should set much of the agenda for the semi-annual meetings with the ACRS, instead of allowing the ACRS to set the agenda. The Commission should also revamp the way that it interacts with the ACRS in meetings. As noted above, interviewees said that meetings between the ACRS and the Commission tend to be superficial and often consist of repeating talking points already provided in written materials. If that is the case, meetings could be eliminated in favor of paper filings.

2. Solution 2: Ensure the effectiveness of the ACRS staff.

a. The Commission

The Executive Director of the ACRS should be a firm, effective, and a credible leader. This individual must be able to interact with the NRC Staff, the ACRS staff, the Committee, and the EDO in an independent way to ensure the effectiveness of the ACRS. The position should always be given to a seasoned executive who has technical credibility and sufficient weight and standing within the Commission to push back against the NRC Executive Director of Operations as well as the ACRS Chairman and members, and who has the experience needed to garner respect, as well as the savvy needed to deal with various disparate personalities. The NRC should avoid using this position as a stepping-stone in the agency to "fast track" high potential managers in the NRC, to train SES executives, or to "park" SES executives who can or cannot be deployed elsewhere, potentially causing frequent changes in leadership.

The Commission should increase the budget of the ACRS supporting staff, as the ACRS staff is critical to the effective operation of the ACRS. The Commission should consider a budget review of the staffing needs to ensure the Executive Director's organization is appropriately staffed to meet the anticipated bow-wave of new reactor reviews. A larger budget would allow the Executive Director to recruit additional members for the ACRS staff and accommodate performance of new reactor reviews at a faster pace, in a high-quality manner.

VIII. Conclusion

This report has laid out NIA's recommendations for improving the effectiveness and efficiency of the ACRS to evolve from its original purpose (as envisioned in 1954 when commercial operating experience was scarce and uncertainty was high) and better align with the expectations Congress set out for the NRC under NEIMA. The authors' hope is that these recommendations (or some semblance thereof) will be implemented to position the ACRS and the NRC to successfully enable safe deployment of advanced nuclear energy to fight climate change and increase energy security. These recommendations are also in line with the ACRS' own suggestions for self-transformation presented to the Commission in 2019.[62]

The ACRS has the potential to play a valuable role in the NRC's licensing review process, but its purpose, processes and practices need to be improved, economized, and modernized. These recommendations can enable the ACRS and the NRC to implement numerous improvements to reduce the scope and length of ACRS reviews, increase focus on safety-relevant issues, gain greater efficiencies, optimize interactions with the NRC Staff, reduce licensee costs, enhance Committee membership and diversity, and improve the ACRS's utility through more proactive Commission investment in its composition and activities. The ACRS and the NRC Staff should work together to optimize the review process, control or reduce process costs for applicants and taxpayers, and cultivate more positive working relationships with each other and all stakeholders.

[62] *See e.g.*, Letter from Peter C. Riccardella, Chairman of NRC Advisory Committee on Reactor Safeguards to Kristine Svinicki, NRC Chairman (Oct. 17, 2019), available at https://www.nrc.gov/docs/ML1929/ML19290F956.pdf (hereinafter "ACRS Transformation Letter"); ACRS, *Commission Meeting with the Advisory Committee on Reactor Safeguard (ACRS)* (Dec. 6, 2019), available at https://www.nrc.gov/reading-rm/doc-collections/commission/slides/2019/20191206/staff-20191206.pdf.

February 19, 2021

Margaret M. Doane
Executive Director of Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Lessons-Learned from the Design Certification Review of the NuScale Power, LLC Small Modular Reactor

Dear Ms. Doane:

This letter presents lessons learned from the U.S. NRC review of the NuScale Power, LLC (NuScale) design certification application (DCA). These lessons learned are not comprehensive. Rather, they are limited to those that can be implemented in time, and offer substantial enhancement, to support submission and review of the next iteration of the NuScale small modular reactor (SMR) for standard design approval (SDA) in 2022.

The recommendations are:

1. Establish an appeal process to resolve disagreements between applicants and the NRC staff with respect to preliminary interpretations of requirements and guidance. The consequence of the absence of such a process is that regulatory burden has steadily increased from one applicant to the next without consideration of whether new staff positions have merit from a safety perspective.

2. Implement risk-informed decision making consistent with the Commission's direction in Staff Requirements Memorandum SRM-SECY-19-0036 to "apply risk-informed principles when strict, prescriptive application of deterministic criteria…is unnecessary to provide for reasonable assurance of adequate protection." NuScale did not observe application of the SRM in the DCA review outside the specific decision rendered by that SRM. The consequence of this limited application increased review durations and costs even where risk insights demonstrated adequate protection.

3. Define credible. Numerous NRC requirements incorporate the concept of credible. The consequence of a lack of a definition is unpredictability in the review as interpretations vary among the staff, resulting in increased review durations and costs even for events of incredibly low frequencies and often with insignificant radiological consequences. In a complementary effort, NRC should endorse IAEA standard SSR-2/1 as an acceptable method for evaluating the adequacy of defense in depth.

4. Rely on downstream requirements and programs to supplement design detail and test data as part of NRC safety findings. Downstream programs include programs required by regulation; e.g., Appendix B quality assurance, the ASME Code, and ITAAC. The

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consequence of not relying on these required programs is increased cost and time to develop applications and obtain approval, without improving the safety of the constructed facility. In the case of the NuScale DCA, application development and review costs exceeded half a billion dollars.

5. Clarify the role of the Advisory Committee on Reactor Safeguards (ACRS). The ACRS's approach during the NuScale DCA review worked because the NuScale SMR was the only advanced reactor design under review. However, it was unnecessarily broad and burdensome and the same approach may not work if there are multiple advanced reactor designs under review, as expected in the near future. The consequence of not clarifying the role of the ACRS is that the ACRS, due to resource constraints, may delay the approval and deployment of nuclear power plants with advanced safety features.

The enclosure to this letter provides more information and justification for these recommendations. It includes an appendix that provides NuScale's detailed perspective on Staff's proposal in SECY-21-0004 to exclude from issue resolution the design of NuScale's combustible gas monitoring equipment. This example illustrates the purpose and benefits of implementing these five recommendations.

The enclosure discusses NuScale's perspective as a light-water SMR applicant seeking design certification; the lessons-learned and recommendations are broadly applicable to other designs and licensing processes under the Part 50 and 52 regulatory framework. Discussions between NuScale and other advanced reactor developers and industry organizations indicates support for these recommendations. Assuming some of these entities express interest to the NRC, NuScale encourages the NRC to engage all stakeholders, industry and the public, to develop implementation approaches for these recommendations on a priority basis. In order to incorporate these recommendations into the forthcoming NuScale SDA application and review of that application, the recommendations should be in place by mid-2022.

I appreciate your consideration of these recommendations. If you have any questions, please contact me at 541.360.0740 or at tbergman@nuscalepower.com.

Sincerely,

Thomas Bergman
Vice President, Regulatory Affairs

Distribution: Scott Moore, Executive Director, Office of the ACRS
Annette L. Vietti-Cook, Secretary of the Commission
Chairman Christopher T. Hanson
Commissioner Jeff Baran
Commissioner Annie Caputo
Commissioner David A. Wright

Enclosure: Lessons-Learned from the Design Certification Review of the NuScale Small Modular Reactor, nonproprietary

Lessons-Learned from the Design Certification Review of the NuScale Small Modular Reactor

1.0 Purpose and Summary

This report identifies and discusses significant issues and challenges NuScale observed during the NRC's review of the NuScale design certification application (DCA), and identifies potential improvements in the review process to ameliorate these challenges for future applications.

Starting with the NuScale DCA, NRC has begun a period of reviewing novel reactor designs at what may prove an unprecedented pace, with several advanced light water and non-light-water reactor designs in various stages of active review, pre-application consideration, or under development for future submittal. Amongst those designs in pre-application is the next iteration of NuScale's light-water small modular reactor (SMR), planned for submittal for standard design approval (SDA) in 2022.

NuScale views the issues addressed herein as major challenges for efficient and timely review of a new reactor design, and those that are of utmost concern for the NRC's attention prior to submittal of the SDA application. This report discusses NuScale's perspective as a light-water SMR applicant seeking design certification. The lessons-learned are broadly applicable to other designs and licensing processes under NRC's Parts 50 and 52 frameworks. The recommendations are:

1. Establish an appeal process. Presently an applicant has no clear or established means to challenge Staff's preliminary conclusions on the adequacy of the design with respect to regulatory requirements. In order to promote regulatory clarity, certainty, and efficiency, a process and arbiter should be established to consider and decide significant disagreements in a timely manner.

2. Implement risk-informed decision making consistent with the Commission's Staff Requirements Memorandum SRM-SECY-19-0036. The Commission directed the Staff to "apply risk-informed principles when strict, prescriptive application of deterministic criteria…is unnecessary to provide for reasonable assurance of adequate protection." NuScale supports this position, but has not yet seen it applied outside the specific decision rendered by that SRM. The Commission's direction needs to be effectively implemented by the NRC Staff, ACRS, and, if established, appeal body in their respective deliberations.

3. Define credible. Numerous NRC requirements incorporate the concept of "credible." The lack of its definition results in inconsistent interpretation of which events must be evaluated by applicants and reviewed by the NRC Staff. NuScale was required to evaluate events with frequencies orders of magnitude less than the Commission's safety goals or the 10^{-6} per reactor year limit the Commission has stated as a threshold of credibility, and in some cases even where no core damage would result. NRC should define credible with respect to various aspects of the regulatory framework. In a complementary effort, NRC should also establish IAEA standard SSR-2/1 as an acceptable method for evaluating the adequacy of defense in depth.

4. Rely on downstream requirements and programs to supplement design detail and test data as part of NRC safety findings. The design review process as currently implemented requires a high degree of design finalization and an enormous amount of design information to be provided by the vendor. In many cases requirements and processes downstream of the design review, such as quality assurance, the ASME Code, and ITAAC, can form the basis for a safety conclusion in lieu of additional design detail and test data. This approach can provide reasonable assurance while reducing the cost of developing applications and maintaining design standardization.

5. Clarify the role of the ACRS. The ACRS review produced useful insights, but was unnecessarily broad and burdensome. In order to handle a potential influx of novel reactor design application in the near future, clarify the role of the ACRS to yield more effective and efficient reviews. This necessitates that the ACRS focus on matters of safety significance only, and rely on NRC staff interpretations of regulatory requirements and regulatory process issues.

2.0 Background

NuScale completed the first NRC review of an advanced reactor application, and overall the NuScale DCA review was a success. Staff completed review of the first small modular reactor design in 41 months following docketing of the application. The review was thorough; it involved over a quarter million review hours, about two million pages of documentation made available for review or audit, and about 100 gigabytes of test data. The ACRS conducted some 40 meetings[1] totaling approximately 440 hours.

While successful, the level of effort for reviewing the NuScale DCA may not be repeatable for future reviews. Significant resources were expended on issues with little bearing on the safety of the design, matters well beyond the purview of reasonable assurance of adequate protection. Several issues were left unresolved by Staff, which could have been avoided were the recommendations here in place. During the course of review, NuScale identified several overarching problems with the review process and review criteria that could yield significant efficiencies in the review of future applications, without impacting the effectiveness of NRC's review.

3.0 Discussion

3.1 Appeal process

3.1.1 NuScale's experience and problem statement

With an application review process as extensive and thorough as NRC's, it is expected and common that technical experts from the applicant and Staff will disagree on aspects of a design under review. Thus, as with each applicant before, Staff and NuScale grappled with numerous disagreements concerning the design, ranging from minor to significant.

Each disagreement is a source of delay in the application review, and thus a cost borne by the applicant. To some extent, of course, such issues are a normal part of the review process and to be expected and, in most instances, such disagreements were resolved through the ordinary course via an exchange of information in RAI responses and in public meetings. Too often, NuScale—like previous applicants—effectively has little choice but to concede Staff's position in order to keep the application moving forward toward completion of the Final Safety Evaluation Report (FSER), even where NuScale's technical experts view a change as unwarranted and of no safety benefit. For NuScale, the impact of conciliations has been significant, cumulatively and in some cases individually. At a minimum, each required a change in the licensing basis and underlying engineering work to support it. For some there was a material change to the design or future operations, with no justified safety benefit.

In some cases, though, because of a significant negative impact to NuScale's design, the cost of complying with Staff's position, or a risk of regulatory uncertainty, NuScale was unwilling to concede for expediency and reached an impasse with Staff. In the case of the inadvertent block actuation (IAB) device, NuScale sought to elevate the issue. Lacking an established process for resolving these issues,

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[1] Advisory Committee on Reactor Safeguards, *Report on the Safety Aspects of the NuScale Small Modular Reactor*, July 29, 2020, App. I.

the only option was to seek resolution from the Commission directly.[2] Even though the Commission was responsive, that resolution was an inefficient use of resources and would be inappropriate for questions where policy issues do not exist. The case of reactivity control and General Design Criterion (GDC) 27, discussed below, is another example where NuScale sought to elevate the issue but lacked an established way to do so.

NuScale's novel SMR technology differs substantially from the designs reflected in NRC's current regulations and guidance. NuScale observed two categories where opposing views arose often: (1) disagreement as to the meaning of a requirement or Staff position, and (2) cases where the design or application departs from an established requirement or position. The former involves a determination of whether the design satisfies or conforms to the regulation or guidance. The latter whether the proposed alternative is adequate to meet the intent of the requirement or position.

The appendix to this report details one of these significant disagreements with Staff: whether potential leakage from combustible gas monitoring following a beyond-design-basis severe accident must be accounted for in evaluating offsite and onsite dose consequences. That example illustrates NuScale's difficulties with the status quo. Staff's position on the issue was fluid and not clearly documented and communicated to NuScale. In fact, other than a partial preview in a December 2019 ACRS meeting,[3] NuScale first learned of Staff's final regulatory basis and position in the Final Safety Evaluation Report and the design certification proposed rule package.[4] As the appendix details, NuScale has fundamental disagreements with Staff on their position because it is a departure from guidance, misconstrues the underlying regulations, and is unnecessary to reach a conclusion of reasonable assurance of adequate protection. However, NuScale lacks a process to resolve this disagreement in a timely and efficient manner. As a result, the Staff's proposed rule for design certification[5] of NuScale's design includes a "carve out"[6] that is unnecessary and inappropriate, but was the only viable option to bring the Staff's review to timely completion.

Combustible gas monitoring is just one example of a substantial disagreement with Staff that resulted in unnecessary impacts to the review and licensing basis. Other examples included:

- Wall penetration radiation shielding. Staff insisted upon a level of detail for shielding penetrations that had not previously been required to reach a safety finding and that lacked a clear regulatory basis. NuScale's DCA, as with previous applicants, describes the general principles and performance characteristics for the shielding of penetrations through the reactor building shield wall, but leaves the specific shielding provisions for the final design stage. Although prior Staff reviews have reached safety findings using this approach,[7] Staff determined that it was inadequate for NuScale's DCA and insisted that if the penetration designs were not completed for DCA, that a Tier 1 interface requirement was necessary.[8] Prior DCDs did not include an interface requirement. In response to Staff, NuScale proposed a Tier 2 interface for penetration designs, an approach with far less regulatory burden. This approach is appropriate because shielding depends on final pipe routing, and penetration shielding methods are well understood. Thus,

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[2] NuScale Power, LLC Request for Commission Clarification on the Application of the Single Failure Criterion to "Active-Passive" Components, Letter LO-1218-63707, Dec. 14, 2018 (ML18351A145).
[3] Transcript, Advisory Committee on Reactor Safeguards Open Session, Dec. 4 2019, pp. 124–125 (ML20029E958).
[4] SECY-21-0004, *Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)*, Enclosure 1, *Proposed Rule*, Jan. 14, 2021.
[5] *Id.*
[6] A "carve out" colloquially refers to excluding an aspect of the design from issue resolution within the rule certifying the NuScale design.
[7] See, e.g., AP1000 DCD Tier 2 § 12.3.1.1.2, and SER § 12.4.1.
[8] *NRC Staff Views on the Bioshield Wall Major Penetrations Shielding, and the Reactor Building Nuclear Power Module Ventilation System Fire Damper Controls, or Ventilation Duct Radiation Source Term Description, for Discussion at July 22, 2019, Public Phone Call*, ML19204A277.

Staff's imposition would increase regulatory burden associated with Tier 1 without a safety benefit. NuScale was unwilling to agree to this increase in regulatory burden, and thus a carve-out was included in the Staff's proposed DC rule on this issue.[9] If presented an opportunity to appeal Staff's position, NuScale would have argued that the information presented in the DCA was sufficient with respect to regulatory requirements and agency positions on the scope and level of detail necessary for design certification, as supported by precedent.

- General Design Criterion 27. As addressed by SECY-18-0099,[10] NuScale disagreed with Staff on the proper interpretation of GDC 27's requirements for reactivity control under postulated accident conditions. Staff's interpretation required NuScale to request and receive an exemption from GDC 27 that NuScale believes was unnecessary. Lacking a process to formally appeal Staff's determination, NuScale could only submit a letter requesting that NuScale's position be included in Staff's informational SECY paper, completed two years later, on their planned approach for resolving the issue.[11] While this issue was ultimately resolved satisfactorily in accordance with Staff's position, if presented an opportunity to appeal, NuScale would have argued that NuScale's design complied with GDC 27.

The problem in these instances, and others of lesser visibility, is that a process is not available to gain a clear understanding of Staff's position and the basis for it, or to appeal that position to a separate decision maker. Therefore, an applicant is forced to concede to Staff's position in order to maintain schedule and manage review costs, or to informally engage management in an effort to reach consensus. In order to promote regulatory clarity, certainty, and efficiency, a process and arbiter should be established to consider and decide significant disagreements in a timely manner.

3.1.2 Proposal

An appeal process should be developed and implemented for new license applications. Specifically, where an applicant believes that Staff are misconstruing regulatory requirements, changing position on meeting requirements, or erroneously concluding the design fails to meet acceptance criteria, an applicant and Staff should be able to present their positions to a neutral arbiter to render a decision.

Important features of this process include:

- That the decision maker be neutral and well informed of both perspectives,
- That the process is carried out and the decision rendered transparently and efficiently, and
- That the decision be binding to the extent feasible on the Staff and ACRS.

Several models of such a process are in use or envisioned, with varying degrees of formality. The Canadian Nuclear Safety Commission, for example, includes in its Project Execution Plan for Vendor Design Review a protocol for escalating issue resolution through successive layers of agency and applicant management with explicit time frames for each step in the process.[12] Such an approach would be an improvement over the status quo, but not fully respond to the problem. Others have proposed that

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[9] SECY-21-0004, *Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)*, Enclosure 1, *Proposed Rule*, Jan. 14, 2021.
[10] SECY-18-0099, *NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability,"* Oct. 19, 2018.
[11] NuScale Power, LLC Submittal of White Paper Entitled "NuScale Reactivity Control Regulatory Compliance and Safety," Revision 0 (NRC Project No. 0769), Letter LO-1116-51829, Nov. 2, 2016, (SECY-18-0099, Enclosure 1).
[12] E.g., *Project Execution Plan NuScale Power, LLC Combined Phase 1 & 2 Pre-Licensing Vendor Design Review*, Rev. 0, July 19, 2019 (e-Doc-5724958v2) (not publicly available; NuScale can make available to NRC upon request).

procedures be established to formally resolve such disputes through the ASLB.[13] While that approach yields maximum finality, the regulatory changes to implement it do not support near-term advanced reactor applications, and it may be too cumbersome for all but the most significant disagreements. A model similar to the NRC's Committee to Review Generic Requirements (CRGR) also presents several features of an effective and efficient process for this purpose, but a formalized opportunity for affected-stakeholder involvement is needed in this case.

Accordingly, NuScale suggests an intermediate approach, inspired by the backfit appeal process. Indeed, recent changes to Management Directive (MD) and Handbook 8.4 note that while a new application is not strictly protected from backfit, "applicants should be anticipated to reasonably rely upon [the SRP] in development of their applications."[14] Therefore, the Commission states in changes to Directive Handbook 8.4 that "any change in requirements or regulatory staff positions from that version of the SRP interpreting the Commission's requirements should follow the same reasoned decision-making process as a forward fit." This direction, if fully implemented, would be a useful tool in identifying and resolving some disputes with Staff. However, an appeal process covering a broader scope of issues should be established separately from the backfit process. This would allow a process and timeline tailored to the purposes of a new reactor application, where an expeditious review is essential.

Similar to a backfit appeal, Staff would identify a change in position or an applicant would raise a Staff determination for appeal. The Staff would be required to clearly and succinctly state, in writing, their interpretation or position and the specific regulatory bases for it, and the applicant would be afforded an opportunity to see and respond to Staff's statement. A public meeting would provide an opportunity for the decision maker to hear from both sides and inform their decision, and they could reach out to other resources (e.g., legal counsel) as needed. Finally, the decision maker would render a determination in writing.

That appeal decision would be presumptively binding on the Staff. Because the new application is not protected by backfit regulations, Staff are not strictly prohibited from changing a position. But if the Staff seek to override the decision, they would be required to inform the Commission via vote paper of their intention to do so. Likewise, the ACRS would need to abide by the decision or state their reason for disagreeing with it. The appeal decision would thusly inform the Safety Evaluation and ACRS reports, but would not be binding on the Commission's responsibility to reach the ultimate safety findings.[15]

While this process needs to have an established structure, it need not be unduly formalized because it is not tied to a regulatory imperative (like the backfit rule) and not binding on the Commission's findings. Completing this process in a 60-day timeframe supports fast-moving application reviews. The effort and time involved dissuades an applicant from appealing trivial disagreements, bringing only consequential matters for appeal.

3.2 Risk-informed decision making

3.2.1 NuScale's experience and problem statement

The NuScale design presents an exceptionally low risk to public safety as demonstrated by both deterministic analyses and the probabilistic risk assessment (PRA): the mean value of the core damage

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[13] M. Segarnick and S. Desai, *Preparing for Advanced Reactors: Exploring Regulatory and Licensing Reform*, NUCLEAR LAW, Nov. 14, 2018.
[14] Staff Requirements Memorandum, *Staff Requirements — SECY-18-0049 — Management Directive and Handbook 8.4, "Management of Backfitting, Issue Finality, and Information Collection*," *Enclosure 2 - Edits to Directive Handbook 8.4*, May 29, 2019.
ML19149A308 - SRM-18-0049: Enclosure 1 - Edit to Management Directive 8.4 (16 page(s), 5/29/2019)
[15] In the case of an SDA, where the Commission's final safety findings only occur in issuing a license that references the SDA, the same principle would apply.

frequency of a NuScale Power Module achieves a core damage frequency several orders of magnitude smaller than the Commission's safety goal subsidiary objective.[16] This lower frequency is coupled to a design with much lower consequences: a source term less than 5 percent that of a large light water reactor. The PRA results are backed by deterministic safety analyses that show fuel damage is prevented for design basis accidents, without reliance on operator actions, electric power, or makeup water.

In the Advanced Reactor Policy Statement, the Commission established that advanced reactors were to provide "at least the same degree of protection of public health and the environment" as the then-current LWRs of 1994.[17] Together with a general expectation for enhanced safety margins and/or simplified or passive safety features, the Commission identified several attributes that could "assist in establishing the licensability" of a proposed design. Among them:

- Highly reliable and less complex shutdown and decay heat removal systems.
- Longer event progressions,
- Simplified safety systems that reduce required operator actions and equipment subjected to severe environments,
- A minimized potential for severe accidents,
- Reduced challenges to safety systems caused by balance of plant SSCs.

Finally, the Commission presented an incentive for exceeding the safety of current LWRs: "Some or all of the above attributes may help obtain <u>early licensing approval with minimum regulatory burden</u>."[18]

The NuScale design satisfies each of the above criteria and others. NuScale engaged in an extensive pre-application process as envisioned by the policy statement. The Staff completed the FSER more quickly than for prior design certifications, yet the regulatory burden of the review was excessive.

A common theme of many contentious issues during the review is that the interrelated design features and the overarching safety profile do not bear on Staff's probing of a particular aspect of the design or an individual acceptance criterion. A holistic review of the design is necessary. For example, the Staff's position that the IAB must be assumed as a potential single failure was due to its role in meeting anticipated operation occurrence (AOO) acceptance criteria when conservatively analyzing deterministic event sequences in Chapter 15. Those acceptance criteria and the safety analyses to demonstrate them are themselves a reflection of the large LWR designs for which the GDCs were created. Staff discounted that the underlying critical safety functions—adequate core cooling to prevent core damage and containment of fission products—would be assured even in the very low likelihood of the postulated events and single failure actually occurring. Those aspects of the design were envisioned by the Advanced Reactor Policy Statement to reduce regulatory burden, but did not prevent a protracted disagreement.

As the Commission concluded in SRM-SECY-19-0036,[19] risk information is an important tool to help resolve such contentious issues. The value of risk information in such cases is that it provides a quantified representation of a new design's safety performance with respect to specific issues that Staff historically consider deterministically, where Staff might otherwise apply acceptance criteria and positions developed for large LWRs with different safety profiles. Risk information effectively quantifies and distills

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[16] The NPM mean value internal events CDF is 3.0×10^{-10} per module critical year, as compared to the CDF goal of 1.0×10^{-4} per reactor year.
[17] *Regulation of Advanced Nuclear Power Plants; Statement of Policy*, 59 Fed. Reg. 35,461, July 12, 1994.
[18] *Id.* At 35,462 (emphasis added).
[19] Staff Requirements Memorandum, *Staff Requirements – SECY-19-0036 – Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves*, July 2, 2019.

into cognizable form the attributes of advanced reactors that were expected by the Advanced Reactor Policy Statement to expedite review and minimize regulatory burden.

Hence, NuScale supports the Commission's direction in SRM-SECY-19-0036, that the "staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria such as the single failure criterion is unnecessary to provide for reasonable assurance of adequate protection of public health and safety." In some cases, Staff successfully did so, such as the GDC 27 example where, although NuScale disagreed with the legal basis for Staff's position, Staff ultimately considered the likelihood and consequences of a return to power in approving NuScale's exemption request. Nevertheless, that result (which predates the IAB decision) came after a lengthy back-and-forth and assertions that the NuScale design should be modified to conform to Staff's interpretation of the deterministic GDC 27 criterion.

In other cases, NuScale proffered risk information to justify features of the design that was not duly considered by Staff in resolving those issues. For example, the surrogate "core damage event" postulated for the purposes of analyzing control room and offsite dose consequences has a probability of less than $3x10^{-10}$ per year and calculated consequences a fraction of the acceptance criteria. [20] Yet, as detailed in the appendix to this report, Staff concluded that it is necessary to calculate the dose contribution from potential leakage from combustible gas monitoring following such an event. This view represents a strict, prescriptive application of deterministic criteria, where a risk-informed view of the issue may have yielded resolution of the combustible gas monitoring system design.[21] Additional examples are cited in the discussion on "credible," below.

New designs such as NuScale's substantially differ from the designs on which the existing regulatory framework was built, in ways that fundamentally alter the presumed safety profile of the plant, such that the existing framework of requirements and guidance identifies apparent gaps in the design with little safety significance when considered holistically. Until a risk-informed, technology-neutral framework is completed and proven, use of risk information in individual licensing decisions is an important tool to close such gaps. Despite Commission's direction in SRM-SECY-19-0036, though, NuScale has seen no indication that risk-informed principles are being applied by Staff when strict, prescriptive application of deterministic criteria is unnecessary to provide for reasonable assurance of adequate protection.

3.2.2 Proposal

Application of risk-informed principles needs to occur early and proactively. NRC Staff have been reluctant to "mix" Chapter 19 with other parts of the application review. In part this stems from the regulatory requirements and guidance that fail to incorporate or point to risk insights in satisfying acceptance criteria in other chapters of the FSAR. In the short term, this can be addressed by reinforcing in top-level guidance, such as SRP Chapter 1, the Commission's direction in the SRM-SECY-19-0036, including explanation and examples of how that position should be implemented. Eventually, direction that is more specific is necessary.

Additionally, Staff's reluctance to incorporate risk insights stemmed in part from perceptions that NuScale's PRA was not developed for that purpose and that NuScale was advancing a "risk-based" rather than "risk-informed" approach. NRC guidance on PRA acceptability for risk-informed applications before fuel loading is incomplete and ambiguous. Despite the extensive resources spent by NuScale to establish the quality of the NuScale PRA, the lack of guidance allowed NRC Staff to dismiss risk justifications that NuScale offered due to uncertain deficiencies in NuScale's PRA. NRC should clarify the guidance on utilizing risk insights for a licensing application during the pre-fuel-load phases of a plant life-

[20] The total core damage frequency for internal-events at full-power is 3E-10 per module critical-year, and the several intact containment events comprising the surrogate CDE are a subset of those. *See also* Appendix Section 2.3.1.
[21] As discussed in the appendix, NuScale believes Staff's position is not supported by the regulatory requirements. Assuming it is, however, a risk-informed review could have allowed resolution of the issue.

cycle, which is essential for new designs.[22] In so doing, the criteria for PRA acceptability should reflect the state of a new design and the broader safety review conducted by NRC to approve a new licensing basis; existing guidance is focused on changes to a previously-approved licensing basis previously found to provide reasonable assurance of adequate protection under a deterministic framework. Use of risk insights to inform Staff's position on whether strict, prescriptive application of deterministic criteria during an initial licensing review is a function that can and should be readily compatible with a high-quality design-application PRA such as NuScale's.

Finally, Staff procedures and training are necessary to ensure that risk insights are incorporated into Staff's review early and effectively. While NuScale will continue to point to risk insights in the SDA FSAR and in responding to RAIs, Staff should be empowered to look to those insights on their own initiative, thereby preventing some RAIs from being issued in the first place and reducing the burden of review. At some point, the very low risk posed by the NuScale design (as exemplified by the very low core damage frequency and large release frequency) is the risk-insight that has meaning. Given a low enough risk level, public health and safety are adequately protected and the NRC mission is satisfied.

3.3 Define "Credible"

3.3.1 NuScale's experience and problem statement

The notion of whether an occurrence is "credible" is the basis for, a pervasive feature of, and an inextricable aspect of NRC's regulatory framework. Whether an event is credible or not informs the scope of numerous requirements and Staff positions, either explicitly or implicitly. Yet the meaning of "credible" in any of its various contexts continues to elude definition, impeding regulatory certainty and review predictability.

NuScale encountered this issue in several aspects of the DCA review, and it continues to challenge the review of NuScale's Emergency Planning Zone methodology. For example:

- NuScale's design precludes large-break LOCAs by eliminating large bore piping. NRC's rules and precedent clearly limit LOCAs to piping ruptures,[23] based on a rupture of the reactor vessel and its appurtenances being incredible due to the quality of design, fabrication, and inspection.[24] Where NuScale initially demonstrated the acceptability of the design with respect to any pipe rupture within the reactor coolant pressure boundary, consistent with 10 CFR 50.46, NRC Staff determined that it was necessary for NuScale to additionally evaluate ruptures of valve nozzles as a design-basis LOCA. This new Staff position was a deterministic extension of existing rules, without regard to the probability of such a rupture actually occurring.

- With respect to reactivity control and GDC 27, NuScale argued that in addition to the regulatory basis for NuScale's interpretation, it is also incredible that an initiating event would yield a return to power.[25] Yet NuScale was unable to reach agreement with Staff on that approach.

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[22] Regulatory Guide 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*, relies on an "expanded role for PRA" beyond its uses in existing facilities or completed design reviews such as NuScale's. No guidance on the acceptability of such a PRA for that purpose is provided.
[23] 10 CFR 50.46(c)(1) (defining LOCAs as "breaks in pipes in the reactor coolant pressure boundary").
[24] *In the Matter of Consolidated Edison Company of New York (Indian Point Unit No. 2)*, 5 A.E.C. 20 (1972). Staff stated their position that "the probability of vessel failure has been found to be 'so low' as not to require consideration of the consequences of such failure in the assumptions employed in determining site suitability."
[25] Although GDC 27 is phrased deterministically, the limits of its scope could be defined by the credibility of the events to be addressed under it. By comparison, GDC 35, which does not explicitly limit LOCAs to piping, was already deemed to exclude RPV rupture prior to 10 CFR 50.46.

- NuScale's position on the "maximum hypothetical accident" that serves as the design basis source term was informed and justified by the credibility of a core damage event occurring for the NuScale design.[26] NuScale argued that such an occurrence was orders of magnitude less likely than a comparable event considered as the design basis source term for large LWRs, and thus not similarly "credible." Staff were unwilling to rely on a probabilistic credibility threshold, which led to an event with less than $3x10^{-10}$ per year likelihood serving as a source term for design basis dose consequence analysis, siting, and EPZ considerations.

- With respect to NuScale's PRA, NuScale responded to several RAIs concerning potential severe accidents of doubtful credibility. In eRAI 8882,[27] Staff questioned the radiological consequences of a beyond-design-basis module drop event with a calculated mean CDF of $8.8x10^{-8}$ per calendar year.[28] In eRAI 8889,[29] Staff questioned NuScale's calculated probability for an induced steam generator tube failure (SGTF) during a core damage event. However, even if NuScale were to conservatively assume SGTF in response to every core damage event, the resulting event frequency would be the same as the internal events CDF of $3x10^{-10}$ per year.

- NuScale continues to engage Staff on review of the EPZ sizing methodology. Regulatory history, precedent, and the proposed EP Rulemaking all explicitly reference "credible" as the scope of events to be considered in evaluating the need for advance emergency planning. However, NuScale's effort to define credibility in probabilistic terms has been opposed in both substance and principle: Staff will not agree with NuScale's specific threshold and have resisted the definition of any specific numeric criterion. Staff have stated that the approval of the Clinch River ESP, with numeric thresholds included in its EPZ sizing methodology, did not constitute approval of those thresholds and have further indicated the those thresholds have no broader applicability.[30]

Other instances abound. In the safety goal policy statement[31] the baseline level of acceptable nuclear risk is established by the Quantitative Health Objectives (QHOs), where it is stated that the operation of a nuclear power plant should not increase the risk to the public by more than 0.1% of the sum of all existing risk. The implication of a nuclear power plant satisfying the QHOs is that the public is adequately protected. Using fatality information from the U.S. Center for Disease Control (CDC) and <u>bounding</u> assumptions on the consequences of a nuclear power plant core damage accident, the QHOs conservatively translate to an equivalent core damage frequency (CDF) of $5x10^{-7}$ core damage accidents per year. It is in this context that NuScale experienced scrutiny of its DCA with respect to potential plant upset conditions that were shown to be below the $5x10^{-7}$ frequency necessary to adequately protect the public health and safety. When NuScale staff questioned the value of engaging on these issues, the NRC Staff response was that clarity on these very low likelihood event sequences was necessary to ensure confidence on the risk insights to be obtained from the DCA PRA. To this end, frequency thresholds below which further scrutiny is unwarranted should be established. As an example, in the context of the Licensing Modernization Project[32] that threshold is $5x10^{-7}$ per year for beyond design basis events.

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[26] *NuScale Power, LLC Submittal of WP-0318-58980, "Accident Source Terms Regulatory Framework,"* Letter LO-0518-59973, May 16, 2018 (ML18136A850).
[27] *NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 63 (eRAI No. 8882) on the NuScale Design Certification Application*, Letter RAIO-0618-60459, June 14, 2018 (ML18165A438).
[28] NuScale FSAR Section 19.1.6.2.
[29] *NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 71 (eRAI No. 8889) on the NuScale Design Certification Application*, Letter RAIO-0118-58186, Jan. 15, 2018 (ML18016A240).
[30] *See* U.S. Nuclear Regulatory Commission, *Summary of the January 12, 2021, U.S. Nuclear Regulatory Commission Category 1 Public Teleconference with NuScale Power, LLC, to Discuss Staff Comments on NuScale Power LLC's Topical Report, TR-0915-17772, Revision 2* (ADAMS accession number pending).
[31] *Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication*, 51 Fed. Reg. 28,044, Aug. 21, 1986.
[32] Regulatory Guide 1.232 and NEI 18-04.

Similar thresholds should be clearly defined and available to LWRs, especially given the vastly greater relevant operating experience base.

Within NRC, the current view of “credible” seems to be that Staff know it when they see it. For example, Staff concluded that the estimated frequencies of ex-vessel and late in-vessel core damage accidents are low enough for evolutionary and passive LWRs that the associated source terms are not considered credible for the purposes of evaluating offsite dose limits.[33] Other U.S. agencies have defined credible or an analogous limit on event occurrence frequency. The National Aeronautics and Space Administration (NASA) defines[34] a “credible failure mode” of concern for spacecraft explosion as a failure probability exceeding $1x10^{-6}$. The Federal Aviation Administration defines an “extremely improbable” occurrence, which is the allowable frequency of a potentially catastrophic failure condition, as one having a likelihood of less than about $2x10^{-6}$ per year.[35]

The lack of a definition, or even clear explanation, of the meaning of credible frustrates applicants and introduces significant regulatory uncertainty. Despite a history of attempts,[38] challenges, and reluctance to define “credible,” a clear and consistent NRC definition should be achievable and is appropriate given the state of PRA technology and the tools available to compensate for it.

3.3.2 Proposal

The NRC, with industry and public involvement, should develop probabilistic criteria defining credible. Possible mechanisms to implement the definition would be through guidance, a regulatory interpretation, or commission action.

Credible may have different meanings within different contexts. For instance, a design basis internal initiating event is a “credible” one, often but unofficially considered one with a likelihood of more than about $1x10^{-4}$ of occurring.[39] Meanwhile, the Commission recognized in the Statements of Consideration for a Part 100 rulemaking that, with respect to event sequences for source term and siting considerations, “events having the very low likelihood of about 10^{-6} per reactor year or lower have been regarded in past licensing actions to be ‘incredible’, and as such, have not been required to be incorporated into the design basis of the plant.”[40] A criterion for the credibility of events to be considered in sizing the emergency planning zone has not yet been determined, but the Clinch River precedent indicates $1x10^{-6}$ is appropriate.[41]

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[33] SECY-34-302, *Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs*, Dec. 19, 1994, p. 6.
[34] NASA Technical Standard NASA-STD-8719.14B, *Process for Limiting Orbital Debris*, April 25, 2019.
[35] An extremely improbable failure conditions is defined as one having a probability on the order of $1x10^{-9}$ or less per flight-hour. FAA Advisory Circular 25.1309-1A, *System Design and Analysis*, June 21, 1988. Conservatively assuming 2,000 flight hours per year yields a probability of $2x10^{-6}$ per flight year. Data from one source indicates that in 2019, U.S carriers utilized their large narrowbody aircrafts an average of 3,082 hours per year, which would yield a higher threshold for extremely improbable.” MIT Airline Data Project, http://web.mit.edu/airlinedata/www/Aircraft&Related.html.
[38] *See*, *e.g.*, Memorandum from James M. Taylor, Executive Director for Operations, U.S. NRC, *Response to SRM for SECY-91-229, "Severe Accident Mitigation Design Alternatives For Certified Standard Designs,"* Jan. 28, 1982, p.2 (“The staff is working on its final definition of what is a "credible" severe accident for possible rulemaking.”).
[39] See, e.g., D. Powers, Advisory Committee on Reactor Safeguards, *Accident Analysis Design Basis and Beyond Design Basis* (ML090060320).
[40] *Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants*, 61 Fed. Reg. 65,157, Dec. 11, 1996.
[41] The Clinch River ESP Site Safety Analysis Report proposes that events with frequency greater than $1x10^{-6}$ per reactor-year be compared against the Protective Action Guidelines. CITE. The EP proposed rulemaking, consistent with precedent, suggests that “credible” sequences are to be compared against the PAGs. CITE.

Those criteria may be intentionally distinct, as they characterize and implement different aspects of the defense-in-depth (DID) regime, but they should be cohesive. In contrast, the current criteria for design basis external initiating events lack consistency either amongst the various hazards or with design basis internal events. For example, a design basis tornado includes one with 1×10^{-7} probability of occurrence, while a design basis earthquake is not defined probabilistically but does not approach that infrequency—nor should it.[42] Staff should determine comprehensively what is or is not credible to consider as any type of design-basis initiating event, design basis event sequence, or beyond design basis event sequence.

NuScale recognizes the state and limits of PRA. NuScale also recognizes Commission's policy on use of PRA and risk information, which preclude exclusive reliance on probabilistic criteria in justifying the licensing basis. Nevertheless, it is acceptable and consistent with those policies to define credibility thresholds in probabilistic terms. First, the existing licensing framework is extensively based on qualitative judgments of likelihood:

> *The deterministic approach involves implied, but unquantified, elements of probability in the selection of the specific accidents to be analyzed as design basis events.*[43]

If the informed, qualitative judgement of Staff members was sufficient to establish the regulatory framework in the first instance, then probabilistic values backed by modern PRA techniques should have sufficient veracity to inform requirements going forward. The NRC has made similar observations:

> *Regulators using a deterministic approach simply tried to imagine "credible" mishaps and their consequences at a nuclear facility and then required the defense-in-depth approach—layers of redundant safety features—to guard against them.*
>
> *These "maximum credible accidents" were, in turn, used to define design-basis events, which were then used to determine the values of controlling design parameters for structures, systems and components (SSCs); the safety classification of SSCs; the contents of licensing-basis documents (such as final safety analysis reports (FSARs) and technical specifications); and needed supporting documents, such as plant procedures. The licensing efforts for early plants focused, therefore, on "design-basis events."*[44]

As this history exemplifies, probabilistic criteria are fully compatible with a risk-informed DID regulatory framework. The entirety of the NRC's regulatory framework, applied comprehensively, ensures DID. Moreover, NuScale believes that at some point a very low risk in and of itself implies sufficient DID. Individual layers of that DID can and should be defined using probabilistic criteria to ensure it serves its intended purpose of achieving adequate protection of the public health and safety and common defense and security. The quality of a PRA to implement the credibility criteria will also require consideration, but this should not prevent establishing the criteria in the first place. To reiterate, NRC should not require undue precision or excessive site-specific or post-construction details in a PRA to bring greater clarity to a licensing framework currently built on qualitative judgement.

In an effort complementary to clarifying the meaning of credible, NuScale recommends that NRC establish an acceptable method for evaluating DID. As the discussion above illustrates, credibility and DID are interrelated: event credibility helps to establish adequate DID, and DID ensures that the design, siting, and emergency planning provide robust protection commensurate with their relevance to events of

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[42] In the context of mixed oxide fuel fabrication facilities, Staff have at least adopted a single probability criterion for all external events—an incredible external event is one with a frequency of occurrence conservatively estimated as less than once in a million years (1x10-5). NUREG-1718, *Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility*, Aug. 2000, p. 5.0-22.
[43] SECY-98-144, *White Paper on Risk-Informed and Performance-Based Regulation*, June 22, 1998.
[44] COMSECY-14-0037, *Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards*, Nov. 1, 2014, Enclosure 1, *quoting* "A Short History of Nuclear Regulation, 1946–2009."

varying likelihood and severity. During NuScale's pre-application engagement, the DCA review, and the EPZ topical report review the Staff position regarding the evaluation of DID fluctuated. This resulted in costs with no benefits realized. The root cause is the NRC has no accepted method to evaluate DID, which, combined with turnover in the NRC staff evaluating NuScale's application, resulted Staff changing position.. NuScale recommends that the NRC endorse the IAEA method documented in IAEA SSR-2/1 to bring consistency and predictability to NRC reviews, which will also benefit vendors who plan to deploy their designs in other countries that use the IAEA guidance.

3.4 Downstream Requirements and Programs

3.4.1 NuScale's experience and problem statement

During the DCA review process, Staff raised design issues addressed by processes and programs "downstream" of the DCA-phase design, which could be appropriately resolved by relying on those downstream requirements to supplement design details and test information provided for DCA review.

The review of radiation protection (RP) included numerous such instances. In one case, Staff asked for a description of the airborne activity assessment for the Radioactive Waste Building with respect to potential alpha-emitting sources. In response, NuScale noted that the potential for airborne radionuclides is not different from currently licensed plants or approved designs, and the control of alpha-emitting radionuclides for the protection of workers will be part of the COL's RP Program.[45] Staff did not accept that as resolution even though a licensee's RP Program is required by 10 CFR 20.1101 to maintain doses within regulatory limits and as low as reasonably achievable.[46] The experience with penetration radiation shielding, discussed above, is another case. There, a generic description of shielding characteristics together with the licensee's and NRC's processes for construction, inspection, and verification of that shielding would have assured that the as-built penetrations are adequately shielded. The staff appeared to be under the assumption that the design is fixed at certification; in reality the shielding design and operational exposures will be modified throughout operation depending on actual radiological conditions in the plant.

The issue of density wave oscillation (DWO) within NuScale's steam generator (SG) tubes presents another example.[47] While DWO is an issue requiring attention post-DCA, it will be addressed by downstream processes that ensure the plant will be designed, constructed and operated safely. The DCA defines an acceptable IFR design as one that ensures ASME Code criteria are met with respect to flow instabilities; the development, validation, and implementation of a methodology for the evaluation of DWO would be addressed by a COL Item; and the adequacy of the design would be assured by completion of an ITAAC requiring that the requirements of ASME Code Section III are met. Each step of this process would be subject to rigorous QA requirements and NRC inspection. Staff initially concurred with NuScale's approach; the Advanced Final Safety Evaluation Report with No Open Items proposed to resolve DWO based on subsequent testing and inspection requirements.[48] However, in the FSER Staff

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[45] *NuScale Power, LLC Response to NRC Request for Additional Information No. 445 (eRAI No. 9255) on the NuScale Design Certification Application*, Letter RAIO-0518-59925, May 9, 2018 (ML18129A415).
[46] Rather than relying on the RP Program, Staff performed an independent assessment to determine that the potential airborne concentration of isotopes will not be significant. *See* NuScale FSER p. 12-19.
[47] See SECY-21-0004, Enclosure 1, Section III.C.3 for a summary of SG integrity with respect to potential DWO.
[48] See, e.g., Advanced Final SER with No Open Items, Section 3.9.2.4.3: "There is no analysis for the array of [steam generator inlet flow restrictors] that diffuses secondary coolant inlet flow and mitigates the possibility of DWO within the [helical coil steam generator] tubing. Instead, various flow restrictor designs were tested in a special fixture…. The final SGIFR design is based on a concept that showed no signs of LFI or any other significant FIV…. The NRC staff concluded that the applicant's LFI analysis methods and calculations are based on validated methods, there is significant estimated margin against LFI, all components evaluated for LFI will be tested during initial startup, and those components are part of the inservice inspection plan. Therefore, the NRC staff finds that there is reasonable assurance of no significant LFI induced vibration and structural damage for the life of an NPM."

concluded that SG integrity due to DWO could not be resolved, implicitly concluding that they would not rely on the downstream processes to resolve DWO at the DC stage.[49]

The design review process as currently implemented necessitates a high degree of design finalization and an enormous and ever-increasing amount of design information from the vendor, typically for a design that has not yet been completed or built. The degree of design finalization achievable or reasonable for a design approval applicant has practical limits, as recognized by SECY-90-377.[50] This is especially the case for advanced reactor designs—while NuScale developed a large test program and benefitted from a large operating experience base, future designs may not have access to such resources. Required downstream requirements and processes, necessary under NRC's existing regulations or from the DC itself, can provide an acceptable basis for Staff to conclude that there is reasonable assurance the design complies with NRC regulations.[51] Applied prudently, this approach will maintain an appropriate level of design completion at the design review stage consistent with the Commission's Standardization Policy,[52] while allowing detailed design to occur at a time compatible with the vendor's design process.

3.4.2 Proposal

Staff guidance, and potentially the NRC's regulations, should be revised to clarify Staff's ability to rely on downstream requirements, programs, and processes as part of their safety findings.

Although NuScale agrees with NEI's recent comment that 10 CFR 52.47 should be revised as part of the *Alignment of Licensing Processes and Lessons Learned From New Reactor Licensing* rulemaking[53] to explicitly allow reliance on design and programmatic controls,[54] the current regulations already contain such leeway. The DC requirements state that an application must provide a "level of design information sufficient to… reach a final conclusion on all safety questions associated with the design," and the Staff need reasonable assurance that the design complies with the Atomic Energy Act and NRC's regulations to issue the DC. Downstream requirements, processes, and programs that are required by regulation or by the DC can form the basis for that reasonable assurance, offsetting the amount of design information that might otherwise be necessary to reach a final safety conclusion.

In the case of Standard Design Approval, the regulations are less prescriptive, commensurate with the degree of finality and final issue resolution provided by that process. Thus, regardless of potential regulatory obstacles for this approach under the DC regulations, NuScale believes Staff may more liberally rely on downstream processes in completing their safety evaluation for an SDA application.

3.5 Role of the ACRS

3.5.1 NuScale's experience and problem statement

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[49] See, e.g., FSER section 3.9.2.4.4.1: "However, the staff cannot make a safety finding on the long-term integrity of the SGIFRs under possible reverse secondary coolant flow (either subcooled or two-phase) due to DWO instabilities. As discussed above in Section 3.9.2.4.3.11, a future COL applicant referencing the NuScale design will be responsible for providing an analysis of the potential for DWO secondary-side instabilities in the SGs for staff review and approval."
[50] SECY-90-377, *Requirements for Design Certification Under 10 CFR Part 52*, Nov. 8, 1990.
[51] *See* 10 CFR 52.54.
[52] *Nuclear Power Plant Standardization*, 52 Fed. Reg. 34,884, Sept. 15, 1987.
[53] *Incorporation of Lessons Learned from New Reactor Licensing Process (Parts 50 and 52 Licensing Process Alignment)*, Docket ID NRC-2009-0196.
[54] Nuclear Energy Institute, *Part 50/52 Lessons Learned Rulemaking – Addressing the term "Essentially Complete*," Sept. 24, 2020 (ML20268C271). NEI recommended that 10 CFR 52.47 be revised to expressly acknowledge, "Design and programmatic controls can substitute for design information where those controls provide reasonable assurance that the as-built design will meet NRC requirements."

The DCA review included a broad and intensive review by the ACRS that ultimately yielded a conclusion that the NuScale design provided enhanced safety margins, presented no undue risk, and should be approved by the Staff and certified by the NRC.[55] The ACRS review was successful and exhibited some innovation for enhancing the process in its use of crosscutting focus area reviews. NuScale observed other aspects that could be improved to achieve a more efficient and safety-focused review.

For example, during the DCA review the ACRS explored some issues that are already well-addressed by the regulatory framework and peripheral to the safety of the design; for example, the use of assumptions (open design items in NuScale terminology), and the flow down of requirements to subsequent processes such as combined licenses and test programs required by NRC. Similarly, absent some unexpected and novel approach to meeting NRC's quality assurance requirements, an ACRS report on the applicant's QA program is unnecessary to allow Staff to reach a conclusion of conformance with NRC requirements.

Beyond those peripheral process issues, NuScale also found that in the review of issues more directly bearing on safety, ACRS's role was abstruse. The applicant's role is clear: to ensure its application meets applicable standards and requirements and that the design meets, with reasonable assurance, applicable regulations and guidance, or justifies alternatives or exemptions from those standards where it differs. The Staff's role is likewise clear: to confirm that, with reasonable assurance, the applicant has fulfilled its responsibility as described above and thus the design presents no undue risk to public health and safety and is consistent with the common defense and security. If necessary due to novel features or approaches, additional standards are developed in the course of the review.

In contrast, the role of the ACRS appeared unconstrained in scope and purpose, or by the NRC's regulatory framework. This often put NuScale in the position of defending NRC requirements and that compliance with those requirements was adequate, which should not be the responsibility of an applicant. To NRC Staff's credit, they were willing to and encouraged NuScale to defer such inquiries to the NRC Staff. Even here, the Staff were not always persuasive. This results in excessive time and cost for applicants as matters of regulation and process are resolved.

Applicants should not be expected to address ACRS (or Staff) requests that fall outside the responsibility to demonstrate that the application meets NRC's standards.

3.5.2 Proposal

The role of the ACRS should be clarified to result in more effective and efficient reviews. This is necessary considering there may be multiple novel design reviews conducted concurrently in the near future. The approach taken for the NuScale DCA will not work; the statutory limitations on ACRS resources preclude that amount of review on multiple designs. If changes are not made the ACRS will constrain the deployment of advanced nuclear power. To succeed, the ACRS should focus on matters of safety significance only, and rely on NRC staff interpretations of regulatory requirements and regulatory process issues.

In their review of an application, the ACRS should explicitly address and evaluate established regulatory criteria in their letter reports. Where the ACRS questions the safety of the design, its concerns should be stated in terms of the specific criteria at issue, and how the design does not meet them. Where ACRS views an established standard as inadequate, the issue should be raised and considered by NRC generically, separate from completing the review of a particular application.[56] Only in such cases where established criteria are not met, or where they do not exist because of an exemption request or other alternative to current standards, would the ACRS consider *de novo* whether an aspect of the design reasonably assures adequate protection. Legal expertise on the regulatory framework and history, either

[55] Advisory Committee on Reactor Safeguards, *Report on the Safety Aspects of the NuScale Small Modular Reactor, July 29, 2020* (ML20211M386).
[56] Compare the hearing procedures, which absent special circumstances preclude attack of the Commission's regulations during an individual license proceeding. 10 CFR 2.335.

from within ACRS or from without, may be useful to facilitate such a review. Additionally, NuScale's above recommendations on risk-informed review and crediting downstream requirements should inform ACRS's conclusions.

With respect to the scope and mechanics of ACRS review, NuScale commends the cross-cutting issues approach used in the DCA review. To build upon that approach, an informational ACRS briefing should occur relatively early in the review, such as the ACRS visit to NuScale's office in July 2019 to tour the test facilities, simulator, and discuss items of mutual interest. Then the Commission should delegate to Staff to identify significant issues or cross-cutting issues that ACRS is requested to report upon. Such issues would exclude those with either minimal safety significance or where no novel question is presented, such as the examples cited above. Based on the informational briefing and subsequent substantive meetings, ACRS could propose additional issues for consideration, but these could be subject to EDO approval.

4.0 Conclusion

NuScale views the DCA review as a successful demonstration of the NRC's ability to review an advanced reactor design in a reasonable timeframe. However, the review process needs to and can continue to be made more efficient and predictable, reducing the substantial burden on applicants and the NRC while enhancing focus on matters important to safety. To that end, NuScale recommends that NRC:

1. Establish an appeal process to resolve disagreements between applicants and the NRC staff with respect to preliminary interpretations of requirements and guidance. The consequence of the absence of such a process is that regulatory burden has steadily increased from one applicant to the next without consideration of whether new staff positions have merit from a safety perspective.

2. Implement risk-informed decision making consistent with the Commission's direction in Staff Requirements Memorandum SRM-SECY-19-0036 to "apply risk-informed principles when strict, prescriptive application of deterministic criteria…is unnecessary to provide for reasonable assurance of adequate protection." NuScale did not observe application of the SRM in the DCA review outside the specific decision rendered by that SRM. The consequence of this limited application increased review durations and costs even where risk insights demonstrated adequate protection.

3. Define credible. Numerous NRC requirements incorporate the concept of credible. The consequence of a lack of a definition is unpredictability in the review as interpretations vary among the staff, resulting in increased review durations and costs even for events of incredibly low frequencies and often with insignificant radiological consequences. In a complementary effort, NRC should endorse IAEA standard SSR-2/1 as an acceptable method for evaluating the adequacy of defense in depth.

4. Rely on downstream requirements and programs to supplement of design detail and test data as part of NRC safety findings. Downstream programs include programs required by regulation; e.g., Appendix B quality assurance, the ASME Code, and ITAAC. The consequence of not relying on these required programs is increased cost and time to develop applications and obtain approval, without improving the safety of the constructed facility.

5. Clarify the role of the ACRS. The ACRS's approach during the NuScale DCA review worked because the NuScale SMR was the only advanced reactor design under review. However, it was unnecessarily broad and burdensome and the same approach will not work if there are multiple advanced reactor designs under review, as expected in the near future. The consequence of not clarifying the role of the ACRS is that the ACRS, due to resource constraints, may delay the approval and deployment of nuclear power plants with advanced safety features.

These recommendations can be implemented relatively quickly in order to prepare the NRC for forthcoming reviews of numerous advanced reactor designs, including NuScale's next application. NuScale looks forward to further discussions with NRC and other interested stakeholders on these topics.

Appendix: Leakage from Combustible Gas Monitoring

1.0 Background

In accordance with NRC requirements, the NuScale design includes the capability for monitoring the concentration of hydrogen and oxygen in the containment atmosphere following a significant beyond design-basis accident. The principal design and function of that combustible gas monitoring capability (CGMC) has not changed since initial DCA submittal. The *Combustible Gas Control* technical report describes the CGMC.[57] During review of a revision to the Accident Source Term Topical Report,[58] NRC first raised the issue of leakage from combustible gas monitoring (CGM).

The CGMC is distinct from monitoring and sampling functions that support normal operations; CGMC is provided because it is prescribed by 10 CFR 50.44 to support beyond-design-basis accident management. It does not serve a purpose in the NuScale design that is unique from other light water reactors. In fact, the NuScale design does not require inerting or hydrogen reduction to ensure containment integrity for at least 72 hours. As compared to large LWRs, CGMC in the NuScale only supports potential long-term emergency actions for an extremely unlikely severe accident scenario.

Staff concluded that they were unable to make a safety finding on the design of the CGMC to minimize and control leakage outside containment, which could impact the offsite dose analyses, the dose analyses for the main control room, and the dose to an operator re-isolating CGM if necessary.[59] NRC's complete regulatory basis and position on the issue was first stated in the NuScale Final Safety Evaluation Report. Staff's evaluation raises numerous interrelated regulations without a clear explanation of how each relates to the issue or how they together justify NRC's position. Based on the discussion provided and informed by prior interactions, the crux of Staff's position seems to be that because the CGMC equipment is required by regulation, the potential doses resulting from using that function during a severe accident must be accounted for in onsite and offsite dose calculations. Staff have stated that their concern arose because NuScale's CGMC design is unique in transporting containment atmosphere outside containment, thus giving rise to a new leakage path not heretofore considered in prior reviews. As discussed below, both Staff's concern and the justification for their ultimate position are novel and unfounded.

2.0 Discussion

2.1 Regulatory Framework

10 CFR 50.44(c)(4) requires a water-cooled reactor design with an inerted containment[60] to include equipment for monitoring oxygen and hydrogen "following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning." In 2003 that

[57] NuScale Letter LO-0319-64935, *NuScale Power, LLC Submittal of "Combustible Gas Control," TR-0716-50424, Revision 1*, March 28, 2019.
[58] NuScale Power, LLC, *Accident Source Term Methodology*, TR-0915-17565, Revision 3.
[59] FSER Section 12.3.4.1.3
[60] Hydrogen monitoring is required for all water-cooled reactors; oxygen monitoring is required only for reactors that rely on an inerted containment atmosphere for combustible gas control. The NuScale containment is not inerted for combustible gas control, but because of the partial vacuum conditions would become inerted during an accident sequence. Therefore the design includes the capability for oxygen monitoring.

regulation underwent risk-informed revision, wherein NRC concluded that combustible gas control was necessary only for severe accidents.[61]

10 CFR 52.47(a)(2)(iv) establishes the offsite dose limits for a DCA. An applicant must show that for a fission product release into containment resulting for a core damage accident,[62] the containment leak rate, any fission product cleanup systems intended to mitigate the accident consequences, and the postulated site parameters maintain offsite doses within the 25 rem total effective dose equivalent (TEDE) limits.

General Design Criterion 19 establishes the control room dose limit.[63] The design must provide adequate radiation protection "to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem" TEDE.

Several Three Mile Island-related requirements address radiation protection following accidents. Each of these rules assumes an accident source term equivalent to the core damage accident postulated for the offsite dose analysis.

- 10 CFR 50.34(f)(2)(vii) requires an applicant to "perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment." The associated guidance[64] prescribes the GDC 19 dose limit (5 rem whole body[65]) for operator access to those "important areas."
- 10 CFR 50.34(f)(2)(xxvi) requires an applicant to "provide for leakage control and detection in the design of systems outside containment that" may contain accident source term following a severe accident. An applicant must "submit a leakage control program…for minimizing leakage from such systems." The "goal" of the requirement "is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency."
- 10 CFR 50.34(f)(2)(xxviii) requires an applicant to "evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under" severe accident conditions "and make necessary design provisions to preclude such problems."

2.2 NRC Staff Position and NuScale's Response

In Staff's proposed DC rule,[66] Staff state:

> *As documented in Section 12.3.4.1.3 of the final safety evaluation report, there was insufficient information available regarding NuScale combustible gas monitoring system and the potential for leakage from this system outside containment. Without*

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[61] *Combustible Gas Control in Containment*, 68 Fed. Reg. 54,123, Sept. 16, 2003.
[62] 10 CFR 52.47(a)(2)(iv) requires an accident with "fission product release from the core into containment" be analyzed. A footnote states that "these accidents have generally been assumed to result in substantial meltdown of the core" (emphasis added). NuScale argued that the general assumption of core damage should not apply to the NuScale design due to an extremely low likelihood of core damage occurring with an intact containment, but Staff disagreed. See *NuScale Power, LLC Submittal of WP-0318-58980, "Accident Source Terms Regulatory Framework,"* Letter LO-0518-59973, May 16, 2018 (ML18136A850), and SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," Aug. 16, 2019.
[63] 10 CFR 50 Appendix A, Criterion 19.
[64] NUREG-0737, Clarification of TMI Action Plan Requirements, Nov. 1990, Item II.B.2.
[65] GDC 19 was subsequently amended to a 5 rem TEDE dose limit.
[66] SECY-21-0004, *Proposed Rule: Nuscale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)*, Enclosure 1, *Proposed Rule*, Jan. 14, 2021.

> *additional information regarding the potential for leakage from this system, the NRC was unable to determine whether this leakage could impact analyses performed to assess main control room dose consequences and offsite dose consequences to members of the public and whether this system can be safely re-isolated after monitoring is initiated due to potentially high dose levels at or near the isolation valve location.*

FSER Section 12.3.4.1.3 provides Staff's specific regulatory basis for their conclusion that the design of CGM could not be resolved by the DC. There, Staff state first:

> *10 CFR 50.34(f)(2)(xxvi) requires applicants to provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term (i.e., a core damage source term) radioactive materials following an accident. However, the applicant has not provided information necessary for the staff to address the leakage control and detection in systems outside containment, such as the maximum allowable total leakage from the systems used to perform combustible gas monitoring.*

Staff omitted essential details concerning this requirement. As NuScale documented in response to eRAI 9690, 10 CFR 50.34(f)(2)(xxvi) does not impose a requirement for a quantitative dose assessment or a dose limit related to leakage from systems outside containment. As described in the corresponding NUREG-0737 Item II.B.2, the only requirement is that licensees and applicants develop and submit a leakage control program "to reduce leakage to as-low-as-practical levels."[67] The wording of 10 CFR 50.34(f)(2)(xxvi) is consistent with NUREG-0737 in that it omits a dose criterion and describes the "goal" as <u>minimizing</u> potential exposures. NuScale surveyed numerous leakage control programs from the industry, and each specifies that the acceptance criterion for leakage in systems subject to this requirement is "as low as practical," not a quantified leakage limit.

NRC knows how to set a dose limit where it intends to do so,[68] but here NRC did not. That intent is confirmed by another, interrelated TMI Action Item. NUREG-0737 Item II.B.2 (the radiation and shielding design review per 10 CFR 50.34(f)(2)(vii)) does prescribe a dose limit for operator access to important areas and requires an analysis to evaluate that dose. Yet Item II.B.2 states unequivocally: "Radiation from leakage of systems located outside of containment need not be considered for this analysis," and "leakage measurement and reduction is treated under Item III.D.1.1." It is specious for Staff to impose a dose limit where there intentionally is none, and to then conclude that NuScale must specify a "maximum allowable total leakage" and analyze the resulting doses from CGM in order to "provide assurance that dose criteria are not exceeded."[69]

Next, Staff state in FSER Section 12.3.4.1.3 that they are

> *…unable to reach a conclusion that [the CGMC] can be re-isolated in order to mitigate potential leakage from these systems that may impact the ability to demonstrate compliance with the requirements of 10 CFR 50.34(f)(2)(xxviii) to limit radiation exposure to control room operators; the requirements of 10 CFR 52.47(a)(2)(iv) to limit exposure to members of the public; and the requirements of 10 CFR 50.34(f)(2)(vii) to perform radiation and shielding design reviews…*

The concern over re-isolation arises because, when postulating that operators use CGM some 24 hours after an extremely unlikely core damage accident, Staff further postulated that those operators may

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[67] See NUREG-0737 Item III.D.1.1, "Documentation Required."
[68] *Cf.* 10 CFR 50.34(f)(2)(viii), another TMI Action Item that requires the capability to sample post-accident "without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities."
[69] FSER Section 12.3.4.1.3.

discover leakage from the CGM equipment that requires re-isolation to limit release. Here again, Staff's regulatory basis does not hold up. Each of these cited regulations are addressed in turn.

<u>10 CFR 50.34(f)(2)(xxviii)</u>

10 CFR 50.34(f)(2)(xxviii) does require an applicant to evaluate and address potential radiological leakage into the control room to ensure control room habitability under core damage source term conditions. But that requirement, TMI Action Item III.D.3.4, is redundant to GDC 19 for any new applicant. As NUREG-0737 makes clear, the intent of the requirement was to ensure existing licensees were in compliance with the control room habitability criteria of Standard Review Plan (SRP) Section 6.4.[70] Older facilities may not have been licensed under the SRP or subsequent design changes may have impacted continued compliance.[71]

For a new reactor, compliance with GDC 19 as implemented by SRP 6.4 achieves compliance with 10 CFR 50.34(f)(2)(xxviii). The current version of SRP 6.4 specifies that for new reactors implementing an alternate source term, "the guidance on dose analysis of Regulatory Guide 1.183 is applicable" for ensuring control room radiological habitability. Regulatory Guide (RG) 1.183 does not require that leakage from severe accident mitigation features be included in the source term analysis.[72] Per RG 1.183 Appendix A, the scope of potential radiation sources analyzed includes containment leakage and engineered safety features—in other words, safety-related systems credited for meeting dose limits following a design basis accident. In contrast, RG 1.183 states that "containment purging capabilities … maintained for purposes of severe accident management [that] are not credited in any design basis analysis" do not need to be evaluated for dose contribution.

NRC Staff reasoned that although CGMC is not an ESF, it does not look like containment purging either.[73] The only function of CGMC is monitoring following a beyond-design-basis accident; it is not credited in any design basis analysis. Instead of recognizing RG 1.183's clear and logical distinction between systems necessary to mitigate a design basis accident and those that perform only severe accident functions, Staff erroneously concluded that whether the radiological source was piping outside containment was the pertinent question. Under this view, a large radiological release during a severe accident (containment purge) could be excluded from dose analysis but small line leakage cannot.

Because CGMC is only a severe accident monitoring feature it is not required by RG 1.183 to be analyzed in assessing control room doses. Therefore, the NuScale control room dose analysis meets RG 1.183, meets SRP 6.4, and thereby demonstrates compliance with 10 CFR 50.34(f)(2)(xxviii) to limit radiation exposure to control room operators.

<u>10 CFR 52.47(a)(2)(iv)</u>

Staff next cite CGMC leakage as precluding their finding that offsite dose requirements of 10 CFR 52.47(a)(2)(iv) are met. It is uncontroverted that CGM is a requirement only for beyond design basis accidents, as stated in the rule itself. The NuScale design does not need and would not use CGM for any design basis accident.[74] The offsite dose limits of 10 CFR 52.47(a)(2)(iv) apply to the most limiting design basis accident (which NuScale refers to as the maximum hypothetical accident, or MHA).[75] Although that

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[70] NUREG-0737 Item III.D.3.4, "Clarification" ("…licensees that meet the criteria of the SRPs should provide the basis for their conclusion that SRP 6.4 requirements are met.")
[71] See NUREG-0660, Item III.D.3.4: "NRR will require all facilities that have not been reviewed for conformance to … [SRP 6.4] to do the evaluations and establish a schedule for necessary modifications."
[72] Regulatory Guide 1.183, Rev. 0, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000.
[73] Transcript, Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee, Open Session, Nov. 20, 2019, p. 356.
[74] Even in a severe accident, the design is demonstrated to maintain containment integrity for at least 72 hours regardless of combustible gas levels.
[75] While RAI 9690 also identified GDC 19 as part of the regulatory basis, it was omitted in Staff's FSER discussion.

MHA assumes a source term that would in reality result only from a severe accident, the rule deterministically imposes it as a design basis source term for the specific purposes of evaluating containment performance and plant site characteristics under design basis conditions.

The MHA dose evaluation is distinct from other regulatory requirements, such as the limiting LOCA for ECCS performance and containment integrity, for which severe core damage is not allowed to occur. A mechanistically-derived core damage event is beyond the design basis of all licensed plants. Where NRC substantiates a safety concern from a potential beyond-design-basis core damage accident, the agency has prescribed design enhancements to assist in coping and reducing the consequences from it. In other words, NRC does not require that potential mechanistic core damage sequences such as one resulting from station blackout meet the design basis dose limits of 10 CFR 52.47(a)(2)(iv). By expecting NuScale to include leakage from the beyond-design-basis accident function of CGM in evaluating conformance with offsite dose limits, NRC Staff's position contradicts agency practice and the regulatory framework.

10 CFR 50.34(f)(2)(vii)

Finally, Staff cite CGMC leakage as precluding their finding on the requirements of 10 CFR 50.34(f)(2)(vii) to perform radiation and shielding design reviews. As noted in Section 2.1 above, this requirement and its guidance do impose a 5 rem dose limit for operator access to "important areas." Again, though, NUREG-0737 explicitly excludes leakage from CGMC from the dose analysis required: "Radiation from leakage of systems located outside of containment need not be considered for this analysis." NUREG-0737 instead relies on the leakage control program of Item III.D.1.1 and its as-low-as-practical leakage control standard to reasonably assure acceptable doses from such systems. The NuScale DCA requires a leakage control program to ensure CGMC leakage is as low as practical. Therefore, the NuScale DCA satisfies 10 CFR 50.34(f)(2)(vii).

For the reasons discussed above, each regulation that Staff cite in FSER 12.3.4.1.3 as a shortcoming is satisfied by the NuScale DCA without a specified CGMC leakage limit or dose analysis. In the proposed DC rule of SECY-21-0004, Staff introduced an additional regulatory basis:

> *The issue may be resolved by performing radiation dose calculations and demonstrating that doses will remain within applicable dose limits in 10 CFR part 20, "Standards for Protection Against Radiation."*

The dose limits of Part 20 are occupational and public dose limits for operational exposures. Part 20 does not impose limits on doses from any accident condition, let alone a beyond-design-basis core damage accident that would necessitate combustible gas monitoring. In fact, 10 CFR 20.1201 contemplates doses in excess of annual occupational limits due to an emergency condition.[76] NuScale recognizes that the Commission has stated that the Part 20 dose limits should remain guidelines for emergency response exposures.[77] However, that general guideline should not be understood to extend the regulatory reach of Part 20 or the various regulations cited by Staff in FSER 12.3.4.1.3, which have a different or purposefully lack a dose criterion. Imposing the Part 20 dose limits as a severe accident design dose criterion changes the NRC's regulations without following the process to do so.

2.3 NuScale's Position

2.3.1 CGM leakage is not a safety concern

[76] "Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits…" 10 CFR 20.1201(b).

[77] *Standard for Protection Against Radiation*, 56 Fed. Reg. 23,360, May 21, 1991, p. 23,365. *But see* NUREG-1736, *Consolidated Guidance: 10 CFR Part 20- Standards for Protection Against Radiation*, Oct. 2001, p. 3-2 ("This section also makes clear that Part 20 regulations do not apply to licensee activities performed in order to mitigate potential health and safety consequences from accidents or from other incidents involving radioactive material.").

Potential leakage from the CGM presents no undue risk to the health and safety of the public or site personnel.

Consistent with NRC requirements and precedent, NuScale postulates for the purposes of radiological dose analyses a “maximum hypothetical accident” source term that would result from substantial damage to the reactor core. NuScale’s “core damage event” (CDE) source term represents a set of key source term parameters derived from a spectrum of surrogate severe accident scenarios. No design basis accident in the NuScale design uncovers the core or damages fuel, let alone results in severe core damage. The CDE has a probability of occurring less than $3x10^{-10}$ per year. The estimated doses from the CDE, conservatively calculated in accordance with traditional Chapter 15 methods and approved codes, are less than 3 percent of the exclusion area boundary dose limit[78] and less than half of the control room dose limit.[79]

Thus, the entirety of containment leakage, which has been conservatively analyzed using maximum leakage rates assured through design, preoperational, and operational testing, amounts to much less than the acceptance limits. Events that would yield a CDE-level source term in the 160 MWt NuScale reactor are predicted to occur with a frequency orders of magnitude less than what the Safety Goals allow for a large release[80] from a gigawatt-scale plant. At the frequency of intact containment core damage at a NuScale reactor, any release from the CGM pathway would necessarily be non-risk significant. The required leakage control program will minimize leakage from CGM.

2.3.2 The DCA meets all applicable requirements

Staff’s conclusion that leakage from NuScale’s CGMC must be included in dose assessments is erroneous because it conflates two distinct aspects of NRC’s regulations. On the one hand, required dose analyses make use of a postulated non-mechanistic, stylized core damage source term. On the other hand, CGM is required to assist emergency response following a mechanistically-possible beyond-design-basis severe accident.

Staff would have NuScale analyze the CGM doses because they reason that CGM would be used following the core damage event that constitutes NuScale’s MHA. If applied elsewhere, this position would foreclose design features NRC has already approved or required to mitigate the consequences of core damage events. For example, containment venting as a remedy to over-pressurization, as mandated for some designs in response to Fukushima, would not be possible if the Chapter 15 dose assessment—based on a core damage source term—has to include the dose from containment venting. The CGM function required by 10 CFR 50.44 is analogous in its purpose and basis.

As detailed in Section 2.2 above, NuScale’s DCA complies with the relevant regulatory requirements because (1) the specified dose limits do not require consideration of radiological sources that are not used to mitigate a design basis event, and (2) implementation of a leakage control program to reduce leakage to as-low-as-practical levels complies with 10 CFR 50.34(f)(2)(xxvi) without modeling the dose consequences of hypothetical leakage from combustible gas monitoring.

2.3.3 Precedent supports NuScale’s position

Precedent supports NuScale's position. For example:

- The ESBWR design includes a Containment Monitoring System (CMS) that is partially routed outside of containment and is used to monitor the hydrogen and oxygen gas concentrations in the

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[78] 0.63 rem TEDE compared to the acceptance limit of 25 rem.
[79] 2.14 rem TEDE compared to the 5 rem TEDE limit of GDC 19.
[80] *Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication*, 51 Fed. Reg. 28,044, Aug. 21, 1986.

drywell and wetwell during post-accident conditions. No leakage is assumed from the CMS in the accident dose evaluations, and the leakage control program specifies that leakage from that and other systems containing accident source term outside containment be "as low as practicable."[81]

- The APR1400 includes a hydrogen monitor outside containment as part of its CMS. The CMS is included in scope of the leakage control program without quantifying an acceptance criterion.[82]

- The ABWR includes 13 systems outside containment that may contain source term following an accident, including post-accident sampling, process sampling, containment atmospheric monitoring, fission product monitoring. Such systems are subject to the leakage control program and leakage from them does not appear to be included in accident dose evaluations.[83] The ABWR SER concludes that a requirement for COL applicant's procedures to "reduce detected leakages to lowest practical levels" satisfied TMI item III.D.1.1.[84]

In response to this precedent, NRC Staff noted differences between the ESBWR CGM equipment and the NuScale design, and cited reasons why the CGM leakage might be more significant for NuScale.[85] However, the differences cited are immaterial to the matter at hand.[86] Staff did not address the other design certification precedents.

Review of operating plant licensing basis information indicates that NuScale's approach is also consistent with the plants that were initially subject to the TMI requirements. In justifying removal of post-accident sampling systems (PASS) from their plants, the CE Owners Group documents that:

> *Based on plant-specific experience using PASS, several utilities have included leakage from PASS during post-accident plant sampling in demonstrating adherence to 10CFR100.11 and GDC 19 exposure guidelines. For example, a 350 cc/hr leakage from the PASS is estimated to contribute approximately 2 rem to the thyroid doses at the exclusion area boundary and low population zone. PASS leakage is also estimated to contribute 1.7 rem to the Control Room thyroid dose.*[87]

Even though PASS is potentially a significant contributor to offsite and control room doses, only "several" plants included that potential dose in their offsite and control room dose consequence analysis. Many, it follows, excluded that dose contributor and relied only on their leakage control program to maintain leakage as low as practical. As with NuScale's CGM, PASS in those facilities served a post-accident monitoring function to inform emergency response, would contain accident source term following core damage, and is not safety-related. Reliance on the leakage control program was compliant for those facilities and must also be for NuScale's design.

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[81] ESBWR DCD, Chapter 7, Figure 7.5-1; Chapter 15; Chapter 16, Generic Technical Specification 5.5.2.
[82] APR1400 DCD, Section 6.2.5.2.2; Chapter 16, Generic Technical Specification 5.5.2.
[83] ABWR DCD, Section 1A.2.34, Rev. 4.
[84] NUREG-1503, Section 20.5.38.
[85] Transcript, Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee, Open Session, Nov. 20, 2019, p. 193.
[86] Staff favorably noted that the ESBWR CMS is redundant. However redundancy would not reduce the likelihood or magnitude of potential leakage. Staff also noted the CMS is safety-related and Seismic Category 1, but those attributes do not preclude leakage or the analysis of it in other contexts (see discussion of RG 1.183 and ESF systems, above). Per NRC regulation and guidance, the CGMC does not need to be safety-related (10 CFR 50.44(c)(3) and Regulatory Guide 1.7, Section 2.1). Still, the ACRS remarked that the system was not "safety grade." Advisory Committee on Reactor Safeguards, *Safety Evaluation of the NuScale Power, LLC Topical Report TR- 0915-17565, Revision 3, "Accident Source Term Methodology," and Source Term Area of Focus Review for the NuScale Small Modular Reactor*, Dec. 20 2019.
[87] CEOG, CENPSD-1157, Rev. 1, pg. 11 (ML003699802) (emphasis added).

2.3.4 If combustible gas monitoring leakage is a safety concern, it should be addressed generically

Based on the above, NuScale believes NRC Staff have either misconstrued the existing regulatory requirements or seek to implement a new interpretation based on perceived safety significance. If the latter, then NuScale believes this issue must be addressed generically and not as a new requirement unique to the NuScale design. As indicated by the CE Owners Group example, small leaks from small lines outside containment may be meaningful dose contributors during a severe accident. This is the precise issue NRC addressed with TMI item III.D.1.1's requirement to reduce leakage to levels as low as practical. If NRC's current view is that approach is inadequate and quantified dose analysis and limits are now a requirement for those same systems, NRC should pursue the matter generically. NuScale believes such a review would conclude that the leakage is not risk significant whether for existing designs or for NuScale's design with a CDF orders of magnitude lower.

3.0 Conclusion

The NuScale DCA presents a CGMC design and sufficient information for the NRC to conclude that it meets relevant regulatory requirements. Therefore, Staff's proposal to exclude CGM leakage from issue resolution is unwarranted.

June 9, 2008

The Honorable Nancy Pelosi
Speaker of the House of Representatives
Washington, D.C. 20515

Dear Madam Speaker:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am pleased to provide an NRC draft bill that would amend the Atomic Energy Act of 1954 (AEA) and the Energy Reorganization Act of 1974. These provisions are intended to enhance the efficiency of NRC operations, the security of NRC-regulated facilities, and compliance with NRC regulatory requirements.

More specifically, this legislation would accomplish the following objectives:

(1) Eliminate the requirement to hold uncontested hearings on applications to the NRC for granting a construction permit for a utilization or production facility, or for granting a combined construction and operating license under the AEA, or for issuance of a license under AEA sections 53 and 63 for the construction and operation of any uranium enrichment facility;

(2) Reduce the program briefings of the Commission on NRC's Equal Employment Opportunity program from two to one per year;

(3) Ensure that NRC certificate holders and their contractors and subcontractors will be subject to civil penalties for AEA violations and violations of certain provisions of the Nuclear Waste Policy Act of 1982; and

(4) Provide the Commission with authority to require fingerprinting of (a) individuals designated by licensees or certificate holders to review the trustworthiness and reliability of individuals who are already required to be fingerprinted under AEA section 149, (b) employees of licensees or certificate holders who have authority to grant unescorted access to a utilization facility, or to designated radioactive material or other property, and (c) principal operating officers (or their equivalent) of individuals and entities already required to conduct fingerprinting under AEA section 149.

A draft bill and a legislative memorandum explaining the need for the provisions of the bill are enclosed with this letter.

Sincerely,

/RA/

Dale E. Klein

Enclosures:
1. Draft Bill
2. Legislative Memorandum

Identical letter sent to:

The Honorable Nancy Pelosi
Speaker of the House of Representatives
Washington, D.C. 20515

The Honorable Richard B. Cheney
President of the Senate
Washington, D.C. 20510

The Honorable Thomas R. Carper
Chairman, Subcommittee on Clean Air
and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, D.C. 20510
cc: Senator George V. Voinovich

The Honorable Barbara Boxer
Chairman, Committee on Environment
and Public Works
United States Senate
Washington, D.C. 20510
cc: Senator James M. Inhofe

The Honorable Rick Boucher
Chairman, Subcommittee on Energy
and Air Quality
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515
cc: Representative Fred Upton

The Honorable John D. Dingell
Chairman, Committee on Energy
and Commerce
United States House of Representatives
Washington, D.C. 20515
cc: Representative Joe Barton

The Honorable Peter J. Visclosky
Chairman, Subcommittee on Energy
and Water Development
Committee on Appropriations
United States House of Representatives
Washington, D.C. 20515
cc: Representative David L. Hobson

The Honorable Byron Dorgan
Chairman, Subcommittee on Energy
  and Water Development
Committee on Appropriations
United States Senate
Washington, D.C. 20510
cc: Senator Pete V. Domenici

DRAFT BILL

Be it enacted by the Senate and House of Representatives of the United States of America in Congress assembled, that this Act may be cited as the "Act to Streamline the Nuclear Regulatory Commission's Licensing Process and Administrative Efficiency".

SEC. 2. HEARINGS UNDER ATOMIC ENERGY ACT OF 1954.

(a) Section 189 a.(1)(A) of the Atomic Energy Act of 1954 (42 U.S.C. 2239(a)(1)(A)) is amended by--

(1) in the second sentence--

(i) deleting that portion of the sentence that begins with "The Commission" and ends with "Federal Register, on" and inserting "On";

(ii) inserting "or an operating license" after "construction permit" each time "construction permit" is used in the sentence; and

(iii) deleting the period at the end of the sentence; and

(2) in the third sentence–

(i) deleting that portion of the sentence that beings with "In cases" and ends with "such a hearing";

(ii) deleting "therefor" and inserting "for a hearing"; and

(iii) deleting "issue an operating license" and inserting "issue a construction permit, an operating license,".

(b) Section 189 of the Atomic Energy Act of 1954 (42 U.S.C. 2239) is further amended by--

(1) in the second sentence of subsection a.(2)(A) (42 U.S.C. 2239(a)(2)(A)), deleting "required hearing" and inserting "hearing held by the Commission under this section"; and

(2) in subsection b. (42 U.S.C. 2239(b)), revising paragraph (2) by deleting "to begin operating" and inserting "to operate".

(c) The first sentence of subsection b. of section 185 of the Atomic Energy Act of 1954 (42 U.S.C. 2235(b)) is amended by deleting "After holding a public hearing under section 189 a.(1)(A)," and inserting "After holding a hearing under section 189 a.(1)(A), or if the Commission has determined that no hearing is required to be held under section 189 a.(1)(A),".

(d) Section 193(b) of the Atomic Energy Act of 1954 (42 U.S.C. 2243(b)) is amended by–

(1) in paragraph (1), deleting "on the record with regard to the licensing of the construction and operation of a uranium enrichment facility under sections 53 and 63" and inserting ", if a person whose interest may be affected by the construction and operation of a uranium enrichment facility under sections 53 and 63 has requested a hearing regarding the licensing of the construction and operation of the facility"; and

(2) in paragraph (2), deleting "Such hearing" and inserting, "If a hearing is held under paragraph (1), the hearing".

(e) The amendments in this section shall apply to all applications and proceedings pending before the Commission on or after the date of enactment of this section.

SEC. 3 REPORT ON EQUAL EMPLOYMENT OPPORTUNITY PROGRAM.

Section 209(c) of the Energy Reorganization Act of 1974 (42 U.S.C. 5849(c)) is amended by deleting "semiannual public meetings" and inserting "an annual public meeting".

SEC. 4. CIVIL MONETARY PENALTIES.

The first sentence of section 234 a. of the Atomic Energy Act of 1954 (42 U.S.C. 2282(a)) is amended by—

(1) inserting "(including a contractor or subcontractor of a licensee or certificate holder of the Commission or of an applicant for a Commission license or certificate)" after "Any person"; and

(2) striking "any licensing or certification provision of section 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, 109, or 1701" and inserting: "any Commission regulatory requirement issued pursuant to or contained in this Act or section 133, 137, 180, or 218(a) of the Nuclear Waste Policy Act of 1982 (42 U.S.C. 10101 *et seq*.)".

SEC. 5. ENHANCED FINGERPRINTING REQUIREMENTS.

Section 149 a.(1) of the Atomic Energy Act of 1954 (42 U.S.C. 2169) is amended by adding the following new subparagraph after subparagraph (B):

"(C) In addition to the foregoing fingerprinting requirements of this paragraph, the Commission may require an individual or entity described in subparagraph (A)(ii) to fingerprint—

"(i) any individual who has been designated by the individual or entity described in subparagraph A(ii) (or by a contractor or subcontractor of such individual or entity) to determine the trustworthiness and reliability of an individual who is required to be fingerprinted under subparagraph (B)."

"(ii) any individual who is in the employ of the individual or entity described in subparagraph (A)(ii) (or a contractor or subcontractor of such individual or entity) and who has authority relating to provision of unescorted access to a facility, radioactive material, or other property described in subparagraph (B)(i);" or

"(iii) any individual who is, or holds a position equivalent to, the principal operating officer, or alternate principal operating officer, of the individual or entity described in subparagraph (A)(ii)."

LEGISLATIVE MEMORANDUM

SEC. 2. HEARINGS UNDER ATOMIC ENERGY ACT OF 1954.

This section would eliminate the requirement of section 189 a. of the Atomic Energy Act of 1954 that the Commission hold a hearing in proceedings on each application for granting a construction permit for a nuclear reactor facility under section of 103 or 104 b. of the Act or for granting a construction permit for a testing facility under section 104 c. of the Act or for granting a combined construction and operating license under section 185 of the Act, even if no person whose interest is determined to be affected by the proceeding has requested a hearing or been granted intervention. Similarly, the requirement of section 193(b) of the Act that would require a hearing in an uncontested proceeding to license construction and operation of a uranium enrichment facility would be eliminated. In the latter case, the requirement that such hearings be on the record – the only such requirement with respect to an adjudicatory hearing contained in the Atomic Energy Act – would also be eliminated.

The Commission has found that there is not much added value in holding uncontested hearings. Over fifty years ago, a 1957 amendment added the requirement for mandatory hearings to the Atomic Energy Act of 1954. Since then, the means and methods for public access to the Commission's actions, both legally mandated and voluntarily undertaken, have become numerous and significant. Enactment of the Freedom of Information Act and the Government in the Sunshine Act, the advent of the internet, and web-based access to NRC's documented actions through the NRC's Agency-wide Documents Access and Management System (ADAMS) have all contributed to making NRC's actions transparent and accessible. Furthermore, even with the elimination of the mandatory hearing requirements, the agency staff would continue to prepare a safety analysis report and an environmental statement, and the Advisory Committee on Reactor Safeguards would continue to provide an independent assessment of each power reactor application. The Commission could not issue a license until it had concluded that all regulatory requirements had been satisfied. And, of course, this would in no way affect the right of persons whose interest are affected from requesting a hearing on specific matters.

The changes in Commission licensing procedures addressed in this section would take effect upon enactment of the legislation, and would apply to new applications and proceedings, as well as any pending proceedings. This would obviate the need for the Commission to expend resources on uncontested proceedings.

These changes will streamline the Commission's licensing process under the Atomic Energy Act, saving time and scarce resources in a period in which a large number of reactor licensing applications are expected to be submitted to the Commission.

SEC. 3. REPORT ON EQUAL EMPLOYMENT OPPORTUNITY PROGRAM.

The Energy Reorganization Act of 1974 currently requires two annual program briefings of the Commission on the NRC's Equal Employment Opportunity (EEO) Program. These briefings involve extensive research and data-collection related to the agency's accomplishments, program assessments, and challenges, and are resource-intensive. History has demonstrated that the agency does not experience substantial changes in a six-month period and that annual briefings would be sufficient to keep the Commission apprised of the EEO and related programs.

To the Commission's knowledge, the NRC is the only Federal agency that is required to hold public briefings on the agency's EEO program. Holding only one briefing per year would not detract from the purpose of the EEO briefings, and would conserve agency resources.

Furthermore, this reduction would not negate the agency's efforts to ensure that equal employment opportunity is, as required by Executive Order 11478, an "integral part of every aspect of personnel policy and practice in the employment, development, advancement, and treatment of civilian employees in the Federal Government."

SEC. 4. CIVIL MONETARY PENALTIES.

This section would expand the Nuclear Regulatory Commission's authority to issue civil penalties. The Commission currently has authority, under section 234 of the Atomic Energy Act of 1954 (AEA), to issue civil penalties to licensees and certificate holders of the Commission. However, that authority only extends to violations of licensing or certification provisions listed in section 234 of the AEA (or any rule, regulation, or order, or any term, condition, or limitation of a license or certificate issued thereunder). The amendment would also clarify that contractors and subcontractors of a Commission licensee or certificate holder, or of an applicant for a Commission license or certificate, are subject to civil penalties.

Congress amended section 234 of the AEA in the Omnibus Consolidated Rescissions and Appropriations Act of 1996 (Public Law 104-134), to give the Commission authority to issue civil penalties for violations related to gaseous diffusion enrichment plants, which must receive a certificate of compliance from the Commission, rather than a license, under section 1701 of the AEA. That amendment made certificate holders subject to civil penalties, but only if the certificates were issued pursuant to one of the statutory provisions listed in section 234.

However, the current authority does not extend to all certificate holders. For example, certificates of compliance are also issued by the NRC for the design of spent fuel storage casks under provisions of the Nuclear Waste Policy Act of 1982 (NWPA). Since these provisions are not listed in section 234 of the AEA, the Commission does not currently have the authority under section 234 of the AEA to issue civil penalties based on these certificates of compliance.

Broadening the scope of section 234 of the AEA would authorize the Commission to assess civil penalties based on violation of any Commission regulatory requirement issued pursuant to, or contained in, the AEA or specified sections of the NWPA.

The proposed amendment is necessary to extend the Commission's civil penalty authority over holders of or applicants for certificates of compliance. There is no real basis to distinguish certificate holders from licensees for the purpose of allowing a civil penalty to be imposed as an enforcement sanction.

SEC. 5. ENHANCED FINGERPRINTING REQUIREMENTS.

Currently, NRC is required to direct certain individuals and entities (generally, those licensed or certified to engage in or who have filed an application for a license or certificate to engage in activities subject to NRC licensing or certification, or who have given written notice to the NRC of an intent to file an application for licensing, certification, permitting, or approval of a product or activity subject to NRC regulation) to require fingerprinting of individuals who have unescorted access to certain facilities or to designated materials or other property, or who are permitted access to safeguards information. This amendment would expand that authority with respect to certain other individuals who have security-related responsibilities.

For example, this amendment would authorize the Commission to extend fingerprinting requirements to any individual designated by a licensee or certificate holder to review the trustworthiness and reliability of individuals who are fingerprinted under section 149 a.(1) of the Atomic Energy Act of 1954, based on the results of the identification and criminal records check information obtained from the Attorney General. Because some licensees' reviewing officials or

Trustworthiness and Reliability Officials[1] do not have unescorted access to a utilization facility or to designated radioactive material or other property or access to safeguards information, the Commission currently is not able to require their fingerprinting.

Other examples of individuals in positions that may be subject to NRC fingerprinting requirements under this amendment are individuals who have authority relating to provision of unescorted access to a facility, radioactive material, or other designated property, and individuals who hold the position of principal operating officer or an equivalent position in an enterprise. It is obvious that individuals who are employed in these types of positions can be in a position to do considerable harm. This amendment would enhance security by authorizing fingerprinting and a subsequent FBI criminal history check of those individuals.

The proposed amendment does not direct the Commission to immediately require fingerprinting of the individuals covered, but gives the Commission discretion to determine which of those employees and officers need to be fingerprinted and when such a program should be implemented.

[1] In this context, the terms “reviewing official” and “Trustworthiness and Reliability Official” are used to designate individuals who are assigned the responsibility for analyzing the results of identification and background checks based on the fingerprints of employees currently required to be fingerprinted under section 149 of the Atomic Energy Act of 1954.