INTERIM REPORT

OF THE

AMERICAN NUCLEAR SOCIETY

PRESIDENT'S SPECIAL COMMITTEE ON

SMALL AND MEDIUM SIZED REACTOR (SMR)

GENERIC LICENSING ISSUES

July 2010

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EXECUTIVE SUMMARY

The American Nuclear Society (ANS) has taken a leadership role in addressing the licensing issues for Small and Medium Sized Reactors (SMRs). The licensing and eventual deployment of "right sized" SMRs would lead to

- job creation
- export of U.S. goods and services
- benefits to national security and energy policy
- reductions in greenhouse gas emissions.

The United States has licensed and built small reactors since the 1950s with numerous land-based and sea-based platforms. These efforts proved the safety and security of light water–cooled, gas-cooled, and metal-cooled SMR technologies. In the past decade, there has been evolving dialogue between SMR reactor designers, project developers, and the U.S. Nuclear Regulatory Commission (NRC). Of particular note was the 2009 NRC SMR workshop, which encouraged dialogue between SMR developers and the NRC prior to license application. The NRC was clear in its message:

> [For each issue] "the SMR community should provide a consensus approach."

Recognizing the potential for SMRs to change the social and energy supply paradigms, ANS President Tom Sanders established the ANS President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) in 2010. The SMR Special Committee took up his message and led the nuclear science and engineering community in organizing a forum for technical dialogue on SMR licensing issues. President Sanders directed the SMR Special Committee to develop solutions to SMR generic licensing issues by being

- issue driven and focused on technology-neutral solutions
- inclusive with crosscutting participation from ANS members with every SMR perspective
- collaborative with the U.S. Department of Energy (DOE), the Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), and the International Atomic Energy Agency (IAEA) and other SMR programs.

The output from the SMR Special Committee would be a set of white papers that would be an ANS product for use by the SMR community.

As shown in Exhibit 1, two dozen SMR generic issues were identified. The issues were prioritized and assigned to one of three subcommittees. The subcommittees were organized as follows:

- Subcommittee A: Licensing Framework Issues
- Subcommittee B: Licensing Application Issues
- Subcommittee C: Licensing Design and Manufacturing Issues.

The subcommittee leaders managed the preparation of white papers for the generic issues. This Interim Report contains the first set of white papers completed by the SMR Special Committee.

ANS members from the SMR community responded with enthusiasm and commitment to an invitation to join the SMR Special Committee. The SMR Special Committee members are associated with more than three dozen organizations representing government, universities, national laboratories, reactor designers, industry consultants, technical service providers, law firms, and electric power companies. The SMR Special Committee membership and affiliations are shown in Exhibit 2.

The SMR Special Committee leadership was provided by the following individuals:

Philip Moor (Chair), *High Bridge Associates*
John Kelly (Cochair), *Sandia National Laboratories*
Charles Hess (Subcommittee A), *The Shaw Group*
Michael Corradini (Subcommittee B), *University of Wisconsin*
Ross Radel (Subcommittee C), *Sandia National Laboratories*.

Further to President Sanders' directive, speed was vital to the SMR Special Committee's work because SMRs now have the attention of legislators, the energy industry, regulators, and the public. Eight white papers were written in six months. Another six white papers are underway, scheduled for completion by November 2010.

The issue topics of the completed white papers are the following:

- Subcommittee A: Licensing Framework Issues
    - staffing
    - NRC fees
    - Price-Anderson
    - applicability of light water reactor (LWR) requirements to SMRs
- Subcommittee B: Licensing Application Issues
    - risk-informed regulation
    - physical security
- Subcommittee C: Licensing Design and Manufacturing Issues
    - manufacturing licenses
    - inspections, tests, analyses, and acceptance criteria.

A clear trend emerges in the conclusions and recommendation of the completed white papers, namely, that the current U.S. nuclear reactor regulations are focused on the safety and security of large LWRs. The papers illustrate the incompatibilities of the current licensing rules with SMR designs. In general, applicants would have three possible approaches for licensing SMRs:

- seek exemptions to current rules
- NRC rulemaking
- legislative changes.

Each of these approaches implies a specific time frame for implementation, and in many cases the white papers provide near-term solutions as well as long-term solutions aimed at achieving regulatory stability.

The white papers are an ANS product produced by expert volunteers. The white papers represent the limit of what a volunteer staff can produce in a timely manner. In order for ANS to participate further in implementing the conclusions and recommendation of the white papers (e.g., via topical reports, rule change documents, or standards), external funding would be needed.

Looking forward, ANS is clearly part of the SMR future. The excellent work by very talented and enthusiastic ANS members has produced eight white papers. The SMR Special Committee will continue preparing white papers and collaborating with industry and government organizations such as NEI, EPRI, IAEA, DOE, Next Generation Nuclear Plant (NGNP), and others.

The SMR Special Committee has raised the profile of ANS by taking a leadership role in the work needed to safely and securely license SMRs. The SMR Special Committee provides a unique opportunity to bring ANS members together from all sectors of the SMR community to develop informed options for dealing with the generic licensing issues.

EXHIBIT 1
SMR Generic Licensing Issues

Emergency Planning
Passive Safety Systems
Staffing – Human Factors and Operational Issues
Physical Security – Aircraft Impact
Financial Issues – Price-Anderson, Insurance, Financial Qualifications, Decommissioning Fund
NRC Fee Rules
Applicability of Large LWR Requirements to SMRs
Nonelectrical Generation and Process Heat Applications
Prototypes and Their Proximity to Industrial Processes
Manufacturing Licenses
International Codes and Standards
Multi-Module Facilities
Risk-Informed and Performance-Based Licensing Approaches
Probabilistic Risk Assessment

EXHIBIT 2
ANS SMR Special Committee Members and Affiliations

Victoria K. Anderson, *Nuclear Energy Institute*

Mike Anness, *Westinghouse Electric Company*

Stephen Atherton, *General Electric Hitachi Nuclear Energy*

Richard Barrett, *Advanced Systems Technology and Management*

Edward Blandford, *University of California, Berkeley*

John Bolin, *General Atomics*

Mark S. Campagna, *Hyperion Power*

Han Kwon Choi, *URS, Washington Group*

Michael Corradini, *University of Wisconsin*

Richard Denning, *The Ohio State University*

Thomas Fanning, *Argonne National Laboratory*

Paul Farrell, *Radix*

John Ferrara, *Babcock & Wilcox*

Vince Gilbert, *Excel*

Eddie Grant, *Excel*

Tony Grenci, *Westinghouse Electric Company*

Budd Haemer, *Pillsbury*

Jeff Halfinger, *Babcock & Wilcox*

Charles Hess, *The Shaw Group*

Dan Ingersoll, *Oak Ridge National Laboratory*

Andy Kadak, *Massachusetts Institute of Technology*

Sergey Katsenelenbogen, *Advanced Systems Technology and Management*

John Kelly, *Sandia National Laboratories*

T. J. Kim, *Babcock & Wilcox Modular Nuclear Energy*

Jim Kinsey, *Idaho National Laboratory*

David E. Leaver, *Worley Parsons Polestar*

Eric P. Loewen, *General Electric-Hitachi*

Gary Mays, *Oak Ridge National Laboratory*

S. Michael Modro, *International Atomic Energy Agency*

Philip Moor, *High Bridge Associates*

Tom Mulford, *Electric Power Research Institute*

Robert Neibecker, *Bechtel*

Scott Newberry, *Advanced Systems Technology and Management*

Jim Powell, *Radix*

Ted Quinn, *Longenecker and Associates*

Ross Radel, *Sandia National Laboratories*

William Reckley, *Nuclear Regulatory Commission*

Jose Reyes, *NuScale*

Roger Reynolds, *Terrapower*

Doug Rosinski, *Ogletree Deakins*

Steve Routh, *Bechtel*

Walter Sawruk, *ABS Consulting*

Finis Southworth, *Areva*

Jon Thompson, *Nuclear Regulatory Commission*

Ed Wallace, *NuScale Power*

Ruth F. Weiner, *Sandia National Laboratories*

Joe Williams, *Nuclear Regulatory Commission*

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ACRONYMS

"123"	Section 123 of the U.S. Atomic Energy Act of 1954
ABWR	Advanced Boiling Water Reactor
ACRS	NRC Advisory Committee on Reactor Safeguards
ALWR	Advanced Light Water Reactor
ANPR	Advance Notice of Proposed Rulemaking
ANI	American Nuclear Insurers
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
BTP	Branch Technical Position
BWR	boiling water reactor
CCFP	conditional containment failure probability
CFR	Code of Federal Regulations
COL	Combined Construction Operating License
COLA	Combined Construction Operating License Application
CP	Construction Permit
DBT	Design Basis Threat
DC	Design Certification
DCD	Design Certification Document
DCWG	Design Centered Working Group
DoD	U.S. Department of Defense
DOE	U.S. Department of Energy
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
ESP	Early Site Permit
FOCI	Foreign Ownership, Control, and Influence
FSAR	Final Safety Analysis Report
GA	General Atomics
GDC	General Design Criteria
GE	General Electric
GEN	Generation
HSI	Human-System Interface
HTGR	High Temperature Gas Reactor
HVAC	Heating, Ventilating, and Air-Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
INPO	Institute of Nuclear Power Operations
IP	Intellectual Property
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
JSME	Japan Society of Mechanical Engineers
LCS	Littoral Combat Ship

LO	Licensed Operator
LOCA	Loss-of-Coolant Accident
LPR	Low Power Reactor
LRB	Licensing Basis Review
LWA	Limited Work Authorization
LWR	Light Water Reactor
MCR	Main Control Room
MHTGR	Modular High Temperature Gas-Cooled Reactor
ML	Manufacturing License
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NGNP	Next Generation Nuclear Plant
NPF	Nuclear Plant Facility
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NPMHTGR	New Production Modular High Temperature Gas-Cooled Reactor
NPT	Nuclear Non-Proliferation Treaty of 1972
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUREG	U.S. Nuclear Regulatory Commission Regulation
OL	Operating License
PAAA	Price-Anderson Amendments Act
PRA	Probabilistic Risk Assessment
PRISM	Power Reactor Innovative Small Module
PWR	Pressurized Water Reactor
R-COLA	Reference Combined Construction Operating License Application
RG	Regulatory Guide
RO	Reactor Operator
RSS	Reactor Shutdown System
S-COLA	Standard Combined Operating License Application
SAMA	Severe Accident Mitigation Alternatives
SAMDA	Severe Accident Mitigation Design Alternative
SAR	Safety Analysis Report
SDA	Standard Design Approval
SDO	Standards Development Organization
SER	Safety Evaluation Report
SFR	Sodium-Cooled Fast Reactor
SMART	Small Modular Advanced Reactor Technology
SMR	Small and Medium Sized Reactor
SRMs	Staff Requirements Memoranda
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SSCs	Systems, Structures, and Components
TMI	Three Mile Island
U.N.	United Nations

OPERATIONS STAFFING ISSUES RELATING TO SMRs

Lead Writer – Tony Grenci
Assistant Writer – Robert ("Budd") Haemer

American Nuclear Society (ANS)

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1.0 INTRODUCTION

One of the assumed characteristics of Small and Medium Sized Reactors (SMRs) is the potential to require a much smaller staff per reactor than existing large reactors. (This paper focuses on operator staffing for SMRs and does not address other aspects of staffing such as for plant administration, maintenance, or security. Security issues are addressed in a separate ANS white paper: "Physical Security for Small Modular Reactors.") In sum, staffing levels may be reduced for a typical SMR Nuclear Plant Facility (NPF) without compromising safety. The small size of the SMR NPF and its inherently safe, passive design eliminate the need for a plant operation staff of the magnitude employed at current commercial Nuclear Power Plants (NPPs). The operations of an SMR are more typically automatic, and less human intervention is required. Given the simpler and more automated operation of advanced SMR designs, operator action to place the plant in a safe condition for either design-basis or beyond-design-basis ("severe") accidents generally requires passive observation and confirmation, not active intervention. Extending this argument, the number of Licensed Operators (LOs) in a multi-modular SMR facility of equivalent cumulative output may also be less than would be required for equivalent large plants of the Generation (GEN) III/III+ designs.

In either of these cases, the reduced staffing requirements could be accomplished with submittal of and approval of exemption requests to current regulations until such time as the regulations would be updated to accommodate the new SMR designs. Because SMR designs provide for simpler operation and increased automation, the number of on-shift LOs can be reduced, and their collateral (nonlicensed) duties can be increased without compromising safety. Therefore, the total operating staff for the facility can be dramatically reduced.

The purpose of this white paper is to promote discussion that results in the U.S. Nuclear Regulatory Commission (NRC) approving reduced operator staffing for SMRs based on clearly identifiable criteria and to obtain tailored guidance on the number and duties of LOs within the framework of existing regulations. Early discussions between representatives of SMR applicants and the NRC staff concerning staffing should be held to determine, among other things, whether seeking such an exemption in one or more areas will be necessary.

2.0 BACKGROUND

The NRC regulates facility staffing through its regulations and a collection of guidance documents issued by the NRC staff. Operator staffing is an important subset of the overall staffing requirements to be considered for SMR designs, and when considering the overall reductions in plant staffing based on the size and simplicity of SMRs, operating staff could be much larger in proportion of the total staffing than for existing plants. NRC rules in 10 CFR 50.54(m)(2)(i) (Ref. 1) regulate reactor plant control room staffing. See Appendix A. The NRC also issued a "Policy Statement on Engineering Expertise on Shift," available at 50 FR 43621 (Ref. 2), which forms guiding principles relating to the qualification of the operating staff. Taken together, the regulations and Policy Statement determine the number of personnel required in the control room. The number of personnel in the control room on-shift must be multiplied by some factor to reflect total operating staffing. (For current operating plants this factor is between 10 and 20. It is anticipated that for smaller, simpler SMRs, this factor may be reduced.) Five shifts of personnel are typically provided to provide 24-hour coverage while accommodating needed time off and training time. In addition, each LO typically has at least one nonlicensed individual in a support role due to the generally practiced limitations on the collateral duties that LOs may be assigned.

In addition, NUREG-0800, Chapter 13 (Ref. 3), provides guidance on the section of an applicant's Safety Analysis Report (SAR) that describes the structure, functions, and responsibilities of the on-site organization established to operate and maintain the plant. NUREG-0800, therefore, also guides the operational staffing requirements of SMRs.

The NRC does, however, allow licensees to seek exemptions from regulatory requirements when warranted. See 10 CFR 50.12 (Ref. 4). Applicants or licensees may request exemptions from the staffing regulations in 10 CFR 50.54(m) and NRC guidance. NRC guidance document NUREG-1791 (Ref. 5) offers the staff guidance on exemption requests from power plant LO staffing requirements. (See also "Technical Basis for Regulatory Guidance for Assessing Exemptions Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," NUREG/CR-6838 (Feb. 2004) (Ref. 6).)

3.0 PROBLEM/ISSUE STATEMENT

NRC regulations and policies stipulate operator staffing requirements for licensed nuclear reactor facilities. These requirements are based on experience with the operation of the large, base-loaded reactors currently in use in the United States. These staffing requirements may not be appropriate or necessary for the new SMR designs, especially considering the simpler and more automated operation of these advanced designs. Additionally, excessive manning requirements need to be addressed early in the design review to avoid placing an undue economic burden on the operation of these SMRs, impacting the perceived viability of SMR vendors' business plans.

For example, using the staffing requirements in 10 CFR 50.54(m)(2)(i), a single-unit 10-MW(electric) Toshiba 4S reactor plant would be required to maintain four LOs per shift on-site. Four on-shift LOs translate into a combined operating staff of 40 to 80 personnel under current requirements. Considering the size and simplicity of the plant, and the minimal operator intervention necessary for either normal operation or accident response, this level of staffing is excessive.

Using 10 CFR 50.54(m)(2)(i) to determine the staffing requirements for a NuScale design plant with twelve modules, for example, is even more problematic, as the table (see Appendix A) does not consider a plant arrangement with greater than three units (reactors) or all the modules being operated from a single control room. Regardless, extrapolating the requirements of 10 CFR 50.54(m)(2)(i) to a twelve-module SMR facility would result in staffing numbers far in excess of those believed necessary to safely operate the reactor facility.

It should be noted that the SMR Special Committee is not suggesting that the level of qualification be reduced for the operators of SMRs. Ensuring the safe operation of smaller reactors will still require extensive training and testing for the operating staff, in line with existing NRC and Institute of Nuclear Power Operations (INPO) requirements. Experience from other industries shows that staffing can be reduced as automation and simplicity are increased. For example, airlines routinely operate with two-man flight deck crews when three-man crews for long-haul flights used to be the norm, reflecting the increased automation and reliability of flight controls. The U.S. Navy has significantly reduced the manning of the new Littoral Combat Ships (LCS) compared to previous frigate-sized warships. The LCS manning strategy includes reliance on "cross-rate" training—in other words, increasing the training of each operator to allow him or her to perform additional collateral duties. Experience in other industries shows that less manning usually is associated with increased training and experience of the operating staff.

Appendix B contains a discussion of selected SMR design features that reflect the simplicity and automation that can allow implementing the strategies to reduce operating staff manning described above.

4.0 DISCUSSION AND ACTUAL WORK

1. RISK PERSPECTIVE ON STAFFING

1.1. Justification for a Risk-Based Approach to Determine Staffing Size

A risk-based approach can be used to inform staffing requirements for SMRs. The risk-based approach could be used to establish that staffing requirements for a simple, Low Power Reactor (LPR) may be smaller than those for existing reactors.

It is expected that the SMR designs in development will have a much lower calculated probability of core damage and radioactive release than current-generation plants. This degree of risk reduction is consistent with the significantly improved risk profile due to the smaller core inventory, the vastly simpler design (fewer systems), and the inclusion of advanced design features such as passive safety systems.

The key differences between staffing for current power reactors and that proposed for staffing SMRs are in the areas of control room design, LO responsibilities, and control room staff organization. Specific proposals to address each of these areas are required to demonstrate the acceptability of the process in the concept of operation of each SMR design.

If current regulations were complied with, the number of LOs mandated in a multi-modular SMR plant control room would be greater than required for the control room of a current large operating unit of the GEN II/III/III+ design. However, a number of the proposed SMR concepts coming forward address a change in the responsibility for each reactor operator to monitor and provide control over more than one unit or module at a time. Thus, the number of operators per unit or module could be lower than the number of operators per unit listed in current regulations in 10 CFR 50.54(m) and NUREG-0800, Chapters 13.1.2 and 13.1.3.

The discussion that follows addresses some of the key features of SMRs that contribute to a reduced likelihood of core damage and release in comparison to the large, current-generation facilities. These features could be taken into account in supporting reduced staffing requirements for SMRs.

1.2. Accident Initiators

Potential accident initiators are grouped into two categories: "internal" events and "external" events. Internal event initiators include system failures such as loss of site power. External events include natural occurrences such as earthquakes and common mode failures such as fires. The potential remote location of an SMR facility introduces the possibility that some external events initiators may have a higher frequency than typically observed for LPRs. For example, external initiating events associated with extreme weather conditions might be more likely. Thus, the SMR design must compensate for potential increased initiator frequencies if a detailed Probabilistic Risk Assessment (PRA) demonstrates this to be the case.

In general, it is anticipated that the frequency of events that could lead to core damage in an SMR design is less than that for current-generation plants due to the simplicity of the design, the enhanced seismic protection (some designs), the reduced need for operator action, and the physical capability to passively accommodate heat removal functions from both the reactor and containment.

1.2.1. Internal Events

The spectrum of internal events typically considered as accident initiators for the current-generation light water reactor (LWR) includes anticipated transients during normal operation and the less likely postulated accidents such as a loss of reactor coolant. Transients may be associated with the reactor function (e.g., failure to scram) or with the power generation function (e.g., closure of steam stop valves). Some of these events have a reduced frequency or can be eliminated as accident initiators in SMR designs based on the plant's capability to cope with the event. While a design-specific PRA would identify initiators that are unique to that given design, and the associated frequencies of such initiators, general conclusions can also be made about the operating actions needed to respond to these events and conclusions drawn about the impact on manning. For example, if operator action is required in minutes rather than hours, the need for backup manning in the control room is clear. General guidelines on when the number of, timing of, and complexity of tasks require a second operator provides guidance to the designer and establishes clear goals for Human-System Interface (HSI) engineering.

1.2.2. External Events

The characteristics of potential remote sites introduce the possibility that certain external events may be the dominant accident initiators. In particular, earthquake risk is a dominant contributor in some Japanese reactors; several remote U.S. locations could introduce a similar situation. Additional external events that would be of particular concern for SMRs include the following:

- *Flood*: For some SMR designs the reactor is located underground, and groundwater intrusion or flooding of the buildings would be a design consideration.

- *External fire*: If the site includes wooded areas, an off-site forest fire could challenge plant operation.

- *Extreme cold*: Temperatures of -60°F and below represent unique challenges to equipment. A reactor trip under extreme cold conditions could challenge plant equipment until auxiliary power is available to provide heat (e.g., a long station blackout coping period).

- *Extreme snow and/or ice*: Extreme snow and/or ice conditions could prevent access to the plant.

- *Volcanic ash conditions*: Volcanic ash could affect machinery and limit access to the plant.

Although formal demonstration in a risk assessment would be required, it is expected that the safety design of some SMRs could accommodate these challenges because of the capability to provide core cooling with natural circulation in the absence of off-site power and without operator intervention.

1.3. Probability and Consequences of Containment Failure

Except for SMR designs that do not require containment, maintaining the integrity of the containment function remains an important NRC regulatory requirement, regardless of reactor design. Accordingly, there is a need to demonstrate the containment effectiveness as a radionuclide barrier; a typical means of doing so is to evaluate the Conditional Containment Failure Probability (CCFP). The CCFP illustrates the probability of a release given core damage.

SMR designs may use various methods to mitigate events that challenge the containment and reduce the potential for containment failure. Some examples include the use of double and/or low enthalpy containments [Light Water Reactor (LWR) designs] or coolant systems operating at atmospheric pressure in sodium-cooled fast reactor designs.

Containment bypass conditions are also less likely in an SMR than in current-generation LWRs because there are fewer active systems (thus fewer penetrations).

A reduced potential for containment failure supports the suggested reduction in staffing requirements. The severity of the accident consequences does not justify staffing at the level for existing large reactors.

1.4. Timing of Releases

The time of potential releases should be determined to establish the range of required emergency response actions and their impact on staffing decisions. Current advanced designs for large power reactors demonstrate that releases will not occur for at least 24 hours without operator intervention or active safety systems. For the SMR designs, for comparison purposes, it should be possible to demonstrate a longer release time. Analyses performed for the Power Reactor Innovative Small Module (PRISM) design indicated that for all but the most energetic release categories, the time to guard vessel/containment dome failure exceeds 24 hours.

Given the lower power level associated with the SMR designs, and the other design features discussed above, it is anticipated that credible release scenarios would require an even longer time for releases to occur. Adequate time will be available to supplement the initial on-site staffing if necessary in the case of a potential release.

2. CHANGES IN ROLE OF THE LICENSED OPERATOR AND OPERATIONS STAFF

A number of the SMR concepts moving forward in detailed design and in NRC preapplication licensing includes multi-modular designs where modules may be grouped so that one Reactor Operator (RO) can monitor and control multiple modules from a single control station within the main control room (MCR). This is a key difference between staffing for current power reactors and that proposed for some SMR designs in the area of control room design, LO responsibility, and control room staff organization. In these cases, the number of LOs in both the RO and Senior Reactor Operator (SRO) classification is expected to change based on the submittal and NRC acceptance of an exemption request per design, to the current regulations in 10 CFR 50.54(m) and the guidance in NUREG-0800, Chapters 13.1.2 and 13.1.3.

In the multi-modular SMR designs, the role of the operator does not change. The LO maintains responsibility for plant safety by selecting operating state, monitoring and verifying parameters, and initiating manual trip of a module, if trends indicate that auto trip is imminent. The HSI provides the operator with the information required to monitor and control multiple modules during an event. Automation reduces the burden on the operator by performing routine tasks including some tasks performed manually on current reactors.

This is made possible by modern digital controls and the use of proven modern digital technology to perform automated control functions, within the framework of the simple and passive SMR designs. With this technological capability, and the small, simple, and passive SMR design, the workload for traditional operator tasks is expected to be significantly reduced. This allows time for more in-depth monitoring of systems, structures, and components using automated data collection to support tasks such as trending, system evaluation, and planning for corrective actions. The operator can take on additional collateral duties without impacting the timely and effective performance of his or her safety function.

3. STAFFING REQUIREMENTS FOR EMERGENCY RESPONSE

Emergency response considerations for SMRs are the subject of a separate white paper being provided by ANS. Staffing aspects for emergency response are briefly treated here.

SMRs can be designed to function without operator intervention during normal, accident, and postaccident conditions. The passive safety design of the plant places fewer requirements on the staff when dealing with emergencies. Abnormal and emergency plant procedures are expected to minimize the required immediate actions. The required actions would largely be in the nature of monitoring the plant's condition, which can be accomplished by a small staff. Remote-monitoring capabilities are inherent in digital controls reducing, if not eliminating, many of the reporting responsibilities of the on-site operators in an emergency. Once an input or a measured parameter is converted to a digital signal, no significant information loss or degradation occurs regardless of the distance the digital information is transmitted.

The physical layout and reduced size of an SMR plant also contribute to making management of an emergency simpler. The smallest SMRs will occupy less than one acre with perhaps three acres of land needed to support plant activities. Limited radiological controls are required during normal or accident conditions.

The time interval of greatest activity for the licensed ROs is the period immediately after an accident/transient or other plant event. The responsibility of the LO(s) is to establish that the plant is performing within its specified safety limits and is achieving a known safe state in accordance with the plant emergency procedures. The emergency procedures identify the actions that need to be taken in a given plant condition. For events where there is no security risk, the guard staff can also provide predefined administrative, communications, and planning help such as making initial notification of government agencies, calling up the duty roster, or calling for fire or medical support.

4. OTHER RELEVANT CONSIDERATIONS

Under 10 CFR 50.12, the NRC may grant NPP licensees an exemption from otherwise applicable regulatory requirements upon determining that (1) the requested exemption is "authorized by law, will not present an undue risk to public health and safety, and [is] consistent with the common defense and security" and (2) "special circumstances are present" that warrant the granting of the exemption. The regulation identifies the "special circumstances" or justifications for which an exemption may be granted.

If requesting an exemption for staffing requirements were to become necessary or advisable, the basis for seeking it could be the provisions of 10 CFR 50.12(a)(2)(ii), which authorize an exemption where no undue risk to public health and safety is otherwise presented upon showing that application of the regulation "is not necessary to achieve the underlying purpose of the rule."

Any requests for exemptions from the requirements of 10 CFR 50.54(m) concerning the number of licensed personnel should be justified and reviewed using the NRC's "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)" (NUREG-1791).

NUREG-1791 provides the guidance necessary for submittal of the exemption request and all required task analysis necessary to justify the exemption. The task analysis steps include simulation capability to verify the capability of the human operators to manage multiple SMRs. The verification process identifies issues that need to be addressed in the design of the control room HSI to reduce the potential for human errors in the context of one RO being responsible for monitoring multiple SMRs, under the oversight of a control room supervisor/SRO and with assistance from other licensed staff present in the Main Control Room (MCR).

While licensees can pursue exemptions after the design is complete, the exemption process provides little up-front guidance to the designers. Tailored guidance for the designers is needed early in the process. As discussed above, the current NRC guidance does not extend to the number of units being co-located for some designs or would require an excessive number of operators for other designs. Designers will need to make assumptions about what will be appropriate deviations from NRC requirements. Regulatory certainty and transparency warrant the NRC engaging definitively early in the design process to ensure designers are not making overly aggressive assumptions that result in costly redesign during licensing.

5.0 CONCLUSIONS

As indicated in Section 4.0, this paper focuses primarily on staffing requirements necessary to support safe operation of the new generation of SMR designs. Evaluation of design and operation features for small and advanced reactors indicates that staffing requirements for the new SMR designs may be reduced in comparison to those applied for larger plants without compromising overall safety. The factors that contributed to this demonstrable potential for a reduced number of operating staff include the following:

- Inherent safety, reduced number of systems, and passive safety design require less operator intervention.

- Small source term compared to existing plants reduces the potential consequences of accidents.

- The small site can be monitored and maintained by fewer people.

- A greater proportion of the radioactive systems is contained within the containment structures, and health physics requirements are greatly reduced.

- Even when multiple modules of an SMR design are combined in one facility so as to have a cumulative capacity comparable to a large plant of the GEN III/III+ designs, the above factors suggest that the number of LOs may be less than would be currently required.

- Simplicity of operation allows for additional collateral duties for LOs without compromising essential safety functions.

While formal PRAs for the new SMR designs have yet to be issued, the calculated probability of a significant release and potential for off-site dose consequences can be expected to be lower than those for both advanced reactor designs and current-generation reactors. The reasons for this are the following:

- The simple, passive features should result in a lower calculated probability of core damage than current-generation plants.

- The capability of the containment structure and its passive nature cooling capability provide a reliable barrier to release for those designs that rely on containments.

- The radionuclide inventory is orders of magnitude less than that used in the current large reactors in use.

6.0 RECOMMENDATIONS

1. Updated regulatory guidance is needed. ANS will collaborate with the NRC to develop alternate staffing requirements for SMRs that result in the reduced operator staffing based on clearly identifiable criteria, and such approval will be obtained within the framework of existing regulations.

2. SMR applicants may pursue exemptions on a case-by-case basis. New regulatory guidance addressing staffing requirements for SMRs may not be available at the time of submittals for Design Certification or conditions of licenses. 10 CFR 50.12 allows seeking exemptions from regulatory requirements when warranted. SMR applicants should be prepared to ask for such exemptions in the staffing area if the need for them is identified after discussions with the NRC staff. Early discussions between representatives of SMR applicants and the NRC staff concerning

staffing should be held to determine, among other things, whether seeking such an exemption in one or more areas will be necessary.

7.0 REFERENCES

1. C*ode of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Sec. 50.54, "Conditions of Licenses," U.S. Nuclear Regulatory Commission.

2. *Federal Register*, 50 FR 43621, "Policy Statement on Engineering Expertise on Shift" (Oct. 1985).

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Rev. 6, U.S. Nuclear Regulatory Commission. (Sep. 2007).

4. C*ode of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Sec. 50.12, "Specific Exemptions," U.S. Nuclear Regulatory Commission.

5. NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)" (July 2005).

6. NUREG/CR-6838, "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," U.S. Nuclear Regulatory Commission (Feb. 2004).

APPENDIX A

Staffing Requirements Reproduced from 10 CFR 50.54(m)

(m)(1) A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

(2) Notwithstanding any other provisions of this section, by January 1, 1984, licensees of nuclear power units shall meet the following requirements:

(i) Each licensee shall meet the minimum LO staffing requirements in the following table:

Minimum Requirements[1] Per Shift for On-Site Staffing of Nuclear Power Units by Operators and Senior Operators Licensed Under 10 CFR Part 55

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

[1]Temporary deviations from the numbers required by this table shall be in accordance with criteria established in the unit's technical specifications.
[2]For the purpose of this table, a nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling as defined by the unit's technical specifications.
[3]The number of required licensed personnel when the operating nuclear power units is controlled from a common control room are two senior operators and four operators.

(ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

(iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a LO or senior operator shall be present at the controls at all times.

(iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

(3) Licensees who cannot meet the January 1, 1984 deadline must submit by October 1, 1983 a request for an extension to the Director of the Office of Nuclear Regulation and demonstrate good cause for the request.

APPENDIX B

Survey of Selected Key Design Features of SMR Designs with Implications for Staffing

The following discussion summarizes those features typical of SMRs that most directly affect the necessary staffing for safe operation. These features may or may not be present in each SMR design.

In general, SMRs are both significantly smaller and simpler than the reactors currently licensed by the NRC. The necessity for active operator participation is reduced for both normal steady-state operations and responding to transients and postulated accidents. The potential radiological consequences of any accidents are also orders of magnitude smaller than those of existing plants, due to the smaller source terms. This suggests that a smaller operating crew would be acceptable for normal monitoring and evolutions and for accident response.

B.1. ACCIDENT PREVENTION

B.1.1. Normal Operation

A desirable feature of a power generation source such as a reactor power plant is the ability to follow the system load, that is, to adapt the power output to meet moment-to-moment demand in the electric load it serves, in order to ensure that the power source is producing neither too little nor too much energy. Load-following is achieved in SMR design in various innovative ways.

One method may be by controlling the water flow to the steam generator, thus manipulating the core inlet temperature. As the generator output matches the load, changes in the coolant temperature introduce a positive or negative reactivity effect in the core, causing the reactor power to follow. The load-following capability simplifies operation of the power plant and reduces the likelihood of reactor trips. The ability to remain operating during significant load changes increases plant safety by avoiding the occurrence of off-normal events. The simplicity of such a design also reduces the need for online testing of safety systems. Online testing is itself a source of plant transient initiators.

SMR designs using liquid sodium as a coolant for the reactor permit operation at nearly atmospheric pressure with a large margin to the boiling point of the coolant (subcooling margin). Maintaining the core coolant subcooled provides assurance that the fuel cladding is not being overheated. The subcooling margin for these reactors is much greater than in an existing pressurized water reactor. Operation at atmospheric pressure eliminates the possibility of pressure transients.

B.1.2. Safety Systems

Safety systems for an SMR will include the systems used to shut down the reactor and those used to remove decay heat.

The safety systems of the SMR designs all include some version of a Reactor Shutdown System (RSS). The RSS in an SMR will be inherently simpler than that of the current generation of nuclear plants, primarily due to the smaller size of the reactors. The RSS may be activated either by loss of power, by the neutron detection instrumentation, or by some other process parameter such as the core outlet temperature of the reactor vessel. When activated, the RSS causes the reactor to shut down. Should the RSS fail to be activated, the reactor power level would nonetheless drop if the design incorporates a negative power coefficient of reactivity, bringing the reactor to a shutdown state.

After the automatic shutdown, passive systems remove energy from the reactor and connected loops, respectively. These passive safety systems do not require power for valve movements to initiate them. These systems may rely on natural circulation of the process fluid and/or air and do not depend on operator action.

The inherent capability of these designs to remove decay heat through passive means avoids the need to resort to active systems to maintain the plant in a safe shutdown condition. Table B.1 illustrates the simplicity of the typical SMR safety systems by comparing them to those in current-generation NPPs. The improvement in plant safety of the SMR designs over conventional designs is illustrated by the fact that many or all of the systems/features upon which a current-generation reactor relies are not required to maintain plant safety in a typical SMR design. The SMR designs eliminate the need for these active systems and thus increase plant safety.

TABLE B.1
Comparison of Current-Generation Plant Safety Systems to Potential SMR Design

Current-Generation Safety-Related Systems	SMR Safety Systems
High-pressure injection system. Low-pressure injection system.	No active safety injection system required. Core cooling is maintained using passive systems.
Emergency sump and associated net positive suction head (NPSH) requirements for safety-related pumps.	No safety-related pumps for accident mitigation; therefore, no need for sumps and protection of their suction supply.
Emergency diesel generators.	Passive design does not require emergency alternating-current (ac) power to maintain core cooling. Core heat removed by heat transfer through vessel.
Active containment heat systems. Containment spray system.	None required because of passive heat rejection out of containment. Spray systems are not required to reduce steam pressure or to remove radioiodine from containment.
Emergency Core Cooling System (ECCS) initiation, Instrumentation and control (I&C) systems. Complex systems require significant amount of online testing that contributes to plant unreliability and challenges of safety systems with inadvertent initiations.	Simpler and/or passive safety systems require less testing and are not as prone to inadvertent initiation.
Emergency feedwater system, condensate storage tanks, and associated emergency cooling water supplies.	Ability to remove core heat without an emergency feedwater system is a significant safety enhancement.

B.1.3. Support Systems

Auxiliary or supporting systems can affect the reliability of safety systems. Use of passive systems in place of active systems improves reliability. In the typical SMR design, elimination of all active cooling systems from the reactor side and elimination of all emergency cooling systems from the reactor building result in greatly improved plant simplicity and reliability.

Radiated heat from the reactor vessel is removed by passive means. The conducted heat into the containment may also be removed by the natural air cooling from the surface of the containment. An integral nuclear steam supply system (NSSS) may use an immersed primary pump, so no motor or pump seal cooling is required. As the result, all active cooling systems may be eliminated. This is illustrated in Table B.2 below.

TABLE B.2
Comparison of Current-Generation Plant Safety Systems to Potential SMR Design

Current LWR Support Systems	SMR Support Systems
Reactor coolant pump seals. Leakage of seals has been a safety concern. Seal maintenance and replacement are costly and time-consuming.	Integral designs eliminate the need for seals.
Ultimate heat sink and associated interfacing systems. River and seawater systems are active systems, subject to loss of function from such causes as extreme weather conditions and bio-fouling.	SMR designs are passive and reject heat by conduction and convection. Heat rejection to an external water heat sink is not required.
Closed cooling water systems are required to support safety-related systems for heat removal of core and equipment heat.	No closed cooling water systems are required for safety-related systems.
Heating, Ventilating, and Air-Conditioning (HVAC). Required to function to support proper operation of safety-related systems.	The plant design minimizes or eliminates the need for safety-related room cooling eliminating both the HVAC system and associated closed water cooling systems.[a]

[a]S. Hattori and A. Minato, "Passive Safety Features In 4S Plant," *1993 Proceedings of the 2nd ASME/JSME Joint Conference Nuclear Engineering: Volume 1*, ASME.

B.2. THERMAL INERTIA

Many SMR designs have a higher thermal inertia than existing licensed designs. This results in fewer severe transients and reduced necessity for operator intervention.

Liquid sodium is a coolant with excellent heat absorption capacity, very high thermal conductivity, low operating pressure (basically atmospheric), and superb natural convection capability. Decay heat can be removed from the core by natural circulation of the primary coolant and discharged to a heat exchanger. Passive cooling can also be provided by natural air circulation around the exterior of the reactor vessel. The large heat capacity of liquid sodium provides a large heat sink for the core. The time to heat up the fluid is substantially longer than for water-cooled reactors, and the available time for responding to accidents is thus significantly increased.

High Temperature Gas Reactors (HTGRs) also exhibit a large thermal inertia of the reactor core, with a large temperature margin between the operation limit and the safe operation limit, and slow temperature variations during power changes in a maneuvering mode.

Small and medium LWRs can also benefit from higher thermal inertia in comparison to existing plants by including a larger reactor vessel relative to the core size, contributing to longer response times in transients and accidents.

B.3. CONTAINMENT

SMR designs reduce the level of challenge to containment vessels/buildings in relation to existing designs. Most LWR SMR designs make use of a primary-system-in-one-vessel approach. The entire primary system is totally contained within one American Society of Mechanical Engineers (ASME) III, Class 1 vessel. By definition, such a vessel is not assumed to fail catastrophically so Loss-Of-Coolant Accidents (LOCAs) are eliminated. However, it remains necessary to have a separate containment vessel to deal with combustible gas and secondary system failures that could lead to core damage. These containment systems can be smaller and less robust than large LWR containments because the range of possible events results in lower pressures and/or temperatures.

HTGR SMR designs have a very robust fuel design that cannot melt under any circumstances encountered in the core. They also use a compressed gas such as helium in the reactor, not subcooled water. This significantly reduces the potential internal challenges to the containment in the event of a leak. There is no event possible in an HTGR that results in the physical challenges to a conventional reactor containment building. Not only is the pressure in the containment lower after a LOCA, but also the resultant impact on the core does not lead to core damage or the accompanying release of fission products. However, there is a range of accidents that can lead to the generation of hydrogen in significant quantities. In addition, there are always small amounts of tramp fission products and activation products in the coolant. The cumulative effect of all these factors lessens the demands on the containment structure so that its cost and complexity are significantly reduced.

The containment for a Sodium-Cooled Fast Reactor (SFR) is typically composed of a steel vessel and may also include a nearly impenetrable outer concrete vault. The entire assembly can be installed underground. Pressurization of the SFR containment appears much less likely than in existing reactors because the reactor coolant system is operated at ambient pressure. The high boiling point of liquid sodium means that less energy is transferred to the containment vapor space if the reactor pressure boundary fails. Use of liquid sodium eliminates hydrogen generation due to water-cladding interaction. As a result, the containment volume can be small, which allows for effective passive cooling. These features mitigate potential releases of radioactive materials in the event of an accident.

APPLICABILITY OF THE NRC LIGHT WATER REACTOR LICENSING PROCESS TO SMRs

Lead Writer – Roger S. Reynolds
Assistant Writer – Richard Barrett

American Nuclear Society (ANS)

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1.0 INTRODUCTION

Since the first days of the nuclear era, researchers and designers have conceived and developed a wide variety of reactor concepts. A rich variety of theoretical analyses, experimental studies, and prototype reactors demonstrated that each of these approaches had its own set of advantages. The technology base included a range of coolant materials, moderators, fuel types, and system configurations. Nevertheless, as the nuclear power industry grew and evolved, plant designs focused on light water–cooled reactors of the boiling water reactor and Pressurized Water Reactor (PWR) type. Knowledge gained from operational experience was incorporated into each succeeding generation of reactors, and the plants became increasingly similar in their overall characteristics.

The regulations and regulatory guidance of the U.S. Nuclear Regulatory Commission (NRC) have evolved in response to these developments in plant design. In addition, the NRC continuously examines operating experience to identify opportunities for improvements in the regulatory process and enhancements to plant designs. As a result, the regulations and regulatory guidance of the NRC are focused largely on the current fleet of Light Water Reactor (LWR) designs.

Every new plant design presents its own set of challenges for the licensing process. This is true for single units seeking Operating Licenses (OLs) and standardized plants applying for Design Certifications (DCs). The NRC makes use of time-tested regulatory processes to promote effectiveness and efficiency in its licensing reviews. These processes are supported by a wealth of regulatory guidance in the form of Regulatory Guides (RGs), the Standard Review Plan (SRP), and the consensus codes and standards produced by Standards Development Organizations (SDOs). As a result, the schedule, cost, and complexity of a licensing review can be greatly affected by the degree to which a plant concept deviates from existing designs.

Much of the design of a nuclear plant is independent of the type of reactor proposed. For example, the mechanical, structural, electrical, and Instrumentation and Control (I&C) characteristics of all Nuclear Power Plants (NPPs) are similar, as are the treatment of quality assurance, the environmental qualification, and the design for site hazards. The most challenging regulatory issues for a new reactor design tend to center on core and reactor coolant design, materials applications, system configuration, accident analysis, and containment. In addition, the conduct of Probabilistic Risk Assessment (PRA) and severe accident analysis can present new challenges.

Small and Medium Sized Reactors (SMRs) of a light water design differ in important ways from each other and from the current fleet of operating reactors. These designs incorporate innovative approaches to achieve simplicity, improved operational performance, and enhanced safety. Gas-cooled and liquid metal–cooled reactors represent an even greater departure from current designs and consequently greater challenges to the application of current regulatory guidance.

Several of the most challenging issues have been identified and analyzed in recent years. The next section of this paper will discuss this history in some detail. If SMR licensing is to succeed, these issues must be resolved to the satisfaction of the NRC and the public.

On the other hand, SMRs present an opportunity to develop a new generation of power plants with enhanced safety performance. Many of the designs make use of passive safety systems with simpler components, fewer dependencies, and less stringent operation/maintenance requirements. Some designs incorporate inherent safety features such as higher thermal inertia. In some cases, fast-moving accidents such as Loss-Of-Coolant Accidents (LOCAs) have been eliminated, and transient response is more benign. Some designs present less of a challenge in the severe accident arena and have favorable source term characteristics. These differences can ease the burden on operating staff and create opportunities for more effective accident management and should therefore result in a more efficient licensing process than that used for current LWR designs.

Light water reactor requirements provide assurance of safety system quality, capability, reliability, and redundancy commensurate with the safety characteristics of current designs. To the extent that SMR designs incorporate passive safety features, enhanced safety margins, slower accident response, and improved severe accident performance, opportunities to simplify and streamline the regulatory process and requirements should be considered.

2.0 BACKGROUND

The fundamental issues for *non-LWRs* have been detailed in SECY-03-0047 (Ref. 1). The staff identified seven issues and made recommendations for each:

1. "How should the Commission's expectations for enhanced safety be implemented for future non-LWRs?"
2. "Should specific defense-in-depth attributes be defined for non-LWRs?"
3. "How should NRC requirements for future non-LWR plants relate to international codes and standards?"
4. "To what extent should a probabilistic approach be used to establish the plant licensing basis?"
5. "Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability?"
6. "Under what conditions, if any, can a plant be licensed without a pressure-retaining containment building?"
7. "Under what conditions, if any, can emergency planning zones be reduced, including a reduction to the site exclusion area boundary?"

In assessing the options and developing the recommendations on the seven issues, the following general guidelines were employed by the staff:

- "Keep the risk to the population around a nuclear power plant site consistent with the Commission's 1986 Reactor Safety Goal Policy (51 FR 28044)."
- "Choose a risk-informed and performance-based approach, wherever practical, consistent with the Commission's 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (60 FR 4 2622) and the March 11, 1999, White Paper on Risk-Informed and Performance Based Regulation."
- "Use a technology-neutral approach."
- "Use the Commission's four performance goals to assess the advantages and disadvantages of the options and to develop recommendations."
- "Consider previous Commission guidance on these issues."
- "Consider the practicality of the options and recommendations."

On June 26, 2003, the NRC approved the staff recommendations for issues 2, 4, 5, and 7. The NRC approved the staff's recommendation for issue 1 on implementation of the NRC's expectations for enhanced safety in future non-LWRs, with the exception of accounting for the integrated risk posed by multiple reactors at the same site.

The NRC disapproved the staff's recommendation for issue 6, related to the requirement for a pressure-retaining containment building, indicating that there was insufficient information for the NRC to prejudge the best options and make a decision on the viability of a confinement building. The staff was directed to develop performance requirements and criteria working closely with industry experts (e.g., designers, Electric Power Research Institute) and other stakeholders regarding options in this area, taking into account such features as core, fuel, and cooling systems design. Further, the staff was directed to pursue the development of functional performance standards and then submit options and recommendations to the NRC for this policy decision. These requirements have not yet been developed.

On August 20, 2004, the NRC published SECY-04-0157 (Ref. 2), which outlined the staff's proposed regulatory structure for new plant licensing and potentially new policy issues. The objective of the regulatory structure for new plant licensing is to provide a *technology-neutral approach* to enhance the effectiveness and efficiency of new plant licensing in the longer term (beyond the advanced designs currently in the preapplication stage). The staff is developing a regulatory structure with four major parts (as discussed in SECY-04-0157):

"(1) a technology-neutral risk-informed framework (to be documented in a NUREG report) that will provide guidance and criteria to the staff for the development of technology-neutral requirements

(2) the content for a set of technology-neutral risk-informed requirements that will be based on the guidance and criteria established in the technology-neutral framework NUREG

(3) a technology-specific framework (to be documented in a NUREG report) that will provide guidance and criteria for the staff on how to apply the technology-neutral framework and requirements on a technology-specific basis

(4) technology-specific RGs that will be derived from the implementation of the technology-specific framework and will provide guidance to licensees on how to apply the technology-neutral regulations on a technology-specific basis."

NUREG-1860 (Ref. 3) was published in December 2007 to establish the framework described in part (1). NUREG-1860 documents a "framework" that provides guidance to staff to develop a set of requirements that would serve as an alternative to 10 CFR 50 (Ref. 4) for licensing future NPPs. The framework does not represent a complete process since there are several policy and technical issues to be resolved. NUREG-1860 refers often to advanced reactor designs, which are interpreted in the document to be non-LWRs. There is no mention of light water designs beyond the current Generation III/III^{+}, such as NuScale and Babcock & Wilcox (B&W) mPower.

Developing the requirements must consider the applicability of each of the General Design Criteria (GDC) and other relevant requirements relative to the reactor design in question. For example, for liquid metal–cooled, pool-type reactors, i.e., Power Reactor Innovative Small Module (PRISM), the following requirements merit reconsideration:

- accident evaluation, GDC 4
- source term, GDC 60, as low as reasonably achievable (ALARA) [TID 14844 (TID 14844 replaced by SECY-92-127 (source term evaluation); NUREG-1465 (Ref. 5)]
- containment performance, GDC 16, 38, 39, 40, 41, 42, 43, 50–57
- emergency planning, (defense-in-depth philosophy)
- reactivity control system, GDC 26 (necessity for two independent systems)
- operator staffing and function (minimum staffing requirements)

- residual heat removal, GDC 34, safety-grade systems versus passive systems
- positive void coefficient of reactivity, GDC 11, (negative power coefficient).

The key documents used by the staff to complete a DC are 10 CFR 50, 10 CFR 52 (Ref. 6), and NUREG-0800 (SRP) (Ref. 7). The next step after DC, the combined OL, will involve the use of 10 CFR 51 (Ref. 8) and NUREG-1555 (Ref. 9). Currently, a combined OL application is submitted according to RG 1.206 (LWR edition) (Ref. 10) and Office Instruction NRO-REG-100 (Ref. 11). Similar documents do not yet exist for non-LWRs. Prior to operation, the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) process must be completed. Chapter 14 of NUREG-0800 (SRP) establishes the ITAAC, but guidance for completion of ITAAC is not yet complete for any reactor design.

The documents discussed above must each be further reviewed for applicability and consistency relative to specific SMR designs and the NRC's policy for enhanced safety margins for advanced designs. The results of the review will likely be different for the specific design being considered, e.g., light water, gas, or liquid metal–cooled; metal or oxide fuel matrices.

Additionally, multi-module SMR sites should be accommodated by the combined OL process. Prototype and manufacturing licensing processes and definitions will also require review relative to specific SMRs.

There have been design reviews in the past. In 1992 General Atomics (GA) issued a report to the U.S. Department of Energy (DOE), "NP-MHTGR, Assessment of the Applicability of NRC Regulatory Guides and Branch Technical Positions." This report gives the flavor of the assessment that must be made for a new design. The following quote from the report clearly states the issue with respect to RGs and the GDC:

> "Since the RGs and BTPs [Branch Technical Positions] were developed by the NRC primarily for light water-cooled reactors, many of them are not directly applicable to HTGRs. It is, therefore, necessary to conduct an assessment of them for applicability to the New Production Modular High Temperature Gas-Cooled Reactor (NPMHTGR). Some of the RGs and BTPs will be found to be directly applicable to the NP-MHTGR, while others will be found to be not applicable at all. In general, however, most will be found to be applicable with qualification. In these cases the intent or spirit of the RG or BTP is applicable, but revisions to the guidance are needed to make it technically meaningful in the context of gas-cooled reactor technology."

The issue relative to RGs and the GDC for non-LWR designs is even larger than characterized since the reviews that have been done do not take into account the requirements of 10 CFR 52, the advent of risk-informed and performance-based regulation, or the environmental reviews that are necessary for a combined OL application.

The NRC position relative to liquid metal–cooled reactors is well outlined in NUREG-1368 (Ref. 12), published in 1994. The extensive review concluded that "on the basis of the review performed, the staff, with the NRC Advisory Committee on Reactor Safeguard (ACRS) in agreement, concludes that no obvious impediments to licensing the PRISM design have been identified." These two preapplication reviews reveal the effort required on the part of the designer/vendor and the regulator in the future as SMR designs become more common.

3.0 PROBLEM/ISSUE STATEMENT

Much of the NRC regulatory guidance and technical requirements apply equally well to SMRs as to the current generation of large LWR plants. The challenge is to define a safe, credible, and efficient process to ensure that the SMR designs meet the basic requirement of the Atomic Energy Act, i.e., to provide reasonable assurance of adequate protection of the public health and safety. Realistically, consistent with NRC policy, it will also have to be shown that the level of protection provided by an SMR is at least equivalent to, or better than, what is provided by current designs.

4.0 DISCUSSION AND ACTUAL WORK

All SMR designs deviate to a greater or lesser extent from the standard large-LWR template upon which the current regulatory framework is focused. Many of the differences relate simply to the small scale of these designs. Others relate to significant differences in the configuration of systems, structures, and components that are important to safety. SMR designs that are not LWRs have the additional factor of deviating in fundamental design concept: different coolant, different moderator. These factors lead to more fundamental differences with the current regulatory framework, i.e., different neutronic and thermal-hydraulic responses and different design-basis accidents. As a result, the certification of non-LWR designs will require specific deviations on a case-by-case basis from the requirements and guidance that currently govern the licensing of nuclear power reactors in the United States.

In some cases, an SMR design characteristic will deviate from regulatory guidance but will not violate a binding requirement. These include deviations from the provisions of the RGs, NRC policy statements, or the SRP. In these cases, approval for the deviation can be obtained by making an acceptable technical argument. The technical challenge may be significant in some instances, but the approval process is straightforward.

By contrast, if a design feature violates the specific provisions of an NRC regulation, the approval process becomes more complicated. The following sections discuss the options for obtaining approval in these cases.

1. BACK FIT

Some aspects of non-LWR designs present hazards to safety that are not covered by current NRC requirements. For example, liquid metal–cooled plants have the possibility of sodium fires. Engineered features to deal with such cases can be incorporated into the design and into the licensing basis of the plant in two ways. The first is for the applicant to incorporate the features voluntarily into the DC documents. These can then be endorsed in the staff's Safety Evaluation Report (SER). The second option is for the NRC staff to mandate a remedy for the hazard through 10 CFR 50.109 (Ref. 13), the backfit process. In either case, the resulting design feature would then be incorporated into the design through the 10 CFR 52 rulemaking process.

2. EXEMPTIONS

In some respects, SMR design characteristics are favorable to safety and will tend to justify less stringent application of current requirements. For example, some non-LWR designs are not susceptible to LOCAs and will not require an emergency core cooling system (ECCS). In such cases, the applicant would request relief under 10 CFR 50.12 (Ref. 14) or 10 CFR 52.7 (Ref. 15), the exemption process. To apply this process, the applicant must demonstrate that the deviation does not represent an undue risk to the public health and safety and that it is needed because a "special circumstance" exists. One example of a special circumstance that might apply to SMRs is that "application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." At this time, the exemption process (10 CFR 50.12 or 10 CFR 52.7) is the only NRC change process that can be applied by an SMR applicant to gain approval for deviations from NRC requirements.

In the history of regulation in the United States, the exemption process has been successfully applied numerous times, both for the initial licensing of plants and during their operational periods. Exemptions for any given plant have been infrequent, particularly in recent years since changes to certain regulations have eliminated some of the more frequent occasions for exemptions. The NRC has sought to avoid the excessive use of 10 CFR 50.12.

Moreover, exemption applications have generally been for changes that were limited in scope. For example, a plant might be exempted from in-service inspection for a specific pipe or weld but not for large portions of the plant. Many of the past exemptions were for minor changes to the schedules for containment leakage testing. The justifications for these exemptions were relatively straightforward and uncomplicated. There was no need to contemplate secondary impacts from approval of the request.

For SMR plants of LWR design, the number and complexity of needed exemptions may be modest. However, the exemptions needed for certification of non-LWR designs will deviate from these patterns in two respects: they may be numerous, and they may be complex. This situation will create two difficulties for the licensing process: (1) the effort on the part of the NRC and the applicant to grant approval of these exemptions will be significant and (2) the public perception associated with the issuance of so many complex exemptions may be unduly negative.

While the exemption process probably can be used judiciously for SMRs of LWR design, the licensing of non-LWR designs may require a different approach. Any new approach will require a time-consuming change to the Code of Federal Regulations (CFR) and will not be available in the near term. Sections 3 and 4 discuss two options for long-term changes to the CFR to accommodate the licensing of SMRs of the non-LWR variety.

3. TECHNOLOGY-NEUTRAL FRAMEWORK: GENERAL-SAFETY-STANDARDS OPTION

As discussed above, proposals have been made for enactment of technology-neutral regulations to govern the licensing of designs other than large LWRs (NUREG-1860). These regulations might be less specific than the requirements currently found in 10 CFR 50. Examples of this type of requirement can be found in International Atomic Energy Agency (IAEA) Safety Series Number NS-R-1 (Ref. 16). In this option, the details will be relegated to guidance documents such as the RGs or SRP.

In addition, the issuance of technology-specific regulatory guidance for each of the major non-LWR design types has been proposed. This guidance would be at the same level of specificity as the current regulations but would be appropriate to the features that are typical of that design type. The combination of the technology-neutral requirements with the technology-specific guidance will eliminate the need for multiple, complex exemptions in the licensing of non-LWRs.

The disadvantage of this approach is the difficulty of making a technical change to an NRC regulation. There have been numerous examples in the past of technical changes that have taken many years to enact. For example, the proposal to remove hydrogen recombiners from the design basis of large dry PWR containments was first introduced in the regulatory arena in 1992. In spite of wide agreement that the recombiners were of little safety significance, the rule change did not receive final approval until 2003. If one relatively modest change can require that much time and attention, the enactment of a new regulatory framework is likely to be complex and time-consuming.

4. TECHNOLOGY-NEUTRAL FRAMEWORK: PARITY OPTION

As noted above, the only option currently available for gaining approval of a deviation from a binding requirement is the exemption process. In this option, the licensee is exempted from meeting a requirement based on a demonstration of low public risk and the presence of "special circumstances." The implication of granting an exemption is that the design feature is deficient in some way but is acceptable because the safety impact is minimal. The granting of numerous exemptions has the disadvantage of raising the question whether the combined result of these minimal effects might be significant.

SMR designs are not deficient; they are in fact inherently safe in many ways. The parity option allows an applicant to gain license approval by demonstrating the inherent safety qualities of the design. The essence of this option is to enact an NRC change process that justifies deviations from the current regulations based on an integrated analysis of the fundamental features of the plant. The acceptance criteria for approval under the new change process would require demonstrating that the design provides a level of protection of the public health and safety that is equivalent to or better than what is provided by compliance with the current regulations.

The approval of a non-LWR design would still be technically complex under this proposal. The advantage is that the complexity will be dealt with in the NRC review process, not in the rulemaking process. Many options exist for facilitating the treatment of technical complexity in the context of the review process. Two examples of processes that have proved effective to gain NRC staff approval, in principal, for a new approach to achieving compliance and assuring safety are the industry consensus submittal and the topical report process, described as follows:

- *Industry Consensus Standards*: Review of consensus standards or Nuclear Energy Institute (NEI) task reports provide SER conclusions that can be relied on in individual licensing decisions. These industry consensus reports have been used on issues like fire protection, quality assurance, emergency planning action levels, operator training, and other administrative procedures. The industry and NRC can work toward a review structure for common issues in DC submissions, much the same way the industry Design Centered Working Groups (DCWGs) resolve common issues in DC, Reference Combined Construction and Operating License Application (R-COLA), and Standard Combined Construction and Operating License Application (S-COLA) reviews. This avenue takes advantage of the vast resources available to the SDOs. However, the standards development process can be time-consuming.

- *Topical Reports*: For vendor-specific issues, vendors can submit topical reports, the goal being to resolve a specific licensing issue applicable to that vendor or to preserve vendor proprietary information that could not be protected in an industry consensus standard. When a topical report is approved by the staff, it represents staff approval of the use of that approach that can be relied on in the review of future regulatory submittals. This approach is likely to be quicker than the standards development process. However, the vendor will have to bear the entire resource burden.

Finally, for any process used, there should be transparency for the benefit of all stakeholders to understand the level of protection provided by the innovative designs. Early definitive decisions by the NRC aid transparency. The public benefits not only from knowing what the applicant proposed but also from knowing what the NRC conclusions are. Currently, the NRC is finalizing SERs for many of the designs undergoing DC and combined OL review on a chapter-by-chapter basis. Early decisions on individual chapters of the SER for designs enhance transparency rather than making the public wait years for the entire SER.

5.0 CONCLUSIONS

For SMRs of the LWR design, the exemption process is sufficient for licensing.

For non-LWR designs, a technology-neutral framework is needed. Two options for a technology-neutral framework are presented in this paper: the general-safety-standards option and the parity option. It may be impractical and expensive to pursue rulemaking to accommodate each of the areas in which SMRs differ from LWR designs. It may also be impractical to contemplate the issuance of numerous exemptions to approve the ways in which SMRs do not conform to current requirements. It would be more efficient to pursue rulemaking to implement a technology-neutral framework based on the parity change process, where the advantages of SMRs can be compared on an equal footing to those of current designs. In this way, decisions about compliance of SMRs with the regulations can be made in a balanced manner. This process would allow an innovative design to be approved by demonstrating "parity" with current plants, that is, protection equivalent to or greater than that provided by compliance with the current regulations. The change process proposed allows for the provision of a coherent safety case, i.e., a convincing demonstration that the design is safe enough.

6.0 RECOMMENDATIONS

1. The ANS President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) recommends implementing the 10 CFR 50.12 exemption process for special circumstances for DC of LWR-type SMRs.

2. The SMR Special Committee suggests presenting two technology-neutral framework options, as described in this paper, to the NRC and stakeholders: the general-safety-standards option and the parity option, for resolution and selection for a path forward for DC of non-LWR SMRs.

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PHYSICAL SECURITY FOR SMALL MODULAR REACTORS

Lead Writer – Mark S. Campagna
Assistant Writer – Walter Sawruk

American Nuclear Society (ANS)

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1.0 INTRODUCTION

This paper discusses the issues of security risk management and potential risk mitigation strategies associated with the deployment of small modular reactors in the United States and abroad. It is crucial that these issues be addressed early in the small modular reactor lifecycle to ensure superior and reliable security and safety performance and maximum economic efficiencies in the design, fabrication, installation, and lifetime operation of small modular reactors worldwide.

2.0 BACKGROUND

The worldwide deployment of peaceful nuclear technology is predicated on conformance with the Nuclear Non-Proliferation Treaty of 1972 (NPT) and with modern standards for physical protection. Under the NPT, countries have relinquished pursuit of nuclear weapons in exchange for access to commercial nuclear technology that could help them grow economically. Realistically, however, most nuclear technology has been beyond the capacity of the NPT developing countries to afford. Even if the capital cost of the plant is managed, the costs of the infrastructure and the operational complexity of most nuclear technology could remove it as a viable option for consideration by the nations who need it the most.

This paper examines the functional requirements for both small modular reactors for deployment in the United States and also for those planned for export to other world countries. To enable proper export of U.S. technology, the highest standards for such an important design/engineering/performance issue as International Atomic Energy Agency (IAEA) safeguards to detect diversion or undeclared production of nuclear materials, and physical protection to prevent theft and radiological sabotage, need to be considered early and thoroughly; this is especially applicable for small modular reactors since much of the assembly work may be done in a factory setting prior to shipment.

A new class of small modular reactors has been specifically designed to meet the electrical power, water, hydrogen, and heat needs of small and remote users/communities and of medium to large industrial applications. These reactors feature small size, a long refueling interval, and simplified operations, all of which assist in minimization of security threats. Sized in the 10- to 50-MW(electric) range (very small) and up to the 300-MW(electric) range (small to medium), these reactors utilize factory modularization for rapid site deployment and assembly of single or multiple reactor "modules," placing an even greater premium on design standardization.

With large (mostly light water) 1000-MW(electric)+ reactors limited to the two to three dozen heavily industrialized countries, it is evident that distributed power using small modular reactors could be a very feasible solution to addressing the energy needs of the remainder of the world's nations in both the short and long terms provided issues such as physical security can be successfully addressed.

Furthermore, to emphasize the importance of maintaining high U.S.-based standards, any Small and Medium Sized Reactor (SMR) Nuclear Power Plant (NPP) manufactured by licensee [e.g., via a U.S. Nuclear Regulatory Commission (NRC)–issued Manufacturing License (ML)] may not be exported unless the ultimate customer meets all U.S. legal and regulatory export requirements, including 10 CFR 110 (Ref. 1) and 10 CFR 810 (Ref. 2). An export license should be complementary to the ML in an integrated fashion and should address all Federal export control requirements, not only those of the NRC but also those of the U.S. Department of Energy (DOE), U.S. Department of Commerce, and U.S. Department of State. [NOTE: The ML topic is the subject of another paper for the American Nuclear Society (ANS) President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee): "Utilization of NRC Manufacturing License for Small Modular Reactors"].

3.0 PROBLEM/ISSUE STATEMENT

The extent and relevance of this issue is considerable for SMR-NPPs; this since the worldwide deployment of peaceful nuclear technology is predicated on conformance with the NPT. We must consider various U.N. Resolutions (e.g., 1540) and the impact of other international agreements (e.g., Bilateral 123 Nuclear Technology Agreements between the United States and other countries). "123" refers to Section 123 of the U.S. Atomic Energy Act of 1954, which provides the legal framework for peaceful nuclear energy commerce. The United States has more than 30 such agreements in place with key partner nations. It therefore becomes imperative that the issues of nuclear proliferation resistance and physical protection of SMRs be addressed prior to addressing other key concerns such as fuel, waste, and economic/legal/political-stakeholder issues.

Since SMRs are generally in the early stages of development, a significant opportunity exists to affect designs in a way that (1) minimizes the future need for either substantial security forces, excess engineered devices, and/or complex procedural methodologies and (2) allows for the design optimization needed for more effective deployment of new applications. Early-stage design input can compensate in part for later possible design vulnerabilities against intentional acts of sabotage or theft.

Therefore, IAEA safeguards and physical security of the SMR must be included in the early design phase in order for the SMR to be an economically feasible solution when built. It is imperative that any SMR design demonstrate proof of requisite high levels of safe survivability from all credible threats, including malevolent terrorism, theft, or aircraft impact. An approach such as the proliferation resistance and physical protection evaluation methodology developed for Generation IV (GEN-IV) nuclear energy systems (Ref. 3) offers an attractive framework for application to SMRs. Stakeholders must understand the risks (i.e., financial and functional); the actual level of threat and required protection must be carefully assessed and understood by the appropriate qualified engineers/designers during very early stages of design/engineering.

4.0 DISCUSSION AND ACTUAL WORK

The operational experience of the existing fleet of Light Water Reactors (LWRs) provides valuable data that can be utilized in the development of sound design decisions for the next wave of advanced reactors and in the establishment of a well-structured approach to providing an appropriate security posture for the large LWRs within the U.S. National Response Framework (Refs. 4, 5, and 6). For other advanced reactors, including SMRs, that are either non-LWR based (e.g., gas- or liquid metal–cooled reactors) or small LWRs, designs are sufficiently different in their safety and operating characteristics such that the means to address safeguards and security requirements should be carefully evaluated to take into consideration the different design characteristics in satisfying ultimate performance objectives.

SMR developers can benefit from the advantages of favorable characteristics such as (1) small (target) size, (2) greater use of inherent security characteristics and passive safety features, and (3) smaller fission product inventory on a per-reactor basis. Conversely, with modular reactors a larger number of reactors must be protected. The objectives outlined in this paper are intended to meet or exceed the revised design basis threat (DBT) and requirements for enhanced security features set forth by the NRC in the recently revised 10 CFR 50.150 (Ref. 7), 10 CFR 73.1 (Ref. 8), and 10 CFR 73.55 (Ref. 9) without diminishing either the safety simplicity or economic feasibility/opportunities of SMRs. Since security requirements will most likely increase over the lifetime operation of the plants, it would be prudent for SMR designers to consider costs versus benefits of incorporating some additional design margins or provisions in the conceptual phase to lessen the impact of future changes to the DBT.

In order to address security design issues that provide for design optimization and maximum economic feasibility, the following areas were considered:

- *use of modern tools*: evaluation of risk-informed and performance-based methods such as are already being utilized quite effectively in analogous physical security applications by the DOE and U.S. Department of Defense to explore design functional vulnerabilities to defined security threats unique to individual SMR designs (Ref. 10, 11, and 12). Collaboration with industry standards bodies to formulate consensus methods for utilizing processes for achieving this goal (Refs. 13 and 14)

- *planning conceptual advantage*: physical separation of active systems to the extent practical to avoid limiting localized consequences from security breaches or internal acts or external damage

- *consideration of remote/passive features*: maximization of inherent characteristics and passive features that do not depend on immediate or short-term operator actions to assure extra protection, therefore deriving beneficial time delay advantage when analyzing effects on nuclear Systems, Structures, and Components (SSCs) or mitigating the consequences of security-driven transients or accidents

- *identification of improvements to redundancy*: arrangement or design of multiple reactor modules such that no single security threat is capable of creating an unacceptable radiological response in more than one reactor unit at a time

- *minimization of reliance on personnel*: careful examination of dependencies on reactor operator actions during any security-induced transient or event to assure that plant and public safety requirements are achievable with desired small staffing levels, without unnecessary dependence on operator actions for the first 24 to 72 hours

- *evaluation of increased utilization of remote and automated technology*: allowance for reducing or eliminating internal security staff requirements for normal operations and maintenance conditions that do not add to the security posture

- *consideration of geo-location and other functional effects*: establishment of a standard approach to integrating security requirements within the evolving requirements for large industrial facilities where SMRs are used as a process heat source.

5.0 CONCLUSIONS

1. **In order to address security design issues that provide for design optimization and minimize operational staff requirements, the SMR physical security approach should include the following five basic objectives:**

   (1) **Rely on government response for SMR facilities with vital assets underground or otherwise well protected.** Shallow burial or a hardened structural design provides excellent protection against large explosive weapons and aircraft impact as well as an excellent means of enhancing security system effectiveness against sabotage. Application of the traditional multilayered defensive approach of detection, deterrence, delay, and defeat can be used effectively for physical protection of SMRs. Detection, deterrence, and delay concepts must be integrated into the early design phase of the facility in order to provide sufficient lead time for government response.

(2) **Plan for DBTs that will evolve over facility lifetime.** Significant increases in the DBT should be expected and planned for starting at the conceptual design phase so as to minimize impact on operations and overall facility configuration and design. For example, establish a perimeter with sufficient standoff for protection against explosive threats in excess of the DBT and incorporate line-of-sight barriers into the design for protection against standoff weapons.

A definitive DBT, including aircraft impact, is necessary at the outset of the conceptual design phase in order to fully realize the potential benefits of integrating design, security, and preparedness. Although aircraft impact is sometimes treated separately from other physical security threats, it differs only in the scale of potential consequences and likelihood of occurrence from a facility design viewpoint. Mitigation measures developed for protection against physical security threats are likely to also contribute toward mitigation of the effects of the aircraft impact threat (Ref. 13). NRC's policy issue information statement SECY-10-0034 (Ref. 15) conflicts with this desired industry approach in that it suggests SMR designers determine the DBT for NRC's review and acceptance and it also implies that the NRC may impose supplemental acceptance criteria for non-LWR designs for aircraft impact after initial NRC reviews.

(3) **Risk-informed licensing approach.** A risk-informed and performance-based licensing approach including physical security to the extent practical has the potential to provide a more balanced physical security system than the current prescriptive approach to defending against the DBT.

(4) **Design the facility with limited access points and multiple passive barriers.** A defense-in-depth approach incorporating multiple passive barriers and limited access points to vital areas at the conceptual design stage will enhance overall security system effectiveness. Passive safety systems that do not require routine access for surveillance and maintenance can be hardened to provide long passive delay times.

(5) **Security system technology.** Significant advances in security system technology and countermeasures will most likely occur over the facility lifetime. Plan for security system technology obsolescence during the conceptual design phase. Build in redundancy and separation of systems to allow for future system overhaul or replacement with minimal need for compensatory measures.

2. **SMR-NPPs will require finalized up-front plans for advanced physical security implementation methods.** The performance spectrum and tools available for 21st century NPP design, engineering, and operation will be used for SMR-NPPs. Advanced planning and conceptual engineering implements would include objectives outlined in the points discussed above. Definitive regulatory requirements will be necessary at the outset to minimize licensing process uncertainties and unnecessary overdesigns and redesigns. Imposition of new criteria during the licensing review process (Ref. 15) would be counterproductive.

3. **Successful SMR-NPP security performance outcomes will depend upon advanced measurement tools/techniques.** Analysis methods should be utilized that result in more accurate measurement of effects. Use of risk-informed analysis can be very accurately and comprehensively applied to the limited number of vital SSCs and the smaller geo-radiological footprint of an SMR-NPP; this could

result in a much higher expectation of more reliable accuracy to ensure proper deterministic outcome.

4. **SMRs by their very nature may require new rule or regulatory guidance**. Upon examination of the various technologies utilized in the design of SMRs, it is evident that there is a substantial difference between these new designs and existing NPP technology sufficient to justify a "bottoms-up" assessment of such important issues as physical protection and safety. Basic parameters such as operating pressure/temperature; fission product inventory; and type-nature of coolant, materials, and moderator would appear to require further detailed assessment.

6.0 RECOMMENDATIONS

1. A new NRC Regulatory Guide is needed to address specific design aspects of the SMR-NPPs and to provide guidance for physical security and IAEA safeguards to assist engineers/designers and SMR developers.

2. The SMR Special Committee recommends using and exploring the Design Centered Working Group (DCWG) approach to reflect proactive improvement of detail for like reactor designs.

3. The use of automation, remote plant operations, and remote security for SMRs utilizing information relative to similar use in both the government/military and civilian commercial operations should be evaluated for applicability to SMR designs.

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UTILIZATION OF NRC MANUFACTURING LICENSE FOR SMALL MODULAR REACTORS

Lead Writer – Mark S. Campagna
Assistant Writer – Richard A. Govers

American Nuclear Society (ANS)

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1.0 INTRODUCTION

If the governing U.S. Nuclear Regulatory Commission (NRC) regulations are revised, an NRC Manufacturing License (ML) under 10 C FR 52 (Ref. 1) may be considered as a viable option for licensing future small modular reactors. The purpose of this paper is to examine the challenges with the current licensing process and set forth an alternate, potentially streamlined approach that could result in significant licensing efficiencies for both the NRC as well the industry in certain situations. This paper will summarize issues such as ML interaction with Early Site Permits (ESPs), combined construction permits (CPs) and Operations Licenses (OLs). This paper will also discuss environmental reviews, transport, export control, and intellectual property (IP) protection. Finally, areas such as quality assurance; foreign ownership, control, and influence (FOCI); duration shelf life; transferability; hearings; Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC); and other factors that could affect the decision to pursue an ML are also documented in detail in this paper's tables and appendices.

In sum, this paper makes the following general recommendations:

1. The ML, with or without a standard design approval, should enable small modular reactor vendors to file one application that approves the reactor design, as alternate to a Design

Certification (DC), and permits manufacture and transport of the reactor to a licensed site anywhere in the U.S., with separate export licenses required for export overseas. (Tables 1, 2, and 3 illustrate and compare the attributes of an ML with a DCD. Appendices A and B provide additional detail. For the application of new GEN III NPPs, most if not all, applicants have utilized the DCD for the large reactors planned for construction within the United States. Increased interest in use of the ML is likely for those vendors who plan to export small modular reactors to other countries.)

2. All environmental, site, construction, operation, and export issues would be addressed in connection with applications submitted by the small modular reactor customer, which may reference in their respective applications a design approved for manufacture under the modified ML regulations. Alternately, the manufacturer can pursue such licensing on a common basis with transfer to qualified owners and operators.

2.0 BACKGROUND

A new class of small modular reactors has been specifically designed to meet the electrical power, water, hydrogen, and heat needs. In general, small modular reactors differ from current-generation Light Water Reactors (LWRs) in many ways: size, moderator, coolant, fuel design, projected operation parameters, etc. These new reactors feature longer refueling intervals and simplified operations. Sized in the 10- to 50-MW(electric) range (very small) and up to the 300-MW(electric) range (small to medium), these reactors are built through modularized factory production and designed for rapid site deployment and assembly. The anticipated fuel source is <20% ^{235}U uranium fuel with a nominal core life of 10+ years**.** Many small modular reactors have been designed to operate as multiunit integrated facilities with as many as 4 to 16 small modular reactors operating in unison.

Small modular reactors also differ commercially from the current generation of LWRs. Small modular reactors are factory built and may be fabricated entirely off-site. The fabricated reactors will be shipped to a site for installation, which may include locations overseas. As commercialization proceeds, small modular reactor vendors may intend to fabricate small modular reactors without advanced long-term orders for installation. As such, advanced site licensing with environmental reviews may not be viable.

At the onset of the small modular reactor market, a clear understanding of the licensing process is needed to assist small modular reactor manufacturers as they proceed with the design, engineering, and manufacture of small modular reactor systems, structures, and components. Past consideration/use of the ML provision was not common. The NRC's only experience with reviewing and issuing an ML occurred in the early 1980s (i.e., Offshore Power Systems' ML-1 for the Floating Nuclear Power Plant, issued in 1982). 10 CFR 52, Subpart F (Ref. 2), was not fully updated in 2007 when the NRC issued revisions to its streamlined power reactor licensing process including updating the DC and combined OL regulations to reflect lessons learned from initial licensing reviews. An ML could be a vital element of a small modular reactor vendor's technical/business plans and strategy in this endeavor. Successful development of the small modular reactor industry in the United States may turn on whether a clear licensing framework exists, perhaps utilizing the ML.

3.0 PROBLEM/ISSUE STATEMENT

Revisions to the NRC licensing process culminated in 2007 with capturing lessons learned in validating the licensing process under 10 CFR 52 using single-step licensing of the combined CP and OL, generally in conjunction with a DC. Similar validation and updating of the ML process under 10 CFR 52, Subpart F, has not been done. The ML process offers an alternative that can enhance the commercial viability of small modular reactor designs in some circumstances. Absent an ML, small modular reactor vendors do not have a clear method to license the manufacture of small modular reactors with formal NRC involvement. By thoroughly understanding the key elements of the ML, a small modular reactor vendor and related stakeholders will have clear information and a defined path for obtaining an ML if this is determined to be the most feasible path forward. This is a decision that should be made early in the program/project lifecycle.

The following key issues with the current ML licensing process under 10 CFR 52, Subpart F, should be addressed:

- The ML need not approve the design of the reactor and may rely on a separate application being submitted for a DC [10 CFR 52, Subpart B (Ref. 3)]) or Standard Design Approval (SDA) (10 CFR 52, Subpart D (Ref. 4)], Because either a DC or SDA can be referenced by a qualified applicant, it affords less control to the small modular reactor vendor and may offer less intellectual property protection as the public interest in the basis for a DC or SDA may be higher [10 CFR 2.390(b)(5) (Ref. 5)].

- ML licensees may only transport manufactured reactors to licensed sites with either a CP or combined OL [10 CFR 52.153(a) (Ref. 6)].

- The ML regulation does not explicitly reflect interaction with 10 CFR 110 (Ref. 7) or 10 CFR 810 (Ref. 8), export regulations to permit shipping to non-U.S. locations.

Appropriate modifications to 10 CFR 52, Subpart F, will clarify the licensing process and bolster the development of the small modular reactor industry. An ML can potentially enable vendors to set up facilities to manufacture and sell small modular reactors without the necessity/undue burden of final site characterization prior to manufacture. NRC approval for siting and construction will be the responsibility of the ultimate customer upon its submission of either an ESP, CP, and/or Combined Construction Operating License Application (COLA). Furthermore, an ML in lieu of a DC or SDA may provide additional control for a licensee compared to a SDA holder or DC applicant. In the case of fabrication of small modular reactors for overseas utilization, NRC's oversight will shift from siting and environmental issues to verifying ITAAC prior to export. Modifications to the ML may also address issues associated with export restrictions under 10 CFR 110 (requirements for export and import of nuclear equipment and material) and the U.S. Department of Energy's (DOE's) 10 CFR 810 (requirements for assistance to foreign atomic energy activities).

These issues are explored below in more detail.

4.0 DISCUSSION

The ML regulations under 10 CFR 52, Subpart F, currently evaluate (1) the final design of a manufactured reactor, (2) the organization and technical control to be exercised for designing and manufacturing the reactor, (3) the ITAAC to be used by the licensee in determining whether the reactor has been properly manufactured in accordance with NRC requirements and the ML, and (4) the possession (but not the transport to or use of a reactor plant site) of the manufactured reactor. The ML does not approve any specific location, building, or facility where the actual manufacture of the reactor may occur, and the NRC does not require the applicant to submit any information on these matters as part of its application.

Key provisions of the 10 CFR 52 Series (Ref. 1) outline the limitations of the ML as applied to U.S. nuclear power plant installation and limitations for export consideration:

1. *Design Approval—10 CFR 52.157 (Ref. 9) (Content of Applications; Technical Information)*: "The application must contain a final safety analysis report containing the information set forth below, with a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that the manufacturing conforms to the design and to reach a final conclusion on all safety questions associated with the design, permit the preparation of construction and installation specifications by an applicant who seeks to use the manufactured reactor, and permit the preparation of acceptance and inspection requirements by the NRC." As such, the streamlined ML process may include the following elements.

   An ML application may, but need not, reference a DC or an SDA. A DC allows any qualified vendors to supply the design [10 CFR 52.73(a) (Ref. 10)]. While the requirement for a combined OL applicant that references an SDA, but not a DC, is less explicit, it is reasonable to assume the qualification requirements are similar [see 10 CFR 52.79(a)(32) (Ref. 11)]. In contrast, only a licensee may manufacture the design under the ML.

   The required contents of the ML application and the required technical information described in 10 CFR 52.157 closely follow those required under 10 CFR 52.137 (Ref. 12) for SDAs and those under 10 CFR 52.47 (Ref. 13) for Standard Design Certification. (Table 1 shows how closely 10 CFR 52.157 and 10 CFR 52.47 track each other.)

   Any revisions to the existing ML regulation addressing license content should assure that the responsibility of addressing site-specific issues, including environmental issues, remain with the ultimate customer. 10 CFR 52.158(b) (Ref. 14) requires the ML application to contain a generic environmental report while 10 CFR 51.54 (Ref. 15) directs the applicant to include in the site-specific environmental report an analysis of severe accident mitigation design alternatives (SAMDAs). In lieu of requiring an environmental analysis as part of the ML, it may desirable to make such a review optional at the discretion of the vendor.

   If a SAMDA analysis is not performed as part of an ML review, it would be consideration as part of the Severe Accident Mitigation Alternatives (SAMA) analysis during site-specific review.

(SAMDA analysis is a subset of the SAMA analysis. If performed as part of a DC review, the conclusions from a SAMDA analysis are given preclusive effect in the site-specific SAMA analysis, assuming the site parameters are bounded by those assumed in the DC SAMDA analysis. See, for example, 10 CFR 52, Appendix D, Sec. VI.B.7 (Ref. 18).) Reactor vendors may find it acceptable to conduct the SAMDA analysis during the site-specific SAMA analysis. Any risk insights that would be derived from a SAMDA analysis would be derived from the Probabilistic Risk Assessment (PRA) required as part of the ML application. For a reactor vendor that expects there is a low accident risk presented by its small modular reactor, no design changes not already evaluated by the PRA are likely to be cost-effective. The SAMDA analysis part of the licensing of advanced reactors to date shows the insights from the PRA are adequate and no additional considerations come from the SAMDA analysis. Also, if an ML is being used for export, incorporating risk insights from the PRA is likely adequate. (For example, the latest draft International Atomic Energy Agency (IAEA) standard on reactor design incorporates consideration of severe accident alternatives into the safety review, not a separate environmental review. Safety of Nuclear Power Plants: Design, IAEA Safety Requirement NS-R-1, Sec. 5.31 (2000 review draft). An ML should be addressed by regulatory changes to make SAMDA analysis as part of the ML review optional, but the requirement for a PRA should remain unchanged.

2. *Transportation to Installations Within the United States*: 10 CFR 52.167(c)(1) (Ref. 16) (Issuance of Manufacturing License): "A holder of a manufacturing license may not transport or allow to be removed from the place of manufacture the manufactured reactor except to the site of a licensee with either a construction permit under part 50 of this chapter or a combined license under subpart C of this part."

   The CP or combined OL authorizes the construction of a nuclear power facility using the manufactured reactor(s). An approved ML should permit transport of the reactors within the United States to the final destination but does not permit installation at the final location or operation. As it stands, although procedures for transportation to a plant site must be submitted as part of the ML process per 10 CFR 52.157(26)(iv), the actual transportation to a plant site is prohibited under 10 CFR 52.153(a) and 10 CFR 52.167(c) unless there exists either a CP or a combined OL for the site.

   As discussed above, the responsibility for all site-specific construction or environmental issues should be borne by the ultimate customer. The ML application contents should also either include requirements for shipping or reference the appropriate sections of 10 CFR 71 (Ref. 17). This requirement will be very difficult to establish if small modular reactor vendors seek to transport reactors containing fuel. Shipping cask requirements are stringent and may handicap the proposed streamlined ML framework.

3. *Considering Small Modular Reactors to Sites Outside the United States*: What is not evident within the 10 CFR 52 Series (Ref. 1) is any mention of or determination regarding export to other world locations; therefore, some form of integrated application of the applicable 10 CFR 110 (Ref. 7) and 10 CFR 810 (Ref. 8) rules is needed in combination with the ML in order for a small modular reactor vendor to export U.S. technology and materiel properly.

Specifically, the ML does not authorize export. A reactor manufactured by a licensee may not be exported unless the ultimate customer meets all U.S. law and regulatory export requirements, including 10 CFR 110 and 10 CFR 810. However, export changes to 10 CFR 110 permitting a more streamlined export permit procedure in relation to equipment and materials approved under an ML may be appropriate. There could be a need for necessary and parallel exemptions to ensure continuity of first launch client sales, design/engineering, and delivery while the appropriate statutory rulemaking proceeds—this appears to be acceptable to the NRC.

4. *Intellectual Property Protection*: The issue of IP protection is of critical importance to all nuclear power plant (NPP) technology vendors. Each vendor will go to great lengths to protect its investment in its product by means of a combination of patents, trademarks, and retaining expertise in-house and via contractual agreements. Recent DOE/NRC work since the 1990s has resulted in a very serious risk/concern for most small modular reactor vendors regarding IP protection, as there are limitations on the exclusive use provisions after a Design Certification Document (DCD) is granted in the public domain. The NRC may grant access to proprietary information or trade secrets, if the public interest outweighs the economic value. For small modular reactor vendors where the IP in the small modular reactor design is most, if not almost all, of the company's value, the risk of NRC release of the proprietary information can be a serious impediment to such small modular reactor vendors raising equity capital.

   Relying on the NRC to apply current regulations in 10 CFR 2.390 to withhold proprietary information from public disclosure has historically proven adequate for protecting the interests of larger, more diversified, reactor vendors. Where the DC applicant or SDA holder is a smaller, specialized small modular reactor vendor, larger competing, diversified vendors may view demonstrating technical qualifications equivalent to the DC applicant or SDA holder is not a significant economic barrier. An important objective of this paper is the identification of methods to more effectively assure reliable small modular reactor development via more robust IP protection schemes [whether in 10 CFR 52 or 10 CFR 2 (Ref. 19)] for small modular reactor vendors or, as a minimum, to identify the pitfalls/limitations of the current scheme.

   The General Electric (GE) Advanced Boiling Water Reactor (ABWR) development evolution provides an example of the risk to IP associated with the use of the DCD. The following is an excerpt from the Ref. 20 discussion on the same topic: "The original General Electric design has been certified, but any plant built from that design would face the need for modifications . . . . [although] The certified design also includes exclusive intellectual property of GE Hitachi; [however] South Texas -3 and -4 would use a design in which Toshiba is replacing the GE Hitachi exclusives [via use of the Combined Operations License Application (COLA)] with its own features developed from Toshiba's ABWRs in Asia . . . . GE Hitachi and Toshiba have both notified the NRC that they will seek the renewal of the ABWR certification [in parallel]." The observation can be made, as a result of this type of DCD exploitation and uncertainty, that there is limited or no meaningful protection of a vendor's design using the DCD/rule-making venue against a large, diversified competitor.

   On the other hand, the ML would be controlled by the small modular reactor manufacturer as a licensee; therefore, the ML appears to offer desirable protections for the vendor's design though not the duration of DC that the DCD approach offers (fifteen-plus years under DCD versus an effective twelve years under ML. [Further details are shown in 10 CFR 52, Appendix A (Ref. 21); 10 CFR 52.173 (Ref. 22) versus 10 CFR 52.55 (Ref. 23); and in 10 CFR 52, Appendix B (Ref. 24).] Unfortunately, the effectively shorter duration of the ML could potentially increase product costs for more frequent license renewals and thus result in reducing profit margins.

This issue demonstrates the need for coordination with the ESP/COLA provisions currently being utilized by the large Generation (GEN) III NPPs in the United States; perhaps some new protocol is needed for small modular reactors. Combined OL applicants are permitted to reference the ML in the same manner in which SDCs are referenced, as stated in 10 CFR 52.73. In doing so, a customer's combined OL application referencing an ML-approved design must contain information sufficient to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the ESP.

5. *ML Interactions with Other NRC Licenses/Permits*: In order to streamline licensing, the ML should be paired with an OL. 10 CFR 50, Appendix N (Ref. 25) permits applicants to submit an application for license to operate nuclear power reactors of essentially the same design to be located at different sites. Therefore, the small modular reactor vendor may apply for both the ML as well as an OL. Small modular reactors designed and manufactured by one vendor are essentially the same design but may be located at different sites. Upon successful approval of these two licenses, the small modular reactor vendor would now be able to manufacture the small modular reactor and subsequently sell the reactor with transfer of an approved OL to a qualified owner and operator. Importantly, the ML and OL application process would be completed only once and would not need to be repeated for each reactor manufacture/sale. Prior to shipment to the ultimate customer, the vendor and customer should commence a license transfer under 10 CFR 50.80 (Ref. 26) to transfer the OL to the customer. This license transfer would close out any license conditions related to operational procedures. (It is likely not commercially reasonable for a reactor vendor to demonstrate operational qualifications. Such operational issues would be addressed by license conditions in the OL. Those license conditions could be resolved as part of an amendment proposed with the OL transfer application.) This approach removes the burden of each customer being required to apply for an OL. The license transfer standards [10 CFR 2, Subpart M (Ref. 27)] for approval have proven not to be burdensome and therefore present little administrative burden to slow the process.

As discussed above, the responsibility for all site-specific construction or environmental issues should be borne by the ultimate customer. The customer would need to obtain a CP that demonstrates the site falls within the site parameters assumed in the ML and its paired OL. Alternatively, the customer could pursue a combined OL under 10 CFR 52 and reference the vendor's approved ML, in which case a transfer of the OL from the vendor would not be necessary. Flexibility could be increased if the purchasers were allowed to resolve the site-specific issues with an ESP. Prior to the license transfer, the ultimate customer could prepare his or her site to accept and install the purchased reactor via an approved ESP [10 CFR 52, Subpart A (Ref. 28)] as well as a limited work authorization [10 CFR 52.27 (Ref. 29) and 10 CFR 50.10 (Ref. 30)].

5.0 CONCLUSIONS

1. **Within the United States, the ML currently does not authorize shipment of a small modular reactor-NPP to a site; further licenses such as a combined OL are needed for equipment shipments, construction, and final operation**. The application of ML versus DC is conditional/limited and dependent upon certain business drivers such as speed to market, but the ML could be utilized if the planning cycle allows for proper application and granting of a Standard Combined Operating License Application (S-COLA). Environmental review of an ML should be simple. Currently, a manufacturer can assemble a reactor in a factory with no new Federal environmental review. Similarly, the environmental review associated with an ML should be at the reactor vendor's option as the conclusions from PRA insights can be expected to preclude a SAMDA analysis from identifying any design changes. Therefore, it may be commercially reasonable to conduct a SAMDA as part of the site-specific SAMA analysis; in fact, it may not even be needed based on the low risk expected to be demonstrated by the PRA that could show severe accidents are too remote and speculative to warrant considering severe accidents in the environmental review of that small modular reactor design.

2. **Outside the United States, the ML appears to offer an excellent vehicle to enable proper and well-controlled export of U.S. technology and expertise**. This issue is of substantial importance to small modular reactor-NPP vendors who have business models that depend upon significant global sales/export. However, there must be extensive coordination with other U.S. export provisions to authorize proper delivery. For purposes of this discussion we assume that NPP shipments outside the United States will be allowed (with export permits) to a foreign site that may not have approval for NPP construction/operation. An export license should be able to be combined with an ML in a seamless fashion and cover all Federal export controls, not only from the NRC but also from the DOE, and the U.S. Departments of Commerce, Treasury, and State.

3. **Manufacturing licenses may provide superior IP protection—with a trade-off of shorter duration and need for further development of rigor/reliability**. This IP protection aspect could be very attractive when combined with other IP protections such as trademarks and patents to help ensure that widespread application of small modular reactors is economically feasible both within the United States and worldwide. An ML should allow a manufacturer higher assurance of maintaining control of its design. Unlike a DC or SDA, which can be referenced by any qualified applicant, a small modular reactor vendor with an ML will solely enjoy the benefits of referencing a standardized design with marginally greater competitive certainty. Much effort remains to properly define protocol and useful precedent that would provide adequate assurance/certainty to small modular reactor vendors and properly defend their business case if the ML were chosen versus the DCD.

6.0 RECOMMENDATIONS

1. **Additional regulatory guidance is needed**. The NRC should clarify the means to cooperate with the DOE and U.S. Departments of Commerce, Defense, Treasury, and State perhaps resulting in a new Regulatory Guide (RG) that combines the aspects of 10 CFR 110 (export of material) and (DOE) 10 CFR 810 (export of technology) rules, given the increased interest in exporting small modular reactor technology globally. A good interim solution would also be development of a related task force to develop a Nuclear Energy Institute (NEI)–style guideline that could more efficiently evolve into an NRC RG. This approach enables thorough review and update of priorities with respect to relevant documentation.

2. **Design Centered Working Group (DCWG)–style collaboration should examine relevant federal precedent(s).** A new American Nuclear Society (ANS)–sponsored DCWG should further investigate other analogous industries (e.g., aircraft and weapons) that manufacture and export high-technology equipment for precedents, in parallel with further study. The DCWG could also be an NEI Working Group. A DCWG would enhance visibility and establish accountability of NEI members, including utilities. The ANS can continue to plan or initially set up the DCWG, but there are limits as to what can be accomplished with volunteer support.

3. **Use DCWG Forum to optimize certification pathway(s).** A post-June 2010 ANS-sponsored Technical Working Group/DCWG should further examine the ML for small modular reactor designs. This evaluation should include a critical integrated review of viable alternate certification and approval pathways, including DCs, MLs, combined OLs, SDAs, CPs, and common (10 CFR 52, Appendix N) OLs.

4. **Consider a parallel certification path; near-term exemptions/waivers along with rule changes.** The NRC should establish a parallel path whereby near-term exemptions or waivers can be granted for small modular reactor lead-launch clients while in pursuit of rule change process improvements. The small modular reactor industry should pursue a petition for rulemaking to provide additional flexibility and certainty to the ML process for small modular reactor projects. Such revisions will enhance the effectiveness of standardization by better matching the ML process to the commercial/business needs of small modular reactors manufactured and assembled for delivery, essentially ready to use at a prepared site.

5. **Enhance ML IP protective features.** Because of the potential heavy risk associated with IP for small modular reactors, the ML process needs to be reexamined to ensure that no loopholes or weaknesses exist that would place the small modular reactors at undue deleterious risk during the development and implementation process. Furthermore, the use of a task force approach (as outlined in Section 6.1 above) could significantly enhance the effectiveness and outcome of the process.

7.0 REFERENCES

1. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

2. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart F, "Manufacturing Licenses," U.S. Nuclear Regulatory Commission.

3. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart B, "Standard Design Certifications,"U.S. Nuclear Regulatory Commission.

4. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart D, "[Reserved],"U.S. Nuclear Regulatory Commission.

5. *Code of Federal Regulations*, Title 10, "Energy,"Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," Sec. 2.390, "Public Inspections, Exemptions, Requests for Withholding," U.S. Nuclear Regulatory Commission.

6. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.153, "Relationship to Other Subparts," U.S. Nuclear Regulatory Commission.

7. *Code of Federal Regulations*, Title 10, "Energy," Part 110, "Export and Import of Nuclear Equipment and Material," U.S. Nuclear Regulatory Commission.

8. *Code of Federal Regulations*, Title 10, "Energy," Part 810, "Assistance to Foreign Atomic Energy Activities," U.S. Department of Energy.

9. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.157, "Contents of Applications; Technical Information in Final Safety Analysis Report," U.S. Nuclear Regulatory Commission.

10. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.73, "Relationship to Other Subparts," U.S. Nuclear Regulatory Commission.

11. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report," U.S. Nuclear Regulatory Commission.

12. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.137, "Contents of Applications: Technical Information," U.S. Nuclear Regulatory Commission.

13. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.47, "Contents of Applications; Technical Information," U.S. Nuclear Regulatory Commission.

14. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.158, "Contents of Application; Additional Technical Information," U.S. Nuclear Regulatory Commission.

15. C*ode of Federal Regulations*, Title 10, "Energy," Part 51 "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Sec. 51.54, "Environmental Report—Manufacturing License," Nuclear Regulatory Commission.

16. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.167, "Issuance of Manufacturing License," U.S. Nuclear Regulatory Commission.

17. C*ode of Federal Regulations*, Title 10, "Energy," Part 71, "Packaging and Transportation of Radioactive Material," U.S. Nuclear Regulatory Commission.

18. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Appendix D, "Design Certification Rule for the AP1000 Design," U.S. Nuclear Regulatory Commission.

19. C*ode of Federal Regulations*, Title 10, "Energy, "Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," U.S. Nuclear Regulatory Commission.

20. "Design Certification, ABWR," *Nuclear News*, p. 27 (Apr. 2010).

21. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Appendix A, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," U.S. Nuclear Regulatory Commission.

22. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.173, "Duration of Manufacturing License," U.S. Nuclear Regulatory Commission.

23. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.55, "Duration of Certification," U.S. Nuclear Regulatory Commission.

24. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Appendix B, "Design Certification Rule for the System 80+ Design," U.S. Nuclear Regulatory Commission.

25. C*ode of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix N, "Standardization of Nuclear Power Plant Designs: Permits to Construct and Licenses to Operate Nuclear Power Reactors of Identical Origin at Multiple Sites," U.S. Nuclear Regulatory Commission.

26. C*ode of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Sec. 50.80. "Transfer of Licenses," U.S. Nuclear Regulatory Commission.

27. C*ode of Federal Regulations*, Title 10, "Energy, "Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," Subpart M, "Proceedings for Hearings on License Transfer Applications," U.S. Nuclear Regulatory Commission

28. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart A, "Early Site Permits," U.S. Nuclear Regulatory Commission.

29. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.27, "Limited Work Authorization After Issuance of Early Site Permit," U.S. Nuclear Regulatory Commission.

30. C*ode of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Sec. 50.10, "License Required; Limited Work Authorization," U.S Nuclear Regulatory Commission.

Table 1 **Identical or Essentially Identical Submittal Requirements** **Comparison between Manufacturing License and Design Control Certification**		
Summary of Requirement in 10 CFR 52	**Subpart F: ML Paragraph Ref**	**Subpart B: DCD Paragraph Ref**
FSAR must include:	157	47(a)
Principal Design Criteria	157(a)	47(a)(3)(i)
Design Bases & relation to Principal Design Criteria	157(b)	47(a)(3)(ii)
Description & analysis of Structures, Systems, Components (SCCs):	157(c)	47(a)(2)
Intended use	157(c)(1)	47(a)(2)(i)
Extent generally accepted eng practices are applied	157(c)(2)	47(a)(2)(ii)
Extent reactor uses enhanced safety features	157(c)(3)	47(a)(2)(iii)
Safety features as barriers to radiological release:	157(d)	47(a)(2)(iv)
Evaluation at exclusionary area boundary	157(d)(1)	47(a)(2)(iv)(A)
Evaluation at outer boundary of Low Population Zone	157(d)(2)	47(a)(2)(iv)(B)
Kinds & quantities of radiological materials; means to control & limit	157(e)	47(a)(5)
Information to establish design complies with technical req'ts:	157(f)	
Analysis & evaluation of design & performance of SCCs	157(f)(1)	47(a)(4)
Fire protection	157(f)(2)	47(a)(18)
Pressurized thermal shock	157(f)(3)	47(a)(14)
Combustible gas control	157(f)(4)	47(a)(12)
Station blackout	157(f)(5)	47(a)(16)
Electrical equipment important to safety	157(f)(6)	47(a)(13)
Anticipated transients without scram	157(f)(7)	47(a)(15)
Criticality accidents	157(f)(8)	47(a)(17)
Information required by §20.1406 (minimize contamination)	157(f)(9)	47(a)(6)
Control over gaseous & Liquid radiological effluents	157(f)(11)	47(a)(10)
Three Mile Island requirements	157(f)(12)	47(a)(8)
Compliance with earthquake engineering criteria	157(f)(14)	47(a)(20)
Demonstrate new safety features by test, analysis or prototype	157(f)(15)	47(c)(2)
Technical qualification of applicant	157(f)(16)	47(a)(7)
Description of quality assurance program	157(f)(17)	47(a)(19)
Proposed technical specifications	157(f)(18)	47(a)(11)
Site parameters & analysis & evaluation of design	157(f)(19)	47(a)(1)
Interface requirements between reactor and rest of plant	157(f)(20)	47(a)(25)
Interface req'ts to be verified by inspections, tests, or analysis	157(f)(21)	47(a)(26)
Representative conceptual design for nuclear power facility	157(f)(22)	47(a)(24)
For LWRs, description & analysis vs. severe accidents	157(f)(23)	47(a)(23)
For modular designs, possible operation configurations	157(f)(25)	47(c)(3)
Resolutions of Unresolved Safety Issues & generic safety issues	157(f)(28)	47(a)(21)
How operating experience has been incorporated	157(f)(29)	47(a)(22)
For LWRs, evaluation of design vs. NRC Standard Review Plan	157(f)(30)	47(a)(9)
Design-specific probabilistic risk assessment	157(f)(31)	47(a)(27)
Aircraft impact assessment per §50.150	157(f)(32)	47(a)(28)
Inspections, tests, analyses, and acceptance criteria (ITAAC)	158(a)	47(b)(1)
Environmental report per §51.54 or §51.55	158(b)	47(b)(2)

Table 2
Other Identical or Essentially Identical Requirements
Comparison between Manufacturing License and Design Control Certification

Summary of Requirement in 10 CFR 52	Subpart F: ML Paragraph Ref	Subpart B: DCD Paragraph Ref
Standards for review of application	159	48
Referral to Advisory Committee on Reactor Safeguards (ACRS)	165	53
Issuance of manufacturing license / standard design certification:	167	54
Applicable standards are met, reasonable assurance for compliance	167(a)(1), (2)	54(a) (1), (3)
Applicant is technically qualified	167(a) (4)	54(a) (4)
ITAAC are necessary & sufficient	167(a) (5)	54(a) (5)
Issuance not inimical to common defense or public's health & safety	167(a) (6)	54(a) (6)
Findings per §51 subpart A have been made	167(a) (7)	54(a) (7)
Site parameters & design characteristics are specified	167(b)(3)	54(b)
Finality of License / Certification:	171	63
Limitations on NRC imposing new requirements	171(a)(1)	63(a)(1)
NRC imposed modifications apply to all reactors	171(a)(2)	63(a)(3)
Other applicant may request departure / exemption	171(b)(2)	63(b)(1)
Criteria for renewal	179	59

Table 3 Potentially Significant Differences in Requirements between Manufacturing License (ML) and Design Certification Document (DCD)		
Topic	**ML Requirement**	**DCD Requirement**
FILING OF APPLICATION		
Who may apply ML: §52.155(a) & §50.38	Citizen, national, or agent of a foreign country or corporation owned, controlled by a foreign corporation is ineligible to apply.	No restriction on foreign ownership or control
CONTENT OF APPLICATION		
Risk-informed evaluation of SCCs ML: §52.157(f)(13)	If applicant uses risk-informed treatment of SCCs per §50.69, must submit information per §50.69(b)(2)	Risk-informed treatment of SCCs not explicitly mentioned
Management Plan ML: §52.157(f)(26)	Detailed Management Plan must be submitted	Management Plan not explicitly mentioned
Shipping Procedures ML: §52.157(f)(26)(iv)	Procedures for shipping must be submitted	Shipping procedures not mentioned
ADMINISTRATIVE REVIEW OF APPLICATIONS		
Hearing Procedures ML: §52.163; DCD: §52.51	Hearing requirements reference 10 CFR part 2, subparts C, G, L and N	Hearing requirements reference 10 CFR part 2, subpart O
ISSUANCE OF LICENSE / CERTIFICATION		
Implementation of QA Program DCD: §52.54(a)(8)	QA Program implementation not explicitly mentioned	Certification states applicant has implemented QA program
Shipment of reactor components to non-licensed site ML: §52.167(c)(1)	Manufactured reactor may not be transported from place of manufacture except to a site with either construction permit or combined license	Multiple provisions allow shipment to but not installation at a yet-to-be licensed site
DURATION OF LICENSE / CERTIFICATION		
Limitation near expiration of duration ML: §52.173; DCD: §52.55(a)	Valid for not less than 5 years or more than 15 years. Reactor manufacture may not start within 3 years of expiration until license is renewed	Valid for 15 years. No limitation on starting manufacture before expiration
Limitation on completing manufacturing ML: §52.173; DCD: §52.55(b)	Manufacture of any uncompleted reactor must cease upon license expiration, unless renewal application has been docketed	Certification remains valid after expiration for any docketed COL or OL application
TRANSFER OF LICENSE		
Transfer of License or DCD ML: §52.175	Manufacturing license may be transferred per §50.80	No provision given for transfer of Certification
License Ownership NRC Workshop, 2/3/10	"license is owned by a single entity"	"Certification can be used by any qualified entity"
APPLICATION FOR RENEWAL		
Hearings during renewal process ML: §52.177; DCD §52.57(a)	Allowance for a hearing on the application for renewal is explicitly stated	Commission appears to have discretion to hold hearings

Appendix A
Exact Wording of Differences in Manufacturing License vs. Design Certification Document

Manufacturing License -- 10 CFR 52 Subpart F (§52.151 - §52.181)	Standard Design Certification--10 CFR 52 Subpart B (§52.41 - §52.63)
§52.151 Scope of Subpart This subpart sets out the requirements and procedures applicable to Commission issuance of a license authorizing manufacture of nuclear power reactors to be installed at sites not identified in the manufacturing license application.	§52.41 Scope of Subpart (a) This subpart sets forth the requirements and procedures applicable to Commission issuance of rules granting standard design certifications for nuclear power facilities separate from the filing of an application for a construction permit or combined license for such a facility. (b)(1) Any person may seek a standard design certification for an essentially complete nuclear power plant design which is an evolutionary change from light water reactor designs of plants which have been licensed and in commercial operation before April 18, 1989. (2) Any person may also seek a standard design certification for a nuclear power plant design which differs significantly from the light water reactor designs described in paragraph (b)(1) of this section or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions.
§52.155 Filing of Applications (a) Any person, except one excluded by 10 CFR 50.38, may file an application for a manufacturing license under this subpart with the Director of New Reactors or the Director of Nuclear Reactor Regulation, as appropriate. (b) The application must comply with the applicable filing requirements of §52.3 and 50.30 of this chapter. **Ref: § 50.38 Ineligibility of certain applicants.** Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to believe is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, shall be ineligible to apply for and obtain a license.	§52.45 Filing of Applications (a) An application for design certification may be filed notwithstanding the fact that an application for a construction permit, combined license, or manufacturing license for such a facility has not been filed. (b) The application must comply with the applicable filing requirements of §52.3 and §2.811 through 2.819 of this chapter.

§52.157 Content of Applications; Technical Information (c) A description and analysis of the structures, systems, and components of the reactor to be manufactured, with emphasis upon the materials of manufacture, performance requirements, the bases, with technical justification therefore, upon which the performance requirements have been established, and the evaluations required to show that safety functions will be accomplished.	§52.47 Content of Applications; Technical Information (a)(2) A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.
	§52.47 Content of Applications; Technical Information (a)(3)(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with an adequate margin for safety;
§52.157 Content of Applications; Technical Information (f)(13) If the applicant seeks to use risk-informed treatment of SSCs in accordance with § 50.69 of this chapter, the information required by § 50.69(b)(2) of this chapter	
§52.157 Content of Applications; Technical Information (f)(26) A description of the management plan for design and manufacturing activities, including: (i) The organizational and management structure singularly responsible for direction of design and manufacture of the reactor; (ii) Technical resources directed by the applicant, and the qualifications requirements; (iii) Details of the interaction of design and manufacture within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor, as applicable; (iv) Proposed procedures governing the preparation of the manufactured reactor for shipping to the site where it is to be operated, the conduct of shipping, and verifying the condition of the manufactured reactor upon receipt at the site; and (v) The degree of top level management oversight and technical control to be exercised by the applicant during design and manufacture, including the preparation and implementation of procedures necessary to guide the effort;	

§52.157 Content of Applications; Technical Information (f)(27) Necessary parameters to be used in developing plans for preoperational testing and initial operation	
	§52.47 Content of Applications; Technical Information (c) This paragraph applies, according to its provisions, to particular applications: (1) An application for certification of a nuclear power reactor design that is an evolutionary change from light-water reactor designs of plants that have been licensed and in commercial operation before April 18, 1989, must provide an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink; (2) An application for certification of a nuclear power reactor design that differs significantly from the light-water reactor designs described in paragraph (c)(1) of this section or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design except for site-specific elements such as the service water intake structure and the ultimate heat sink, and must meet the requirements of 10 CFR 50.43(e)
§52.163 Administrative Review of Applications A proceeding on a manufacturing license is subject to all applicable procedural requirements contained in 10 CFR part 2, including the requirements for docketing in § 2.101(a)(1) through (4) of this chapter, and the requirements for issuance of a notice of proposed action in §2.105 of this chapter, *provided, however*, that the designated sections may not be construed to require that the environmental report or draft or final environmental impact statement include an assessment of the benefits of constructing and/or operating the manufactured reactor or an evaluation of alternative energy sources. All hearings on manufacturing licenses are governed by the hearing procedures contained in 10 CFR part 2, subparts C, G, L, and N.	§52.51 Administrative Review of Applications (a) A standard design certification is a rule that will be issued in accordance with the provisions of subpart H of 10 CFR part 2, as supplemented by the provisions of this section. The Commission shall initiate the rulemaking after an application has been filed under §52.45 and shall specify the procedures to be used for the rulemaking. The notice of proposed rulemaking published in the **Federal Register** must provide an opportunity for the submission of comments on the proposed design certification rule. If, at the time a proposed design certification rule is published in the **Federal Register** under this paragraph (a), the Commission decides that a legislative hearing should be held, the information required by 10 CFR 2.1502(c) must be included in the **Federal Register** document for the proposed design certification. (b) Following the submission of comments on the proposed design certification rule, the Commission may, at its discretion, hold a legislative hearing under the procedures in subpart O of part 2 of this chapter. The Commission shall publish a

	document in the **Federal Register** of its decision to hold a legislative hearing. The document shall contain the information specified in paragraph (c) of this section, and specify whether the Commission or a presiding officer will conduct the legislative hearing. (c) Notwithstanding anything in 10 CFR 2.390 to the contrary, proprietary information will be protected in the same manner and to the same extent as proprietary information submitted in connection with applications for licenses, provided that the design certification shall be published in Chapter I of this title.
§52.167 Issuance of Manufacturing License (a) After completing any hearing under § 52.163, and receiving the report submitted by the ACRS, the Commission may issue a manufacturing license if the Commission finds that: (2) There is reasonable assurance that the reactor(s) will be manufactured, and can be transported, incorporated into a nuclear power plant, and operated in conformity with the manufacturing license, the provision of the Act, and the Commission's regulations; (3) The proposed reactor(s) can be incorporated into a nuclear power plant and operated at sites having characteristics that fall within the site parameters postulated for the design of the manufactured reactor(s) without undue risk to the health and safety of the public; (b) Each manufacturing license issued under this subpart shall specify: (1) Terms and conditions as the Commission deems necessary and appropriate; (2) Technical specifications for operation of the manufactured reactor, as the Commission deems necessary and appropriate; (3) Site parameters and design characteristics for the manufactured reactor; and (4) The interface requirements to be met by the site-specific elements of the facility, such as the service water intake structure and the ultimate heat sink, not within the scope of the manufactured reactor. (c)(1) A holder of a manufacturing license may not transport or allow to be removed from the place of manufacture the manufactured reactor except to the site of a licensee with either a construction permit under part 50 of this chapter or a combined license under subpart C of this part. The construction permit or	§52.54 Issuance of Standard Design Certification (a) After conducting a rulemaking proceeding under § 52.51 on an application for a standard design certification and receiving the report to be submitted by the Advisory Committee on Reactor Safeguards under § 52.53, the Commission may issue a standard design certification in the form of a rule for the design which is the subject of the application, if the Commission determines that: (2) Notifications, if any, to other agencies or bodies have been duly made; (3) There is reasonable assurance that the standard design conforms with the provisions of the Act, and the Commission's regulations; (8) The applicant has implemented the quality assurance program described or referenced in the safety analysis report. (b) The design certification rule must specify the site parameters, design characteristics, and any additional requirements and restrictions of the design certification rule. (c) After the Commission has adopted a final design certification rule, the applicant shall not permit any individual to have access to or any facility to possess restricted data or classified National Security Information until the individual and/or facility has been approved for access under the provisions of 10 CFR parts 25 and/or 95, as applicable.

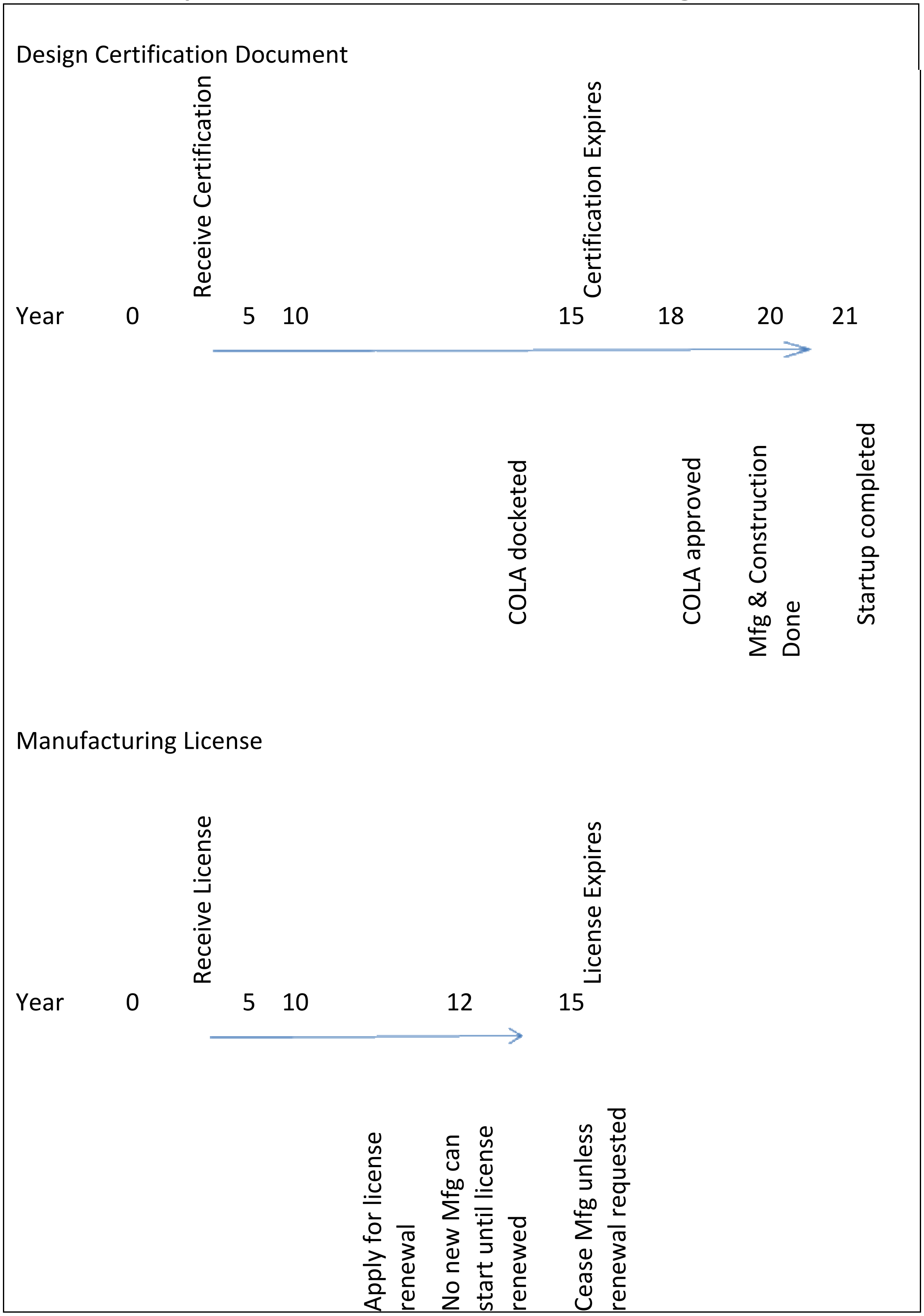
combined license must authorize the construction of a nuclear power facility using the manufactured reactor(s). (2) A holder of a manufacturing license shall include, in any contract governing the transport of a manufactured reactor from the place of manufacture to any other location, a provision requiring that the person or entity transporting the manufactured reactor to comply with all NRC-approved shipping requirements in the manufacturing license.	
§52.171 Finality of Manufacturing License (3) In making the findings required for issuance of a construction permit, operating license, combined license, in any hearing under § 52.103, or in any enforcement hearing other than one initiated by the Commission under paragraph (a)(1) of this section, for which a nuclear power reactor manufactured under this subpart is referenced or used, the Commission shall treat as resolved those matters resolved in the proceeding on the application for issuance or renewal of the manufacturing license, including the adequacy of design of the manufactured reactor, the costs and benefits of severe accident mitigation design alternatives, and the bases for not incorporating severe accident mitigation design alternatives into the design of the reactor to be manufactured. (b)(1) The holder of a manufacturing license may not make changes to the design of the nuclear power reactor authorized to be manufactured without prior Commission approval. The request for a change to the design must be in the form of an application for a license amendment, and must meet the requirements of 10 CFR 50.90 and 50.92.	§52.63 Finality of Standard Design Certification a)(1) Notwithstanding any provision in 10 CFR 50.109, while a standard design certification rule is in effect under §§ 52.55 or 52.61, the Commission may not modify, rescind, or impose new requirements on the certification information, whether on its own motion, or in response to a petition from any person, unless the Commission determines in a rulemaking that the change: (iv) Provides the detailed design information to be verified under those inspections, tests, analyses, and acceptance criteria (ITAAC) which are directed at certification information (*i.e.*, design acceptance criteria); (v) Is necessary to correct material errors in the certification information; (vi) Substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; or (vii) Contributes to increased standardization of the certification information. (2)(i) In a rulemaking under § 52.63(a)(1), except for § 52.63(a)(1)(ii), the Commission will give consideration to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration. (ii) The rulemaking procedures for changes under § 52.63(a)(1) must provide for notice and opportunity for public comment. (4) The Commission may not impose new requirements by plant-specific order on any part of the design of a specific plant referencing the design certification rule if that part was approved in the design certification while a design certification rule is

	in effect under § 52.55 or § 52.61, unless: (i) A modification is necessary to secure compliance with the Commission's regulations applicable and in effect at the time the certification was issued, or to assure adequate protection of the public health and safety or the common defense and security; and (ii) Special circumstances as defined in 10 CFR 52.7 are present. In addition to the factors listed in § 52.7, the Commission shall consider whether the special circumstances which § 52.7 requires to be present outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order. (2) Subject to § 50.59 of this chapter, a licensee who references a design certification rule may make departures from the design of the nuclear power facility, without prior Commission approval, unless the proposed departure involves a change to the design as described in the rule certifying the design. The licensee shall maintain records of all departures from the facility and these records must be maintained and available for audit until the date of termination of the license. (c) The Commission will require, before granting a construction permit, combined license, operating license, or manufacturing license which references a design certification rule, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determinations, including the determination that the application is consistent with the certification information. This information may be acquired by appropriate arrangements with the design certification applicant.
§52.171 Information Requests (c) Except for information requests seeking to verify compliance with the current licensing basis of either the manufacturing license or the manufactured reactor, information requests to the holder of a manufacturing license or an applicant or licensee using a manufactured reactor must be evaluated before issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the	

requested information. Each evaluation performed by the NRC staff must be in accordance with 10 CFR 50.54(f) and must be approved by the Executive Director for Operations or his or her designee before issuance of the request.	
§52.173 Duration of Manufacturing License A manufacturing license issued under this subpart may be valid for not less than 5, nor more than 15 years from the date of issuance. A holder of a manufacturing license may not initiate the manufacture of a reactor less than 3 years before the expiration of the license even though a timely application for renewal has been docketed with the NRC. Upon expiration of the manufacturing license, the manufacture of any uncompleted reactors must cease unless a timely application for renewal has been docketed with the NRC.	§52.55 Duration of Certification (a) Except as provided in paragraph (b) of this section, a standard design certification issued under this subpart is valid for 15 years from the date of issuance. (b) A standard design certification continues to be valid beyond the date of expiration in any proceeding on an application for a combined license or an operating license that references the standard design certification and is docketed either before the date of expiration of the certification, or, if a timely application for renewal of the certification has been filed, before the Commission has determined whether to renew the certification. A design certification also continues to be valid beyond the date of expiration in any hearing held under § 52.103 before operation begins under a combined license that references the design certification.
§52.175 Transfer of Manufacturing License A manufacturing license may be transferred in accordance with § 50.80 of this chapter.	*No provision for transfer of a DCD* *[§50.80 discusses transfer of licenses, not certifications]*
§52.177 Application for Renewal (a) Not less than 12 months, nor more than 5 years before the expiration of the manufacturing license, or any later renewal period, the holder of the manufacturing license may apply for a renewal of the license. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application. (b) The filing of an application for a renewed license must be in accordance with subpart A of 10 CFR part 2 and 10 CFR 52.3 and 50.30. (c) A manufacturing license, either original or renewed, for which a timely application for renewal has been filed, remains in effect until the Commission has made a final determination on the renewal application, *provided, however*, that in accordance with § 52.173, the holder of a manufacturing license may not	§52.57 Application for Renewal (a) Not less than 12 nor more than 36 months before the expiration of the initial 15-year period, or any later renewal period, any person may apply for renewal of the certification. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application. The Commission will require, before renewal of certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if this information is necessary for the Commission to make its safety determination. Notice and comment procedures must be used for a rulemaking proceeding on the application for renewal. The Commission, in its discretion, may require the use of additional procedures in individual renewal proceedings.

begin manufacture of a reactor less than 3 years before the expiration of the license. (d) Any person whose interest may be affected by renewal of the permit may request a hearing on the application for renewal. The request for a hearing must comply with 10 CFR 2.309. If a hearing is granted, notice of the hearing will be published in accordance with 10 CFR 2.104.	(b) A design certification, either original or renewed, for which a timely application for renewal has been filed remains in effect until the Commission has determined whether to renew the certification. If the certification is not renewed, it continues to be valid in certain proceedings, in accordance with the provisions of § 52.55.
§52.171 Finality of Manufacturing License (b)(1) The holder of a manufacturing license may not make changes to the design of the nuclear power reactor authorized to be manufactured without prior Commission approval. The request for a change to the design must be in the form of an application for a license amendment, and must meet the requirements of 10 CFR 50.90 and 50.92.	§52.59 Criteria for Renewal (c) In addition, the applicant for renewal may request an amendment to the design certification. The Commission shall grant the amendment request if it determines that the amendment will comply with the Atomic Energy Act and the Commission's regulations in effect at the time of renewal. If the amendment request entails such an extensive change to the design certification that an essentially new standard design is being pr*oposed*, an application for a design certification must be filed in accordance with this subpart. (d) Denial of renewal does not bar the applicant, or another applicant, from filing a new application for certification of the design, which proposes design changes that correct the deficiencies cited in the denial of the renewal.
§52.181 Duration of Renewal A renewed manufacturing license may be issued for a term of not less than 5, nor more than 15 years, plus any remaining years on the manufacturing license then in effect before renewal. The renewed license shall be subject to the requirements of §§ 52.171 and 52.175.	§52.61 Duration of Renewal Each renewal of certification for a standard design will be for not less than 10, nor more than 15 years.

Appendix B
Sample Timeline for DCD vs. Manufacturing License



NRC Annual Fees for Licensees

Lead Writer – Jose Reyes
Assistant Writer – Charles Hess

American Nuclear Society (ANS)

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1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) is required to collect 90% of its annual appropriated budget through two types of fees. One type is for NRC services such as licensing and inspection activities. The other is an annual fee paid by all licensees, which recovers generic regulatory expenses and other costs not recovered through fees for specific services. These fees are described in NRC regulations 10 CFR 170 (licensing and inspection services) (Ref. 1) and 10 CFR 171 (annual fees) (Ref. 2).

In accordance with 10 CFR 171.15 (Ref. 3), these fees apply to "each person holding an operating license for a power, test, or research reactor; each person holding a combined license under part 52 of this chapter after the Commission has made the finding under § 52.103(g); each person holding a part 50 or part 52 power reactor license that is in decommissioning or possession only status, except those that have no spent fuel onsite; and each person holding a part 72 license who does not hold a part 50 or part 52 license shall pay the annual fee for each license held at any time during the Federal fiscal year in which the fee is due." This paragraph does not apply to test and research reactors exempted under 10 CFR 171.11(a) (Ref. 4). The FY 2010 annual fee proposed for each operating power reactor is $4,719,000. This fee is regularly "re-baselined" by the NRC (Ref. 5).

On March 25, 2009, the NRC issued an Advance Notice of Proposed Rulemaking (ANPR) for public comment in *Federal Register*, Vol. 74, No. 56, Docket ID NRC-2008-0664 (Ref. 6). The NRC was seeking comment on a proposal to amend its rule governing annual fees to establish a variable annual fee structure for nuclear power reactors based on licensed power limits. Current regulations governing annual fees require that each operating nuclear power reactor pay the same annual fee, regardless of the size of the reactor. Numerous comments on the ANPR have been received from the public and considered in the development of the approach proposed herein.

The following sections provide key definitions, a discussion of issues associated with developing an annual fee structure that is equitable to all of the stakeholders, and a recommended approach.

2.0 DEFINITIONS

1. SMALL, MEDIUM, AND LARGE REACTORS

For purposes of establishing an annual fee structure, the licensed thermal power rating of the reactor shall be used to define reactor size. The thermal power is preferred over electric power to include nuclear plants that produce process heat for industrial applications. The following definitions are employed:

- small reactors: ≤1000 MW(thermal)
- medium reactors: 1000 to 2000 MW(thermal)
- large reactors: ≥2000 MW(thermal).

Therefore, Small and Medium Sized Reactors (SMRs) are defined as the class of reactor having a licensed thermal power rating <2000 MW(thermal). The definitions above are roughly equivalent to the International Atomic Energy Agency (IAEA) definitions defined in terms of electric power output. Low Power Reactors (LPRs) are defined as the subset of small reactors having a licensed thermal power rating of ≤250 MW(thermal). This excludes test and research reactors.

2. MULTI-MODULE NUCLEAR PLANT

A multi-module nuclear plant is defined as a facility (1) that houses multiple co-located nuclear reactors (modules), (2) that shares a common Final Safety Analysis Report (FSAR), and (3) where each reactor has a licensed power rating of ≤1000 MW(thermal). *The licensed thermal power rating for a multi-module nuclear plant is the sum of the licensed thermal power rating of each module in operation*.

3. FINAL SAFETY ANALYSIS REPORT

The FSAR is required by 10 CFR 50.34(b) (Ref. 7) to be included in each application for a license to operate a nuclear facility and includes a description of the facility; the design bases and limits on its operation; and a safety analysis of the Systems, Structures, and Components (SSCs) and of the facility as a whole. A multi-module nuclear plant will have one combined FSAR for the total configuration of all modules combined rather than a separate FSAR for each reactor or module.

3.0 ANALYSIS OF PUBLIC COMMENTS AND PROPOSED OPTIONS

The NRC has received numerous comments from the public in response to its ANPR on "Variable Annual Fee Structure for Power Reactors." Several approaches were presented, and some concern was raised regarding assuring an equitable fee structure. This section summarizes the types of approaches submitted through the NRC's public comment process and includes a discussion on the issues associated with developing an equitable annual fee structure for nuclear power plants. The issues include adequately reimbursing the NRC for the cost of oversight, avoiding a fee structure that unfairly penalizes the development and deployment of SMRs, and assuring that the existing fleet of nuclear plants does not unfairly bear the regulatory framework development costs associated with deploying SMRs. Three specific options that incorporate or address the public comments have been developed.

1. OPTION 1: NO RULE CHANGE AND REINSTATE 10 CFR 171.11(c)

One approach is to not change the existing rule until the NRC conducts a new study for SMRs similar to that conducted to establish the final fee rule 51 FR 33224 on September 18, 1986. The previous exercise included a review of inspection and licensing fees over a 1-year period that did not show a correlation between the power level of the nuclear units and the amount of effort expended by the NRC staff on those units. The conclusion of the study indicated that there was "no necessary relationship or predictive trend between thermal megawatt rating of a reactor and the NRC regulatory costs" (Ref. 8). Although there have been significant enhancements in plant performance over the past 24 years that have reduced regulatory burden, other changes such as those related to plant security have had a normalizing effect. Because of this, it is likely that a new study would not show significant differences in NRC regulatory costs for existing reactors above a certain thermal megawatt rating. However, reductions in regulatory burden are expected to accrue for SMRs and multi-module plants because such plants are being specifically designed with state-of-the-art technology. Furthermore, incremental additions to plant power generally do not bear the same regulatory burden as first installations. Since 1970, U.S. nuclear power plants have completed 127 power uprates totaling 5695 MW(electric) (Ref. 9). Utilities have opted to uprate their plant thermal power because it represents a low-cost means of increasing production without increasing the regulatory burden to the overall plant. A typical 5% power uprate on a 1000-MW(electric) plant represents a 50-MW(electric) incremental change, which is the total power output of some SMRs. Similarly, the incremental addition of power modules at an existing site would not bear the same regulatory burden as the installation of the first module. As such, the fees for module additions should also be incremental.

It is recognized that the current annual fee structure provides stability in the budgeting process for the existing fleet of commercial nuclear power plants. Relief in fees for SMRs could be obtained by re-establishing the provisions of 10 CFR 171.11 (c) to allow consideration of reactor size via an exemption request. This provision was eliminated in FY 2005. As an example, Big Rock Point was able to request partial exemption from annual fees using this provision.

This option presents a serious challenge to SMR investors and potential customers who are currently making decisions of significant financial consequence. The "No Rule Change" approach introduces significant uncertainty into the investor/customer decision-making process because it introduces the possibility that SMRs may face unreasonably high annual fees on a reactor basis, particularly for multi-module plants. Similarly, reliance on fee exemption requests on a case-by-case basis produces significant regulatory costs to establish fees and introduces uncertainty in the annual cost of operating the plant. The situation is somewhat similar to the position that the current commercial nuclear fleet was facing prior to the final fee rule.

2. OPTION 2: A SAFETY-BASED ANNUAL FEE STRUCTURE

Because of the significant reduction in risk being postulated by Probabilistic Risk Assessment (PRA) studies conducted for the next generation of nuclear plants, including SMRs, another possible approach was to consider an annual fee structure based on a combination of rated thermal power and safety measures such as core damage frequency. The premise is that nuclear plants with reduced risk of core damage and a smaller source term represent a lesser risk and hence a commensurate reduction in oversight costs for the NRC. In this model, the core thermal power would serve as a measure of the size of the fission product source term. This approach would be applicable to all nuclear power reactors, thereby offering the potential for fee reductions to plants with lower risk factors. It also encourages a risk-based approach to nuclear power.

The primary difficulty with this approach is that it introduces a new level of complexity in establishing an annual fee structure. It introduces the question of PRA uncertainty that may encourage significant analysis efforts, NRC review, and costs related to reducing such uncertainties. This approach does not recognize that the entire commercial fleet currently meets the NRC's safety goal for core damage frequency and that operation over the past 30 years has demonstrated that nuclear plants are safe.

This option could be augmented to address the issue of equity in sharing the regulatory oversight costs by establishing a minimum annual fee that would cover NRC costs associated with the oversight of any nuclear power plant regardless of size.

This option could also be augmented to recognize the reduced regulatory burden to NRC for plants with excellent "plant health." Reduced annual fees could be considered for plants with no significant open findings.

3. OPTION 3: FEE LIMITS AND SLIDING SCALE FOR SMRS AND MULTI-MODULE PLANTS

Another option is to assess a minimum annual fee from all nuclear power plants to cover generic costs associated with the regulatory oversight. In addition, a sliding scale, based on thermal power, would be implemented for reactors with a total licensed thermal power rating ≤2000 MW(thermal). For a multi-module plant, the sliding scale would be based on the sum of the licensed thermal power rating of each module. Lastly, the annual licensing fees will be capped for all nuclear power plants above 2000 MW(thermal). It is proposed that annual fee credits be provided to plants based on plant performance that reduces regulatory burden.

3.1 Minimum Annual Fee

It is proposed that a minimum annual fee, or a "base fee," be applied to all nuclear plants regardless of size to cover NRC generic costs associated with, but not limited to

- rulemaking activities
- regulatory guidance development
- operating experience review
- incident response center operation
- emergency planning and drills.

It is proposed that LPRs [nuclear power plants with a licensed thermal power rating ≤250 MW(thermal)] serve as the basis for establishing the minimum annual fee. The assumption is that LPRs represent the minimum regulatory burden among power-producing nuclear plants. This assumption is supported by the NRC's assessment of reduced fees for two LPRs: LaCrosse and Big Rock Point. With regard to thermal power and radioactive material inventory, an LPR would fall between a test/research reactor and a spent fuel storage facility. A test/research reactor has low thermal power, has a small radioactive material inventory, and is assessed an annual fee of $81,800 (Ref. 5). A spent fuel storage facility has zero power, has significant quantities of radioactive material, and is assessed an annual fee of $143,000 (Ref. 5). Therefore, a reasonable basis for a minimum annual fee would be approximately $110,000, which falls between the two categories already established by the NRC.

3.2 SMR Sliding Scale

It is proposed that the annual fee for SMRs and multi-module nuclear plants be scaled relative to the licensed thermal power rating for the plant. For a multi-module plant, the licensed thermal power rating is the sum of the licensed thermal power rating of each module in operation.

The annual fee would be determined by the following formula for plants with a total licensed thermal power rating ≥250 and ≤2000 MW(thermal):

Annual Fee = Minimum Annual Fee + [Power — 250][MW(thermal)] x Fee Rate [$/MW(thermal)].

The sliding scale provides a linear interpolation between the minimum and maximum annual fees. The resulting fee rate (i.e., slope) using this approach is $2.63/kW(thermal) based on the NRC's FY 2010 proposed fees. The sliding scale is not proposed for test and research reactors.

3.3 Maximum Annual Fee

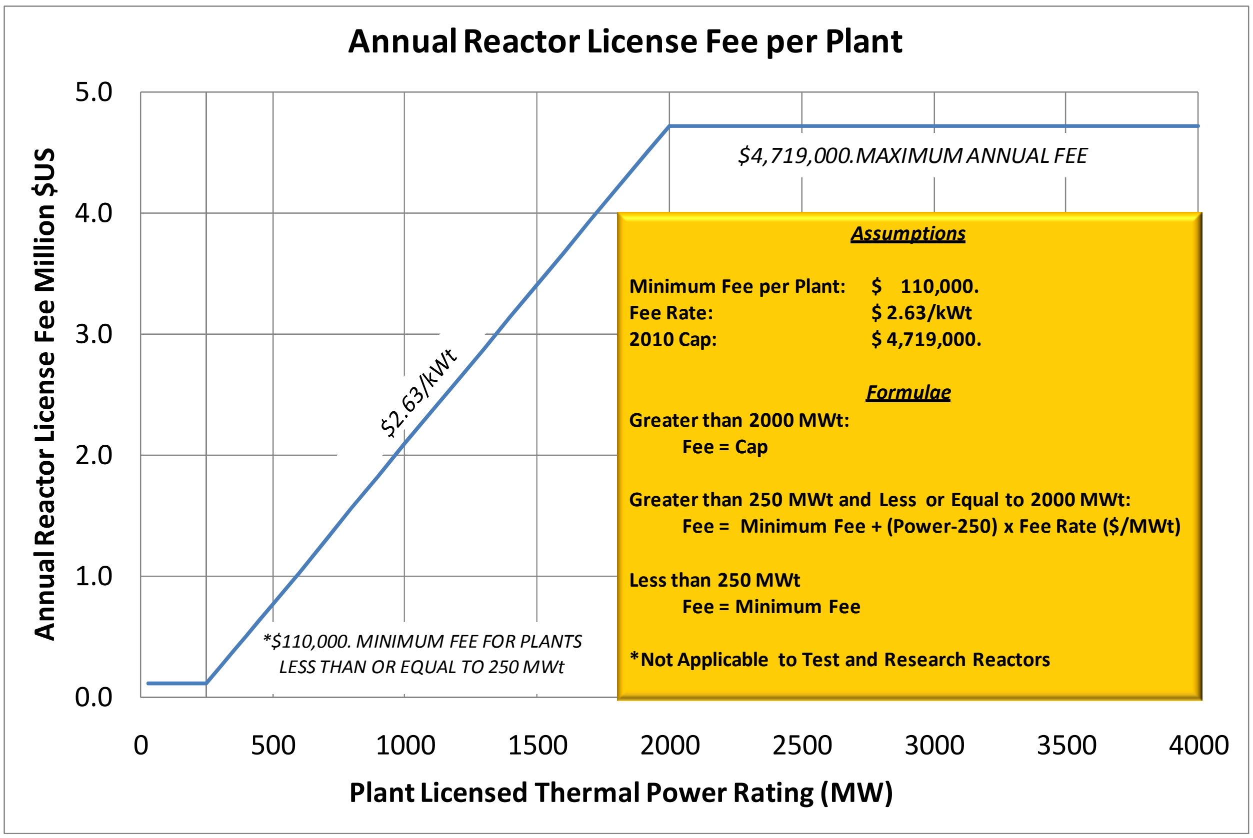
It is proposed that the annual fee be capped for all nuclear power plants with a plant licensed thermal power rating >2000 MW(thermal). This would be consistent with the NRC study that indicated that there was "no necessary relationship or predictive trend between thermal megawatt rating of a reactor and the NRC regulatory costs" [51 FR 33224 (September 18, 1986)]. It is expected that this will remain true for all plants exceeding 2000 MW(thermal). The current approach for establishing the fee structure would be implemented for these reactors. The proposed FY 2010 budget for operating power reactors is $4,719,000.

3.4. Annual Fee Structure

The following chart presents an annual reactor license fee structure based on Option 3 using FY 2010 information for purposes of discussion. Option 3 could also be developed in terms of a stepped-change annual fee structure rather than a linear scale.

4.0 RESOURCES FOR REGULATORY INFRASTRUCTURE DEVELOPMENT

The NRC has access to budgetary resources not provided by fees from the existing fleet of nuclear plants. First, 10% of the NRC's annual budget is obtained as appropriations from the federal government. For FY 2011, this amounts to a net appropriations request for $138.3 million (Ref. 10). Second, applicants submitting information as part of the preapplication or design certification processes are required to reimburse the NRC for their time. The proposed rate for FY 2011 is $259/hour (Ref. 5). Some of these resources have been used in the past to support regulatory infrastructure development for new technologies. This includes conducting confirmatory experiments and analyses to evaluate the safety aspects of new technologies and designs for nuclear reactors, materials, waste, and security. Although a primary portion of regulatory research has been related to the oversight of operating light water reactors (LWRs), recent applications for advanced LWRs and preapplication activity initiated by non-LWR vendors have prompted the NRC to consider long-term research needs (Ref. 10).



5.0 RECOMMENDATIONS

The ANS President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) strongly supports the NRC's effort to develop a variable reactor license fee structure for SMRs and multi-module nuclear plants. The ANS Task Force believes that the following principles are helpful in determining the best annual license fee structure under 10 CFR 171:

1. Ensure public safety by adequately reimbursing the NRC for the cost of regulatory oversight.

2. Utilize a fee structure that equitably shares regulatory oversight costs among both large- and smaller-scale generation facilities and ensure that the existing fleet of nuclear plants does not bear the regulatory framework development costs associated with deploying new technologies.

In light of these principles, the SMR Special Committee recommends that the NRC implement Option 3, a sliding or terraced scale based on fairness to the stakeholders and allocating regulatory expenses to ratepayers rather than investors. Where the owner and operator of the reactor lack the ability to directly or indirectly pass through NRC licensing fees to ratepayers, the annual fee for that reactor should be covered by general tax revenues, subsidies from government agencies [such as the U.S. Department of Energy (DOE)], or other sources. To the extent a facility receives market risks and rewards for sales of its output, increasing profitability would be reflected in increased tax revenue. To the extent such a facility is not profitable, charging a flat reactor licensing fee to that facility would be inconsistent with promoting safety, assuming the link between operating revenue and safety implicit in 10 CFR 50.33(f) (Ref. 11) exists. Basing the amount of licensing fee obtained from general tax revenues or government subsidies for merchant plants based on the thermal output for annual license fees as proposed in Option 3 above is a reasonable approach that balances stakeholder interests. Adjustment of the NRC fee structure and funding of SMR licensing activities through general tax revenue would likely require legislative action.

The structure proposed by this option balances the benefits of smaller reactors and equitably distributes regulatory oversight costs by the nature of the rate-setting mechanism applicable to the facility. Where the facility lacks the authority or market power to pass licensing fees through to ratepayers, charging the facility a flat licensing fee is inconsistent with the safety assumptions embodied in 10 CFR 50.33(f)(2). We believe that this structure will initially help enable the development of SMR and multi-module nuclear plants by reducing financial barriers to entry. From a long-term perspective, this may benefit the safety, security, and efficiency of future large-scale facilities and the nuclear industry as a whole.

The SMR Special Committee recommends that the federal appropriations portion of the NRC budget (i.e., non-fee base) continue to provide for the cost of developing the regulatory infrastructure needed to (1) conduct 10 CFR 170 activities related to SMR and multi-module designs and (2) assure their safe operation subsequent to deployment. This approach recognizes that the existing fleet of nuclear plants should not be expected to bear the regulatory framework development costs associated with deploying new SMR technologies and is consistent with NRC testimony to Congress (Ref. 12). Federal appropriations for such purposes have been mandated for other forms of energy production as part of our nation's efforts to assure energy independence and security (Refs. 13 and 14). The SMR Special Committee further recommends that collaborative efforts between the NRC, the DOE, and reactor vendors be considered for infrastructure development that requires SMR regulatory research.

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NUCLEAR INSURANCE AND LIABILITY FOR SMRs

Lead Writer – Charles W. Hess

American Nuclear Society (ANS)

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1.0 INTRODUCTION

Small and Medium-Sized Reactors (SMRs) are a new and exciting development in the energy industry. They represent a lower threshold for entry into the carbon-free energy market that is independent and reliable. Their technical practicality and small technical footprint are compelling realities. However, their economic attractiveness can be hampered by indiscriminate application of rules and regulations developed for large light water reactors (LWRs). This white paper examines the issue of insurance and liability coverage for SMRs.

The Price-Anderson Act provided financial protection to cover liability claims in the unlikely event a nuclear incident was to occur at an operating SMR site. This financial protection protects the public with private liability insurance (currently about $300 million for existing large LWRs) for each reactor unit and contributes a prorated share to a secondary pool of coverage for liability exceeding $300 million. The Price-Anderson Amendments Act (PAAA) revised the funding mechanism to rely solely on operator policies and a retroactive premium liability. If public liability for a commercial reactor accident exceeds the cap (currently about $10 billion), Congressional action would be needed, or plaintiffs' recoveries would be reduced.

In addition, U.S. Nuclear Regulatory Commission (NRC) regulations require entities seeking to operate nuclear reactors to secure at least $1.06 billion in on-site property damage insurance per reactor. In the event of an accident, the proceeds of this insurance must first be used to cover costs to stabilize the reactor or other similar cleanup and decontamination costs necessary to limit further harm to the public health. This amount was established as the maximum insurance commercially available on reasonable terms at the time (1980s) when the regulations were codified, and the amount of insurance is not adjusted based on actual risk presented by the reactor.

Both of these provisions place an unwarranted financial burden on an SMR, especially a small reactor, i.e., one with an electrical output of 300 MW(electric) or less. In addition, many of the SMR applications are for process heat only, so the use of MW(electric) limits unnecessarily complicates the issues. This paper will explain the issues in greater depth and will make recommendations for regulatory action.

2.0 BACKGROUND

The Price-Anderson Act was enacted in 1957 as Section 170 of the Atomic Energy Act (Refs. 1, 2, and 3). The objectives of the Act are to encourage private participation in the development of nuclear power by removing the deterrent of potentially astronomical liability claims, while simultaneously assuring that sufficient funds would be available to compensate the public for damages sustained in the event of a serious nuclear incident. The Act accomplishes these dual objectives by establishing a mandatory system of financial protection for nuclear power plants that covers persons potentially liable for a nuclear incident and provides compensation to those injured by such an incident.

The Price-Anderson Act was enacted into law in 1957 and has been revised several times. It constitutes Section 170 of the Atomic Energy Act. The latest revision was enacted through the "Energy Policy Act of 2005," and extended it through December 31, 2025 (Ref. 4).

The main purpose of the Price-Anderson Act is to ensure the availability of a large pool of funds (currently about $10 billion) to provide prompt and orderly compensation of members of the public who incur damages from a nuclear or radiological incident no matter who might be liable. The Act provides "omnibus" coverage; that is, the same protection available for a covered licensee or contractor extends through indemnification to any persons who may be legally liable, regardless of their identity or relationship to the licensed activity. Because the Act channels the obligation to pay compensation for damages, a claimant need not sue several parties but can bring its claim to the licensee or contractor.

The PAAA required NRC licensees and U.S. Department of Energy (DOE) contractors to enter into agreements of indemnification to cover personal injury and property damage to those harmed by a nuclear or radiological incident, including the costs of incident response or precautionary evacuation and the costs of investigating and defending claims and settling suits for such damages. The scope of the Act includes nuclear incidents in the course of the operation of power reactors, test and research reactors, DOE nuclear and radiological facilities, and transportation of nuclear fuel to and from a covered facility. (Public liability arising out of nuclear waste activities funded by the Nuclear Waste Fund would be compensated from the Fund.)

Power reactor licensees are required to have the maximum level of primary insurance available from private sources (currently $300 million) and to contribute up to $95.8 million per unit to a secondary insurance pool, payable in annual installments of $15 million or less and subject to adjustments for inflation at 5-year intervals. The combined primary and secondary insurance coverage now totals more than $10 billion.

The NRC codifies the conditions for indemnity agreements, liability limits, and fees for the different classes of licensees in 10 CFR 140 (Ref. 3). Power reactors rated below 100 MW(electric), for example, have lower primary insurance requirements than larger reactors, while the financial protection required for nonprofit educational reactors is a function of their maximum power and the neighboring population. The DOE also establishes indemnity agreements with its nuclear contractors. The liability limit for DOE facilities is $10 billion subject to adjustments for inflation.

In the event of a nuclear incident involving damages in excess of the limits established in the Act, Congress could take further actions, including the appropriation of funds.

3.0 PROBLEM/ISSUE STATEMENT

The PAAA and the implementing regulatory guidance all suffer from an unintended bias toward large, central electrical generating stations based on LWR technology. This is understandable because until recently, that was the only technology that existed. Therefore, there are implicit assumptions about efficiency and safety basis contained within the PAAA and the regulations. SMRs will be many different reactor technologies, many with thermal efficiencies significantly greater than existing LWRs. Also, many of the applications for SMRs are for process heat only. Therefore, the use of electrical output as a means of making definitions will become problematic.

More significantly, the existing Act and regulations result in overinsuring SMRs to the extent that their economic justification comes into question. This is especially true of small SMRs. It should be noted that several of the operating units in the United States today technically qualify as medium-sized reactors. And, while there may be a need to reevaluate them in the future, the biggest issue is not existing fully depreciated assets but rather a new generation of advanced SMRs that are based on Generation IV design principles or are so small as to call into question the fundamental assumptions underpinning the entire discussion of public and private property liability coverage.

Finally, many SMR technologies are based on a modular design approach, that is, several reactors powering a single turbine generator or providing steam to a common process heat load. These designs impact the specifics of the applicability of the Act and the regulations because this design approach is not found in common practice in the nuclear plants in use in the United States.

So, while the Act and implementing regulations recognize that small reactors may exist, neither are in alignment with the definition of small SMRs. Neither recognizes that an SMR may be deployed to make process heat only, and there is no recognition that some concepts may have more than one reactor per plant. The result is to cause the small SMRs to overinsure their operations and to incur costs all out of proportion to any potential revenue stream.

4.0 DISCUSSION AND ACTUAL WORK

1. MECHANISMS FOR PROVIDING FINANCIAL PROTECTION

The PAAA's coverage for public liability claims arising from nuclear incidents at nuclear power plants is implemented through a combination of private financial protection provided by commercial insurance companies and government indemnification. The PAAA and the NRC regulations provide various mechanisms for implementing this coverage depending on the size and operating status of the nuclear reactor.

The PAAA distinguishes between nuclear power plants having a rated electrical capacity of 100 MW(electric) or more and those having a lower rated electrical capacity. The Act requires licensees of plants having a rated capacity of 100 MW(electric) or more to maintain two types of financial protection. The first—known as "primary financial protection"—is "the maximum amount of insurance available at reasonable cost and on reasonable terms from private sources."[1] The second type of financial protection required for such plants—known as "secondary financial protection"—is insurance maintained under an industry "retrospective rating plan" providing for retroactive, deferred premium charges that become due only if needed to pay for public liability claims arising under the Act.[2]

For nuclear power plants having rated capacities <100 MW(electric), the PAAA authorizes the NRC to set the amount of primary financial protection to be maintained by licensees for such plants based on factors such as the cost and terms of available private insurance and the hazards associated with the plant.[3] Further, such plants are not required to maintain secondary financial protection.[4] The NRC has prescribed regulations that establish the amount of financial protection (ranging from $1 million to $74 million) to be maintained by reactors of <100 MW(electric).[5] For reactors with thermal power levels in excess of 10 megawatts, the regulations set forth a formula for calculating the amount of financial protection to be maintained by the licensee based on the power level of the reactor and the size of the nearby surrounding population.

The NRC regulations allow a licensee to meet its financial protection requirements under the Act either through private liability insurance or self-insurance.[6] As noted above, the regulations set forth "exemplary" insurance contracts "acceptable" to the NRC by which a licensee may satisfy its financial

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[1] This amount is currently $300 million. See 10 CFR 140.11(a)(4)
[2] 42 U.S.C. § 2210(b)(1). The maximum amount of deferred premium (adjusted for inflation every 5 years) to be charged each nuclear plant of 100 MW(electric) or more per nuclear incident is currently $95,800,000. Id.; 10 CFR 140.11(a)(4). 42 U.S.C. §§ 2210(b)(1), 2210(t). In addition, plants of 100 MW(electric) or more may be assessed an additional 5% surcharge where claims exceed the maximum amount of financial protection. 42 U.S.C. § 2210(o)(1)(E). Given 104 operating reactors within the United States of 100 MW(electric) or more, the secondary level of financial protection for such plants amounts to approximately $10 billion.
[3] 42 U.S.C. § 2210(b)(1).
[4] Id
[5] 10 CFR 140.11(a)(1)-(3) and 10 CFR 140.12.
[6] 10 CFR 140.14; see also 10 CFR 140.15.

protection requirements under the PAAA.[7] These exemplary contracts require insurance policies provided to meet the financial protection requirements of the Act to include in their coverage the "named insured" and "any other person or organization" who may have legal responsibility for injury or damage caused by a nuclear incident.[8] Hence, in accordance with Congress's intent, insurance provided under the Act covers not only the named insured but also "any other person who may be liable" for a nuclear incident, including "[a]ll vendors, architect-engineers" and other contractors and suppliers responsible for the design and construction of a nuclear facility.[9]

Where the amount of financial protection required to be maintained by a licensee is less than $560 million, the PAAA requires the NRC to enter into an indemnification agreement with the licensee.[10] This agreement is to indemnify and hold harmless "the licensee and other persons indemnified . . . from public liability arising from nuclear incidents which is in excess of the level of financial protection required of the licensee."[11]

The PAAA defines "person indemnified" to mean the NRC licensee "with whom an indemnity agreement is executed . . . and any other person who may be liable for public liability."[12] Thus, the indemnification agreement, like the financial protection provided by the licensee, would cover all vendors, contractors, suppliers, and anyone else who may be liable for a nuclear incident.

The maximum amount of government indemnity provided under an agreement of indemnification under the PAAA is $500 million.[13] This amount is to be reduced dollar for dollar by the excess over $60 million in financial protection insurance maintained by the licensee.[14]

2. LIMITATION OF LIABILITY

The PAAA establishes limitations on the aggregate public liability compensable for a single nuclear incident.[15] For nuclear reactors <100 MW(electric), the limit of liability is the sum of the financial protection maintained by the licensee and the amount of government indemnification provided by the PAAA.[16]

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[7] 10 CFR 140.91 through 10 CFR 109.
[8] 10 CFR 140.91, Appendix A, Article II (exemplary primary insurance policy); see also 10 CFR 140.109, Appendix I, Declaration Items 1 and 2 (exemplary secondary insurance policy).
[9] S. Rep. No. 296, supra, reprinted in 1957 U.S.C.C.A.N. at 1811-12, 1818; S. Rep. No. 454, supra, reprinted in 1975 U.S.C.C.A.N. 2251, 2256.
[10] 42 U.S.C. § 2210(c).
[11] Id.
[12] 42 U.S.C. § 2014(t)(1); see also, 10 CFR 140.92, Appendix B and 10 CFR 140.93, Appendix C.
[13] 42 U.S.C. § 2210(c).
[14] Id.
[15] 42 U.S.C. § 2210(e).
[16] 42 U.S.C. § 2210(e)(1)(C). The statutory limit on liability for nuclear incidents at plants having a rated capacity of 100 MW(electric) or more is equal to the amount of financial protection required to be maintained by such licensees (which, including both the primary and secondary financial protection required for such plants, exceeds $10 billion). 42 U.S.C. § 2210(e)(1)(A).

Neither the licensee nor any other party is liable for claims beyond the aggregate liability limits set by the PAAA. In the event the aggregate public liability for a nuclear incident exceeds the statutory cap, Congress is to review the situation and to take whatever action it deems necessary to provide full and prompt compensation to the public.[17]

3. EXCLUSIONS FROM PRICE-ANDERSON COVERAGE

The three specified exclusions from Price-Anderson coverage for nuclear reactors licensed by the NRC are for (1) “claims under State or Federal workmen's compensation acts of employees . . . who are employed at the site of and in connection with the activity where the nuclear incident occurs”; (2) “claims arising out of an act of war”; and (3) “claims for loss of, or damage to, or loss of use of property which is located at the site of and used in connection with the licensed activity where the nuclear incident occurs.”[18]

The exclusion of claims for property damage at the site is the most significant of the three. This exclusion was added to the PAAA by amendment in 1961 specifically to exclude Price-Anderson coverage for on-site property used in connection with activities licensed by the NRC, particularly the licensed nuclear reactor itself.[19] However, as discussed below, since the Three Mile Island (TMI) experience, the NRC has required utilities to carry separate insurance for stabilizing the reactor and decontaminating the site after a nuclear incident.

The other two exclusions to the PAAA's coverage are narrow. Workmen's compensation claims for on-site personnel are excluded from Price-Anderson coverage because “insurance carriers who pay workmen's compensation” for workers at nuclear facilities are assumed to “know and understand the risks which they are taking and charge accordingly.”[20] Workmen's compensation systems provide benefits to employees injured in the course of their employment regardless of the fault of their employer who in turn is generally protected from tort liability for work-related accidents.[21] This exclusion for “State or Federal workmen's compensation acts” is to be construed to include “any law similar to the compensation acts,” such as occupational disease acts.[22]

Workmen's compensation systems, however, do not generally cover claims of tort liability for injury brought by an employee against a third party.[23] Accordingly, such claims fall under the coverage of the

[17] 42 U.S.C. §§ 2210(e) and 2210(i).
[18] 42 U.S.C. § 2014(w).
[19] See H.R. Rep. No. 963, 87th Cong., 1st Sess. (1961), reprinted in 1961 U.S.C.C.A.N. 2591, 2600.
[20] S. Rep. No. 296, supra, reprinted in 1957 U.S.C.C.A.N. at 1819.
[21] See, e.g., Rolick v. Collins Pine Co., 925 F.2d 661, 663 (3rd Cir.1991) cert. denied, 113 S. Ct. 1417 (1993); Smith v. Gould, Inc., 918 F.2d 1361, 1363-64 (8th Cir. 1990).
[22] S. Rep. No. 296, supra, reprinted in 1957 U.S.C.C.A.N. at 1819. In addition to workrmen's compensation, the insurance policies exclude liability for “bodily injury to any employee of the insured" employed at the site. See 10 CFR 140.91, Appendix A, Article IV(b). According to ANI, this provision is intended to exclude employer's liability involving limited situations in certain states where an employee may bring suit against his employer outside the workmen's compensation system.
[23] See, e.g., Missouri Pub. Serv. Co. v. Henningsen Steel, 612 F.2d 363 (8th Cir. 1980).

PAAA. Thus, for example, a tort claim alleging radioactive exposure brought by an employee of a subcontractor at a nuclear plant against the contractor and the plant owner was adjudicated under Price-Anderson.[24] Similarly, the PAAA was also found to apply in two separate wrongful death actions brought by relatives of employees against the owners of nuclear power plants.[25]

The exclusion for claims arising out of an act of war is not based on any intent to hold licensees or others liable for such claims.[26] Rather, it was enacted in recognition that special governmental measures adapted to the exigencies of war would be required. "Any single act of sabotage would be covered" under Price-Anderson "if it could not be proven to be an act of war."[27] In this respect, both American Nuclear Insurers (ANI) and the NRC have confirmed to Congress that acts of terrorism—such as those that occurred September 11, 2001—would be covered under Price-Anderson.[28]

4. RESOLUTION AND PAYMENT OF PUBLIC LIABILITY CLAIMS UNDER THE PAAA

The resolution and payment of public liability claims arising under the PAAA follow the same process as claims made under any insurance policy covering natural disasters. For example, following the nuclear incident at TMI, ANI established a special nearby office to pay living expenses claims for persons who had evacuated the 5-mile area around the TMI Unit 2 (TMI-2) reactor at the suggestion of the Pennsylvania Governor.[29] Further, the PAAA provides a structured process for resolving disputed claims arising under the PAAA. It establishes a federal cause of action for public liability claims[30] and provides that the U.S. district court for the district where the nuclear incident takes place shall have jurisdiction to hear claims for compensation arising from the nuclear incident.[31] Additionally, the PAAA provides the court with special powers to provide for the prompt, efficient, and fair handling of the myriad claims that could arise under the PAAA.[32]

While the PAAA establishes a federal cause action in federal court, it does not establish substantive legal standards for determining public liability for a nuclear incident. Rather, the PAAA expressly provides that the "substantive rules for decision" for public liability claims arising under Price-Anderson "shall be derived from the law of the State in which the nuclear incident involved occurs, unless such law is

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[24] O'Conner v. Commonwealth Edison Co., 13 F.3d 1090 (7th Cir.), cert. denied, 114 S. Ct. 2711 (1994).
[25] McLandrich, et al. v. S. Cal. Edison Co., 942 F. Supp. 457 (S.D. Cal. 1996) and Corcoran v. N.Y. Power Auth. 935 F. Supp. 376 (S.D. N.Y. 1996).
[26] S. Rep. No. 296, supra, reprinted in 1957 U.S.C.C.A.N. at 1819.
[27] Id.
[28] See Testimony of John Quattrocchi, Senior Vice President, Underwriting, American Nuclear Insurers Before the United State Senate Transportation, Infrastructure, and Nuclear Safety Subcommittee of the Environment and Public Works Committee, January 23, 2002; Letter from Richard A. Meserve, Chairman, Nuclear Regulatory Commission, to Senator Ernest F. Hollings, Chairman, Committee on Commerce, Science and Technology, December 11, 2001.
[29] See NUREG-0957, "The Price Anderson Act—The Third Decade" (Dec. 1983) at I-6.
[30] O'Conner v. Commonwealth Edison Co., supra, 13 F.3d at 1095-1101; In re TMI Cases Consol., 940 F.2d 832 (3d Cir. 1991), cert. denied, 503 U.S. 906 (1992).
[31] 42 U.S.C. § 2210(n)(2). Suits filed in other federal district courts or state courts are to be transferred to the district court where the nuclear incident occurred upon request of the defendant, or of the NRC or the Secretary of Energy, as appropriate. Id.
[32] 42 U.S.C. § 2210(n)(3); see also 42 U.S.C. § 2210(o).

inconsistent" with the PAAA.[33] The PAAA does require, however, insurance policies and indemnity agreements to waive in the event of an "extraordinary nuclear occurrence" key legal defenses that might otherwise be available under the law of some states.[34] These waivers make the application of the PAAA equivalent to strict liability for an "extraordinary nuclear occurrence," which was the intent of Congress in requiring such waivers.[35]

As reflected in the legislative history, Congress intended an "extraordinary nuclear occurrence" to which the waiver of defenses applies to be a "serious nuclear incident" as determined by the NRC in accordance with the PAAA. The NRC has promulgated regulations establishing criteria to govern its determination of whether a nuclear incident qualifies as an extraordinary nuclear occurrence.[36] Applying these criteria, the NRC determined that the 1979 nuclear incident at the TMI-2 plant was not an extraordinary nuclear occurrence because estimated radiation doses and surface contamination levels off-site were about an order of magnitude lower than those specified by the criteria set forth in its regulations.[37]

5. NRC REGULATORY REQUIREMENTS FOR ON-SITE PROPERTY INSURANCE

Following the nuclear incident at TMI, the NRC became concerned that some nuclear utilities may not be able to "finance the clean-up costs resulting from a nuclear-related accident."[38] Because of the "substantial importance to the public health and safety of adequately cleaning up nuclear accidents," the NRC revised its regulations to require that licensees maintain on-site property damage insurance to ensure sufficient funds to clean up and decontaminate the reactor and reactor site after a nuclear incident.[39]

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[33] 42 U.S.C. § 2014(hh).

[34] 42 U.S.C. § 2210(n)(1). The Act and regulations require, for example, the waiver of issues and defenses related to the conduct of the claimant, such as contributory negligence or assumption of the risk, or to the fault of the insured, such as negligence. Id. 10 CFR 140.81(b); 10 CFR 140.91, Appendix A, "Waiver of Defense Endorsement"; 10 CFR 140.92, Appendix B, Article II, Paragraph 4. The Act also requires the waiver of "any issue or defense based on any statute of limitations if suit is instituted within three years from the date on which the claimant first knew, or reasonably could have known, of his injury or damages and the cause thereof." 42 U.S.C. § 2210(n)(1).

[35] See S. Rep. No. 1605, 89th Cong., 2nd Sess. (1966), reprinted in 1966 U.S.C.C.A.N. 3201, 3209.

55 Id. In accordance with Congress's intent, the Act defines an extraordinary nuclear occurrence as an off-site discharge or dispersal of source, special nuclear, or by-product material that the NRC (1) "determines to be substantial," and (2) "determines has resulted or will probably result in substantial damages to persons off-site or property off-site." 42 U.S.C. § 2014(j).

[36] See 10 CFR 140.83, 10 CFR 140.84, and 10 CFR 140.85.

[37] In re Metropolitan Edison Company (Three Mile Island, Unit 2), CLI-80-13, 11 NRC 519 (1980).

[38] Financial Qualifications; Domestic Licensing of Production and Utilization Facilities; Proposed Rule, 46 Fed. Reg. 41,786, 41,788 (Aug. 18, 1981).

[39] Id. See also Elimination of Review of Financial Qualifications of Electric Utilities in Licensing Hearings for Nuclear Power Plants. Final Rule, 47 Fed. Reg. 13,750 (Mar. 31 1982) Simultaneous with eliminating review of financial review of financial qualifications for electric utilities, the NRC promulgated requirements for licensees to maintain property insurance under 10 CFR 50.54(w). Id.

The requirements for on-site property insurance coverage are set forth in 10 CFR 50.54(w) (Ref. 5). This section requires licensees of commercial nuclear power plants to "take reasonable steps to obtain insurance available at reasonable costs and on reasonable terms from private sources or to demonstrate to the NRC that it possesses an equivalent amount of protection" to stabilize the reactor and decontaminate the reactor and the reactor site in the event of a nuclear incident.[40] Absent an exemption, the amount of property insurance coverage required by the regulation is set at "either $1.06 billion or whatever amount of insurance is generally available from private sources, whichever is less."[41] The $1.06 billion amount of insurance prescribed by the regulation was based on a study conducted by Pacific Northwest Laboratory that analyzed the cleanup costs associated with a hypothetical accident for a typical, large pressurized water reactor (PWR).[42]

In addition to specifying the amount of the coverage, the regulations specify that the insurance policy must "clearly state that . . . any proceeds must be payable first for stabilization of the reactor and next for decontamination of the reactor and the reactor station site."[43] The insurance "may, at the option of the licensee, be included within policies that also provide coverage for other risks, including, but not limited to, the risk of direct physical damage."[44] Licensees must report annually to the NRC regarding the current levels and sources of their property insurance or alternative financial security.[45]

The only entity that currently provides nuclear property insurance for commercial reactors operating in the United States is Nuclear Electric Insurance Limited (NEIL).[46] NEIL is a mutual insurance company whose members are the owners of the U.S. commercial reactors to whom it sells property insurance. NEIL provides two layers of property insurance. The amount of coverage provided by the first layer is $500 million, and the coverage provided by the second layer is $1.75 billion, for total coverage of $2.25 billion.[47]

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[40] 10 CFR 50.54(w).

[41] 10 CFR 50.54(w)(1). As discussed in Section 4 above, based on previous exemptions granted by the NRC, the licensee for a SMR should be able to obtain an exemption that would significantly reduce the amount of on-site property insurance required for the facility to an amount on the order of $180 million.

[42] See Changes in Property Insurance Requirements for NRC Licensed Nuclear Power Plants, Final Rule, 52 Fed. Reg. 28,963, 28,964 n. 1 (Aug. 5 1987), referencing NUREG/CR-2601, "Technology, Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents" (Nov.1982). Analyzing the "accident cleanup costs at a reference 1,000 MWe PWR following a scenario 3 accident," ($404 million) and adding "additional costs that can appropriately be ascribed to such an accident" ($656 million), the study determined that the appropriate amount of property insurance for the circumstances studied was $1.06 billion. Id. at 28,964. The NRC adopted this amount, reasoning that more than that amount was commercially available at the time the regulation was adopted and "no other amount is as technically supportable." Id.

[43] Id.

[44] Id.

[45] 10 CFR 50.54(w)(3).

[46] While ANI had traditionally provided nuclear property insurance as well as nuclear liability insurance, it ceased offering property insurance as of about 2000.

[47] See "NEIL Insurance Policies – Summary" at http://www.nmlneil.com/policies.html.

With certain limitations, the policies cover damage and destruction of property generally at the site due to a nuclear accident but, in accordance with NRC requirements, give priority to stabilization of the reactor and decontamination of the reactor and reactor site.

NEIL also offers an "accidental outage insurance policy" to cover the costs of lost power generation due to a prolonged accidental outage of a nuclear plant.[48]

6. POTENTIAL EXEMPTIONS TO NRC PROPERTY INSURANCE REQUIREMENTS

10 CFR 50.54(w) does not expressly provide exceptions to the requirement that licensees hold property insurance in the amount of "either $1.06 billion or whatever amount of insurance is generally available from private sources, whichever is less." The absence of such an express provision in 10 CFR 50.54(w) does not, however, preclude exemptions from its requirements being sought under 10 CFR 50.12 (Ref. 5) of the NRC's regulations, and such exemptions have been granted by the NRC.

Under 10 CFR 50.12 the NRC may grant an exemption from the requirements contained in 10 CFR 50 (Ref. 5) upon determining that (1) the requested exemption is "authorized by law, will not present an undue risk to public health and safety, and [is] consistent with the common defense and security"[49] and (2) "special circumstances are present" that warrant the granting of the exemption.[50] The regulation identifies the "special circumstances" or justifications for which an exemption may be granted.[51] If a licensee believes that its situation warrants an exemption from any requirement under 10 CFR 50, it can apply for an exemption under one of the specific justifications included in 10 CFR 50.12 for seeking an exemption.

One of the justifications included in 10 CFR 50.12 for allowing exemptions from licensing requirements is if "compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated."[52] Since the provisions of 10 CFR 50.54(w) were implemented, a number of licensees have used this justification to argue successfully that complying with the requirement of providing $1.06 billion in property insurance for their plant would present an undue hardship and was unnecessary in consideration of the limited threat to public health and safety posed by their facilities. These exemptions from the requirement to maintain $1.06 billion in property insurance coverage include facilities that were licensed to operate at much lower power levels than that of a typical, large PWR upon which the $1.06 billion amount prescribed in 10 CFR 50.54(w) was based.

As discussed below, an exemption should similarly be obtainable for the SMRs that would significantly reduce the amount of property insurance required to be maintained for the facility in accordance with the much reduced risk associated with the small generating capability and small physical size of SMRs with power ratings <100 MW(electric).

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[48] Id.
[49] 10 CFR 50.12(a)(1).
[50] 10 CFR 50.12(a)(2).
[51] 10 CFR 50.12(a)(2)(i)-(vi).
[52] 10 CFR 50.12(2)(iii).

5.0 CONCLUSIONS

The PAAA provides assurance that injury to the public from a nuclear incident will be compensated by providing a comprehensive Federal program covering SMR operators, vendors, suppliers, contractors, and investors for public liability for personal injury and property damage caused by a nuclear incident in the United States. The PAAA requires all SMR licensees to procure a primary layer of liability insurance coverage, the amount of which depends on the rated capacity of the reactor. Licensees of reactors with a rated capacity of 100 MW(electric) or more are additionally required to maintain a secondary level of financial protection under the retrospective premium plan to satisfy any public liability claims in excess of the primary coverage. Altogether, the public liability protection now amounts to approximately $10.7 billion for nuclear power plants >100 MW(electric).

For nuclear power plants having rated capacities <100 MW(electric), the PAAA authorizes the NRC to set the amount of primary financial protection to be maintained by licensees for such plants based on factors such as the cost and terms of available private insurance and the hazards associated with the plant. Further, such plants are not required to maintain secondary financial protection. The NRC has prescribed regulations that establish the amount of financial protection shall not exceed $74 million for reactors of <100 MW(electric). For reactors with thermal power levels between 10 and 100 megawatts, the regulations set forth a formula for calculating the amount of financial protection to be maintained by the licensee based on the power level of the reactor and the size of the nearby surrounding population. Furthermore, for SMRs of <100 MW(electric), the federal government will provide indemnification coverage against liability claims exceeding the required primary layer of protection up to $500 million. The maximum amount of government indemnity is reduced by the amount that the financial protection required by the NRC exceeds $60 million. For such indemnity, the NRC charges a nominal fee (between $100 and $3,000 per year depending on reactor power level and amount of the indemnity).

In addition, the NRC requires SMR licensees to maintain separate insurance coverage for damage to on-site property and requires these insurance proceeds be reserved in the event of a nuclear incident to ensure that the licensee has sufficient funds to stabilize the facility and clean up the site. The amount of on-site property insurance required by the NRC for these purposes is $1.06 billion.

In developing this paper, the American Nuclear Society (ANS) President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) has studied the Price-Anderson liability insurance and NRC property insurance requirements as applied in the SMR context. We have concluded that these currently applicable insurance requirements are higher and attach at an earlier time than is commercially reasonable for SMR operators, without any significant relationship to the assurance of public health and safety. Based on this conclusion, we recommend the following policy changes.

Exclusion of SMR Licensees from Secondary Financial Protection Requirement

Currently, the exemption from participating in the secondary level of protection (the retroactive premium) does not apply to reactors >100 MW(electric). The SMR Special Committee believes the exemption should be extended to all small reactors [i.e., <300 MW(electric)] with a demonstrated improved level of risk performance over the reactors currently operating. Because of the reduced risk of advanced SMRs relative to those traditional large reactors currently operating, an advanced SMR should not be accountable under the retrospective premium plan for accidents at large plants to the same extent as all existing and future large reactors. Furthermore, maintaining a dramatic difference in required public liability coverage between a small modular advanced reactor technology (SMART) reactor [95 MW(electric)] and an mPower reactor [125 MW(electric)], despite similar design concepts and corresponding low safety risk, is not justifiable. Rather, all advanced SMR licensees should be exempt from liability for the retroactive premium in the secondary level of protection in the event of a nuclear incident. The coverage and any indemnity fee should be risk-informed, not based solely on reactor power level.

Reduction of Property Insurance Coverage Required of SMR Licensees

The smaller size and safer operation of SMRs warrants imposing a less burdensome property insurance requirement than the currently required $1.06 billion based on a large, 1000-MW(electric) plant. In a multi-modular configuration, a credible accident would impact only one module, so the potential amount needed to stabilize a facility after an event would be much less for a 1000-MW(electric) facility built from multiple modules compared to one 1000-MW(electric) reactor. The regulations default to requiring the maximum amount of property insurance available. The maximum available may carry a high premium without providing assurance that the amount of coverage is the right amount; it may be too much or too little.

6.0 RECOMMENDATIONS

The 100-MW(electric) break point set forth in the PAAA and the implementing regulations is arbitrary and unsustainable. A more equitable approach must be developed to avoid the need for exemptions and to encourage technological advances in all areas of SMR development. As stated above, all advanced SMR licensees should be exempt from liability for the retroactive premium in the secondary level of protection in the event of a nuclear incident. The coverage and any indemnity fee should be risk-informed, not based solely on reactor power level. Until a better solution is found, an interim measure of increasing the 100 MW(electric) to 1000 MW(thermal) would be a means to equitably address this issue for all small SMRs. There are many medium-sized reactors in operation in the United States today, and ultimately, a risk-informed approach to this issue may be extended to them as well.

Instead of an arbitrary requirement to carry the maximum amount of property insurance available, the SMR Special Committee recommends that small reactors [i.e., <300 MW(electric) or 1000 MW(thermal)] not carry property insurance for the purpose of assuring that the funds are readily available to stabilize the site in the event of an accident. A cost-effective alternative coverage mechanism would be a common SMR trust fund. Such a fund could be administered similar to the existing decommissioning trust fund. Alternately, facilities could have an agreement with a common industry organization, such as Institute of Nuclear Power Operations (INPO), to provide such funds to cover cleanup and stabilization costs of accidents involving SMRs. INPO obligation could be covered by a re-insurance agreement with an insurer like NEIL. Under either approach, the amount of funds available would be based on the amount anticipated to be needed for the intended health and safety response, and the pooling of payments or premiums by SMR licensees would be reduced relative to an arbitrary requirement unrelated to the reduced risk and size of SMRs compared to plants currently operating.

In order to develop a more appropriate set of regulations, the SMR Special Committee recommends forming a working committee comprising the ANS, Nuclear Energy Institute (NEI), and Electric Power Research Institute (EPRI) to formulate a technical basis for refining the insurance requirements in the regulations. The resultant rules should be equitable and risk-informed to be both fair to all stakeholders and to encourage technological advancement.

7.0 REFERENCES

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2. The Atomic Energy Act, 42 U.S.C. Section 11, § 2014.

3. C*ode of Federal Regulations*, Title 10, "Energy," Part 140, "Financial Protection Requirements and Indemnity Agreements"; Sec. 140.11, "Amounts of Financial Protection for Certain Reactors"; Sec. 140.12, "Amount of Financial Protection Required for Other Reactors"; Sec. 140.14, "Types of Financial Protection"; Sec. 140.15, "Proof of Financial Protection"; Secs. 140.91 Through 140.109, "Appendices A Through I"; Sec. 140.81, "Scope and Purpose"; Sec. 140.83, "Determination of Extraordinary Nuclear Occurrence"; Sec. 140.84, "Criterion I—Substantial Discharge of Radioactive Material or Substantial Radiation Offsite"; Sec. 140.85, Criterion II—Substantial Damages to Persons Offsite or Property Offsite," U.S. Nuclear Regulatory Commission.

4. Public Law 109-58, "The Energy Policy Act of 2005" (Aug. 8, 2005).

5. C*ode of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities"; Sec. 50.54, "Conditions of Licenses," Sec. 50.12, "Specific Exemptions," U.S. Nuclear Regulatory Commission.

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RISK-INFORMED AND PERFORMANCE-BASED LICENSING FOR SMRs

Lead Writer – Charles Bagnal
Assistant Writer – Andy Kadak

American Nuclear Society (ANS)

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1.0 INTRODUCTION

Under the current U.S. Nuclear Regulatory Commission (NRC) licensing schemes, small modular reactor developers lack a predictable and technology-sensitive licensing process for Small and Medium Sized Reactor (SMR) designs. The standard Design Certification (DC) process under 10 CFR 52 (Ref. 1) is too time-consuming to fit within the targeted development time frames for SMRs. A new set of licensing regulations incorporating risk-informed and performance-based criteria, as has been suggested, would likewise entail several years before enactment and implementation. The purpose of this paper is to evaluate the challenges to SMR development posed by the current licensing process and set forth both near-term and long-term approaches that could incorporate risk-informed and performance-based review criteria, resulting in significant licensing efficiencies for both the NRC as well the industry. This paper will discuss the background of NRC licensing approaches to advanced reactor technologies and the difficulties the existing licensing regime poses for SMR development. The paper will explain the use of licensing review basis documents and recommend their adoption in the near-term while new licensing regulations that incorporate risk-informed and performance-based review criteria are under consideration. The use of Licensing Basis Review (LRB) documents can facilitate timely SMR licensing within today's regulatory framework while placing more emphasis on risk-informed methods. Pursuing revisions to regulatory framework in parallel provides both a context for developing LBR documents and a solid foundation for future [Generation (GEN) IV] advanced reactor development.

2.0 BACKGROUND

A new class of SMRs has been specifically designed to meet the electrical power, water, hydrogen, and heat needs. In general, SMRs differ from current-generation light water reactors (LWRs) in many ways: size, moderator, coolant, fuel design, projected operation parameters, etc. These new reactors feature longer refueling intervals and simplified operations. Sized in the 10- to 50-MW(electric) range (very small and up to the 300-MW(electric) range (small to medium), these reactors are built through modularized factory production and designed for rapid site deployment and assembly. The anticipated fuel source is <20% ^{235}U uranium fuel with a nominal core life of 10+ years**.** Many SMRs have been designed to operate as multiunit integrated facilities with as many as 4 to 16 SMRs operating in unison.

SMRs also differ commercially from the current generation of LWRs. SMRs are factory built and may be fabricated entirely off-site. The fabricated reactors will be shipped to a site for installation, which may include locations overseas. As commercialization proceeds, SMR vendors may intend to fabricate SMRs without advanced long-term orders for installation.

For the near term, the preference is to rely on licensing advanced reactors under current regulations, where the experience base is robust and the review process is proven. Projects that largely rely on regulatory certainty prior to significant investment prefer use of the 10 CFR 52 one-step process; a more research-oriented project with funding less dependent on private investment may prefer use of the 10 CFR 50 (Ref. 2) two-step process for first-of-a-kind SMRs to advance construction of the first unit prior to completing design, while using 10 CFR 52 to license follow-on units that can incorporate design finalization work and operating experience from the first (or prototype) unit.

More emphasis on risk-informed and performance-based licensing for SMRs will bridge the gap between current, LWR-focused regulations and new review criteria specific to more advanced, smaller reactors. By focusing review and deterministic analysis on those aspects of SMR design that are most critical to safety, risk-informed methods will facilitate the use of much of today's regulatory framework for near-term construction and operation of SMRs.

3.0 PROBLEM/ISSUE STATEMENT

The DC process, typically lasting several years (and in many cases more than a decade) from preapplication meeting to eventual DC issuance, takes too long to be commercially viable for many SMR developers. However, to promulgate and rely on new regulations specifically tuned to SMRs may add uncertainty to licensing schedules, which would delay SMR construction in the United States.

It has been suggested that SMRs might be licensed more directly under new regulations that are more specifically tuned to the advances in technology that they represent, including the potential for mass production of reactor modules in a factory assembly line. Examples include the proposed 10 CFR 53 (Ref. 3), which would establish a new risk-informed and performance-based framework, or regulations that would focus on integral LWR systems, or gas-cooled or liquid metal reactor technology. However, unless there is a significant change in rulemaking methods for new regulations, establishing 10 CFR 53 or technology-specific rules would entail 5 to 10 years of concerted effort before the review of specific designs could begin. This would defer the potential benefits that SMRs can provide in the near term and delay their timely construction.

These issues are explored below in more detail.

4.0 DISCUSSION AND ACTUAL WORK

As established in NUREG-1368 (Ref. 4), General Design Criteria (GDC) that form the basis of Title 10 of *Code of Federal Regulations* (10 CFR 50, and, by extension, 10 CFR 52) are largely applicable to SMRs, despite their origin in operating LWRs. The exceptions are few and well defined, or focused on reactor core design, and do not represent an impediment to licensing, as pointed out directly in NUREG-1368 and NUREG-1338 (Ref. 5).

4.1 LICENSING REVIEW BASIS DOCUMENTS DEFINED

An LRB document is envisioned as an accord whereby the reactor design applicant and the NRC establish an agreed-upon licensing framework for proceeding in the absence of technology-specific regulations. LRB documents provide an interim means for conducting risk-informed and performance-based SMR design review that is consistent with both the industry's targeted development time frames and the NRC's regulatory goals. This device allows licensing to proceed now, but it does not preclude the pursuit of new regulations that may be of benefit in the longer term.

This is precisely the approach used to certify the first reactor designs [Advanced Boiling Water Reactor (ABWR), System 80+, AP600] under 10 CFR 52, where licensing review basis documents were negotiated to establish the framework for first-of-a-kind certification. This was before Chapter 18 "Human Factors Engineering," and Chapter 19, "PRA and Severe Accident Evaluation for New Reactors," were incorporated into NRC's Standard Review Plan, before NRC had completed its review of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document and before the closure of nearly 1,000 unresolved and generic safety issues, which the new regulations required.

4.2. FINALITY BENEFITS

A standard DC equips the developer with a substantial amount of regulatory certainty regarding future applications that incorporate the approved design. NRC regulations provide for finality of designs under 10 CFR 52.63 (Ref. 6), 10 CFR 52.135 (Ref. 7), or 10 CFR 52.171 (Ref. 8). For example, a certified design [10 CFR 52.63(a)] is not rescindable or modifiable except by notice-and-comment rulemaking on certain enumerated policy grounds. While a rulemaking specific to formalize LBR document approval may prove not timely, in like fashion, NRC policy statements can provide LRB documents a measure of finality, thus maximizing their value by ensuring future adherence to certain agreed-upon review criteria.

4.3 POLICY REASONS FOR NRC ADOPTION OF LICENSING REVIEW BASIS DOCUMENTS

4.3.1. Public Participation and Transparency

Implementation of LRB documents with the NRC on new DC applications would enable stakeholders to provide constructive input on new approaches to the DC process using their wealth of operating experience as well as their unique knowledge of the risk and performance features of proposed SMR designs. The industry would gain increased transparency on the regulatory side, thus facilitating interest in SMR development, while the NRC would benefit from laudable public involvement consistent with its open government initiative.

4.3.2. Regulatory Certainty and Better Safety/Standardization

As the LRB document provides assurances of the terms on which the design is to be evaluated by NRC staff and would have enforceable finality provisions, LRB documents ensure stability of the review process where the current regime lacks regulatory certainty. Further, stakeholder and regulator collaboration on the licensing process may help to standardize the LRB document content and inform any subsequent rulemakings concerning new advanced-technology licensing regulations, as proposed below.

The use of risk insights to develop exemptions that reflect the safety characteristics of SMRs is consistent with NRC's Probabilistic Risk Assessment (PRA) Policy Statement. The process of conforming safety requirements to SMRs is analogous to the risk-informed assessment of changes to a plant's licensing basis, described in Regulatory Guide 1.174 (Ref. 9).

4.4. BEYOND LICENSING REVIEW BASIS DOCUMENTS

4.4.1. New Regulations

In addition to near-term implementation of a regulatory approach to license innovative modular reactors, it is equally important to establish a longer-term regulatory approach for nontraditional reactor technologies in parallel. Many new reactor technologies are being developed but suffer from a constrained regulatory process that is largely based on prescriptive regulations tuned to more traditional LWRs. This forces the developer to show that certain regulations do not apply or seek exemptions instead of focusing on making the safety case for the technology.

Innovation in the nuclear industry has suffered because of this regulatory constraint, which does not incentivize developers to implement design features that would be given safety credit in the regulatory review. While progress is being made using technology-neutral regulatory approaches, more effort is needed to finalize a new set of regulations for a regulatory process based on a risk-informed approach to "reasonable assurance" of public safety using safety goals as the controlling metric. It is understood that with new designs, the database for operational equipment would not be robust, but fundamental design principles in core design and safety are sufficiently developed to allow for a risk-informed decision on safety based on separate effects and integral tests. This approach has proved effective in the application process for designs that have already been certified. Performance-based approaches may also assist in the licensing of new designs with limited operational experience by establishing measurable objectives and required remedial actions if operational objectives are not achieved.

For a truly technology-neutral framework based on risk, it is important to take full advantage of risk analysis in safety assessments. The historical "defense-in-depth" strategy that relies on numbers of physical barriers as a criterion needs to be replaced with a firmer understanding of design features and processes that also serve to prevent and mitigate accidents. The single-failure criterion may not be appropriate to risk-informed safety assessments since it defeats the fundamental purpose of a risk analysis, given that all components, regardless of safety classification, have the opportunity to fail in a probabilistic assessment. Single-failure criteria can be used to assess the importance of components and structures for design improvement, should the consequence be significant, but should not be mandatory.

By a rigorous application of risk analysis in a plant design, the important design-basis events can be deduced from the event and fault trees. In addition, safety classification of systems, structures, and components can be directly determined from the analysis, as can reliability requirements for component performance and the need for inspection, test, and surveillance based on component importance. The risk-informed assessment also allows for explicit treatment of uncertainties, which conventional deterministic analysis largely ignores by applying "margins" and "conservatisms" intended to bound these unknowns. The risk assessment methodology allows for a more transparent understanding of the safety basis of reactors.

A key element to development and implementation of innovative reactors is the use of a risk-informed framework, coupled with a demonstration test program upon which to issue DCs. Thus, the American Nuclear Society President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) recommends immediate development of a rulemaking to establish a new risk-informed, technology-neutral licensing process with a license-by-test element, to allow innovative designs to be developed and deployed more efficiently in the longer term.

5.0 CONCLUSIONS

Licensing Review Basis documents harmonize NRC licensing processes with SMR industry objectives on a temporary basis while new SMR-compatible regulations are under development. Near-term construction of SMRs can be realized by licensing under the current regulatory framework, while managing technology advances through the use of licensing review basis documents that include more direct use of risk-informed methods. This is an effective and proven approach to managing uncertainties in the licensing process for new technology, which in turn affect cost and schedule for a new generation of nuclear power reactors. New technology-neutral regulations should be pursued in parallel for longer-term benefits as technology continues to advance.

6.0 RECOMMENDATIONS

The SMR Special Committee recommends that the SMR community plan for the use of current regulations to license SMRs for near-term deployment, with licensing review basis documents negotiated to provide a framework for approval that will satisfy the regulator where current guidance may not apply. In order to compensate for the current absence of SMR-specific licensing criteria, the SMR Special Committee proposes (1) the incorporation of licensing review basis documents within the existing regulatory process and (2) the recommendation of new licensing criteria for advanced technology such as SMRs, to be implemented in the long term.

1. ***Interim Use of Licensing Review Basis Documents for SMR Design Review Process***. Utilizing licensing review basis documents enables SMR developers to reach agreement with the NRC on acceptable approaches to meeting the intent of current regulations (e.g., GDC) where new technology departs from that of operating LWRs. This approach provides regulatory certainty and standardization benefits to industry stakeholders while promoting the NRC policy goals of transparency and safety.

2. ***Initiation of Rulemaking for Risk-Informed, Technology-Neutral Licensing Process.*** A key element to development and implementation of innovative reactors is the use of a risk-informed framework, coupled with a demonstration test program upon which to issue DCs. Thus, the SMR Special Committee recommends immediate development of a rulemaking to establish a new risk-informed, technology-neutral licensing process with a license-by-test element, to allow innovative designs to be developed and deployed more efficiently in the longer term.

7.0 REFERENCES

1. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

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5. NUREG-1338 (Draft), "Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," U.S. Nuclear Regulatory Commission (June 1995).

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8. *Code of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.171, "Finality of Manufacturing Licenses; Information Requests," U.S. Nuclear Regulatory Commission.

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SMR Utilization of Inspections, Tests, Analyses, and Acceptance Criteria

Lead Writer – Robert ("Budd") Haemer

American Nuclear Society (ANS)

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1.0 INTRODUCTION/BACKGROUND

In the 1980s, recommendations from a task force led by the U.S. Nuclear Regulatory Commission (NRC) suggested streamlining licensing by creating an alternate to the two-stage licensing process for nuclear plants and implementing a single licensing proceeding, to be held prior to construction, in which detailed design plans were to be considered and approved. These recommendations proposed that once a license was granted, jurisdiction to oversee construction and confirm that the plant is constructed consistently with the design plans should be placed with the NRC staff. The crux of these recommendations was to ensure that the plant was constructed consistently with design plans, to promote standardization. It was further recommended that applications for final design approval and Design Certification (DC) should "define the tests, inspections, analyses, and acceptance criteria related thereto necessary to assure that the designs are properly installed in the plant." [1986 Atomic Industrial Forum ("AIF") Position Paper on Standardization. See NRC SECY-02-0067 regarding Programmatic ITAAC, Attachment 2 at 2 (Apr. 15, 2002) (NRC ADAMS Accession No. ML020700641).] In 1987, the NRC announced its intent to standardize nuclear power plants and implement a one-step licensing process that would "give licensees greater assurance that if the facility is constructed in accordance with the terms of the application/permit, it will be permitted to operate once construction is complete." [Policy Statement on Nuclear Power Plant Standardization. 52 Fed. Reg. 34,884, 34,885 (Sept. 15, 1987).] The revised licensing process was codified in 10 CFR 52 (Ref. 1), including changes through 2007 that considered participants' comments and incorporated lessons learned specific to the initial experience with the licensing process by the large reactor projects.

Each reactor vendor has the option to petition for a rulemaking to obtain a DC rule that would cover the criteria necessary for design and construction of the plant; quality assurance programs; and whatever Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) are necessary to assure that the plant is built within the certified design specifications. [See generally i.d.; 10 CFR 52.47(b)(1).] Thereafter, each specific facility seeks a license that references a DC rule, such as a Combined Construction and Operating License (COL). This license would incorporate the standard ITAAC plus any plant-specific ITAAC that must be demonstrated to allow operation.

Operation after construction under a COL is allowed pursuant to 10 CFR 52.103(g) (Ref. 2) if the NRC finds that the acceptance criteria in the COL have been met. The ITAAC included with, or referenced by, a COL Application (COLA) must be sufficient to demonstrate that the facility has been constructed and will operate in conformity with the COL and the NRC regulatory requirements. Generally, in addition to site-specific ITAAC, a COLA incorporates standard ITAAC from the DC—although an applicant can take departures or exemptions from the DC ITAAC if the changes have benefits that outweigh the benefits from standardization. The ITAAC serves as the primary source of acceptance criteria to be applied at the end of construction. As such, the ITAAC must include all significant issues that require resolution before fuel loading.

The NRC ITAAC review and inspection process provides confidence that the licensee's ITAAC completion and verification processes are effective and thereby gives reasonable assurance that the licensee's ITAAC completion notifications to the NRC are sufficient and accurate to provide reasonable assurance that operation of the reactor will be consistent with public health and safety. This paper addresses the potential issues that may arise because the construction sequence and other commercial considerations specific to Small Modular Reactors (SMRs) were not considered when the existing 10 CFR 52 processes were codified. The current ITAAC licensing regime, created in an effort to standardize large, utility-built reactors assembled in their final in-place location, may not address the scope to reflect assembly line construction for the range of potential applications of the SMR plants, beyond those of traditional utility owners and operators.

2.0 PROBLEM/ISSUE STATEMENT

10 CFR 52.99(c)(1) (Ref. 3) states that a "licensee shall notify the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met." (This notification must contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. 10 CFR 52.99(c)(1).) 10 CFR 52.99(c)(2) provides that, "[i]f the licensee has not provided, by the date 225 days before the scheduled date for initial loading of fuel, the notification required by 10 CFR 52.99(c)(1) . . ., the licensee shall notify the NRC that the prescribed inspections, tests, or analyses for all uncompleted ITAAC will be performed and that the prescribed acceptance criteria will be met prior to operation."

This paper addresses the following key issues/questions:

1. Because closure of ITAAC is only prescribed with respect to operation under a COL, how does one address the ITAAC requirements outside the scope of a COL? For example, who and when are ITAAC closed during construction under a Manufacturing License (ML) or in the instances in which the ITAAC closure is pursued in an Operating License (OL) application?

2. How does one sequence or phase 10 CFR 52.99(c)(1) findings? What is the process for making these findings while the SMR is manufactured and assembled? Should these findings be made by the manufacturer before the SMR is provided to the operating licensee?

3. Considering that an SMR may be assembled in 14 months (420 days), what is the approach for preparing and determining the content of a 10 CFR 52.99(c)(2) submittal 225 days before fuel load, especially because this may be shortly after start of construction of the SMR plant? If the 10 CFR 52.99(c)(2) findings are simply predicting/asserting that the plant is going to be built in accordance with the license, what is the difference between the predictive finding pursuant to 10 CFR 52.99(c)(2) and the ITAAC identified in an ML.

4. When considering the use of ITAAC for evaluating multi-modular plants, can the ideas of safety and efficiency be properly balanced via the use of sampling? Or, will each test have to be repeated or repeatedly reviewed for each reactor?

3.0 DISCUSSION AND ACTUAL WORK

1. REVISIONS: ITAAC REQUIREMENTS OUTSIDE THE SCOPE OF A COLA

10 CFR 52.97(c) (Ref. 4) requires that a COLA include the ITAAC that are necessary and sufficient to demonstrate that a specific facility has been constructed and will operate in conformity with the COL; the Atomic Energy Act of 1954, as amended; and the NRC's regulations. 10 CFR 52.103(g) requires that the NRC find that the acceptance criteria in the COL have been met before a facility is authorized to operate. The ITAAC are the primary source of acceptance criteria. As such, it is essential that the ITAAC include all significant issues that require resolution before fuel loading. The regulations provide for NRC acceptance of ITAAC under a COL and as part of the final approval to operate. However, two issues are not addressed by the regulations: (1) how ITAAC could be accepted under licenses other than by amending a COL and (2) how ITAAC are addressed in proceedings for other types of licenses, such as OLs and MLs.

1. Because the NRC acceptance of ITAAC is covered only with respect to operating under a COL, the industry is left without a form of acceptance criteria outside the scope of a COL proceeding. The industry may find alternatives to a COL more appropriate. Two examples are where (a) the operator is overseas (the SMR is for export) and (b) the operator holds an Early Site Permit (ESP), including Limited Work Authorization (LWA) to allow site preparation work and the SMR is built to an ML.

a. For instance, if an SMR is exported, ITAAC will not apply as the NRC approach is currently unique among international regulators. The operator would be subject to local government regulation. A method to harmonize ITAAC with the local regulatory approach would facilitate both safety and efficiency in accepting an exported SMR for operation.

b. It is also feasible that those seeking to build SMRs would seek to obtain MLs and those seeking to deploy SMRs would seek to obtain an ESP (as a partial Construction Permit). The ESP option would allow the operator flexibility to choose among differing SMR designs that fit within the envelope of its ESP. The combination of an ESP with an OL would afford operators flexibility to maximize their ability to obtain an SMR on commercially reasonable terms by deferring the technology decision to as late in the process as possible. Because ITAAC is not a requirement for an OL, a process for accepting ITAAC as completed as part of issuing or transferring an OL would facilitate such potentially commercially viable competitions. Closure of ITAAC is only well defined for plants constructed under a COL, restricting the potential safety and standardization benefit in licensing approaches that do not rely on a COL.

2. 10 CFR 52.80(a)(3) (Ref. 5) allows a COL applicant to include with the application a notification that a required inspection, test, or analysis has been successfully completed and that the corresponding acceptance criterion has been met. If such notification is included with the application, those ITAAC will be identified in the notice of hearing on the application. Timing issues arise. The COLA may be under review while the SMR is under construction. COLA processing times exceed the expected construction time for an SMR on its assembly line. The time to issue a COL could be further extended if the COLA is frequently revised to reflect ITAAC closure during the manufacturing. However, the regulations may not provide an efficient mechanism for the manufacturer to gain NRC acceptance of ITAAC during SMR assembly.

   Furthermore, logistical problems, not limited to the SMR context, arise regarding the lack of ITAAC closure process prior to approval of the COL. While many reactors potentially face some issues arising from fabrication of long lead components—like the reactor vessel or steam generator application prior to a COL—the SMR vendor may have the SMR essentially fabricated before the COL is finalized. An example of the potential administrative complexity that can arise is the Dominion North Anna Unit 3 experience with Economic Simplified Boiling Water Reactor (ESBWR) reactor vessel fabrication. In 2007, Dominion Virginia Power (Dominion) submitted a COL application to the NRC for an ESBWR. Dominion had partnered with GE Hitachi Nuclear Energy and Bechtel Corporation to build the multimillion-dollar ESBWR reactor vessel. In 2010, Dominion selected an alternate technology to an ESBWR. (On May 10, 2010, a World Nuclear News press release announced that Dominion had selected Mitsubishi Heavy Industries' ("MHI's") Advanced Pressurized Water Reactor ("APWR") for the potential third unit at its North Anna nuclear power plant in Virginia. Available at http://www.world-nuclear-news.org/print.aspx?id=27686.) Some paperwork has been prepared to support the closure of ITAAC related to the reactor vessel. If the reactor vessel is resold to another customer, full value can be obtained only if the documentation prepared to date is fully transferable. If closure of the ITAAC by the vendor is allowed, there will be greater certainty in reselling the reactor vessel. If a similar case arose for an in-process SMR, the in-process documentation could be more extensive and lead to even greater due diligence effort for the ultimate customer compared to a process where the vendor could complete the ITAAC.

2. 10 CFR 52.99(c)(1) AND (10 CFR 52.99(c)(2) REQUIREMENTS

Regulations regarding the inspections to be conducted during plant construction, i.e., after the COL is issued and before the completed facility is allowed to load fuel, engender questions regarding how these inspections are to be completed. 10 CFR 52.99(c) ensures that the NRC will have sufficient information to complete all of the activities necessary to determine whether all of the ITAAC have been, or will be, met prior to the initial operation and that sufficient notice will be given to interested persons on both completed and uncompleted ITAAC so that they can decide whether to request a hearing on compliance with the acceptance criteria. For that reason, the information included with the notification provided under 10 CFR 52.99(c)(1) and 10 CFR 52.99(c)(2) concerning the completed and uncompleted ITAAC must be sufficient to allow judgments to be formed by reference to that information.

1. 10 CFR 52.99(c)(1) states that a "licensee shall notify the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met." The notification must contain sufficient information or, at a minimum, a summary description of the bases for the licensee's conclusion that the inspections, tests, or analyses have been performed and that the prescribed acceptance criteria have been met.

    a. In the process of assembling the SMR, the ultimate customer may change for commercial reasons or changing demand in the Owner's business. Restricting the ITAAC acceptance process introduces administrative inefficiencies if the ultimate customer changes.

    b. Some ITAAC are closed based on type testing. Such tests or analysis would not be specific to the individual SMR being assembled. Reports for a type-test–based ITAAC can be expected to cover multiple projects. Such ITAAC should not require recertification absent a design change or other changes in the procurement documents that could impact the conclusions of the type test.

2. 10 CFR 52.99(c)(2) provides that "[i]f the licensee has not provided, by the date 225 days before the scheduled date for initial loading of fuel, the notification required by 10 CFR 52.99(c)(1) . . ., the licensee shall notify the NRC that the prescribed inspections, tests, or analyses for all uncompleted ITAAC will be performed and that the prescribed acceptance criteria will be met prior to operation." This additional notification must provide sufficient information to demonstrate that the inspections, tests, or analyses will be successfully completed and that the acceptance criteria for the uncompleted ITAAC will be met, including, but not limited to, a description of the specific procedures and analytical methods to be used for performing the inspections, tests, and analyses and determining that the acceptance criteria have been met.

SMR vendors can expect to face practical scheduling issues with regard to 10 CFR 52.99(c)(2) findings. For instance, it has been estimated that an SMR may be assembled in 14 months (420 days). If assembly includes fuel load, a 10 CFR 52.99(c)(2) submittal required 225 days before fuel load may be due shortly after start of construction of the SMR plant. At that point, the COL may not have been issued, and the COL applicant may be tentative. For commercial reasons, there may be flexibility with the specific location or operator that will take delivery of a particular SMR, given demand growth or other commercial factors. Furthermore, because 10 CFR 52.99(c)(2) findings are inherent in the findings required for a DC or ML that the vendor is technically qualified, with a 10 CFR 52.99(c)(2) letter simply predicting or asserting that the SMR is going to be built in accordance with the specifications, there seems to be no difference with explanations already made by the vendor in its DC petition or its ML application.

3. MULTI-MODULAR ITAAC

The COL-specific nature of closing ITAAC under the current regulations leaves much to be desired when considering how to handle multi-modular reactor COLs. For example, if a COL covers multiple modules, the 10 CFR § 52.103(g) proceeding to allow operation would need to permit bifurcation—so that each module could start operation while awaiting the installation of the next one. Closure of ITAAC based on a sampling plan or type testing across multiple modules should be allowed. This would facilitate efficiency rather than propagating uncertainty, or associated administrative inefficiencies, by having the same verification package resubmitted for approval for sequential modules. Repetitive review of the same ITAAC closure documentation is inconsistent with the NRC goals for achieving the benefits of standardization and applying the principles of the Design Centered Working Group (DCWG) (one review of each issue one time). Much, if not almost all, ITAAC will be common among SMR designs, including multiple copies of the same design. The repetitive review of the same ITAAC closure documentation in the context of SMRs is an increasing administrative burden, particularly where the ITAAC are closed based on type testing or other sampling-based verification.

4. FREE-RIDERS ON INITIAL ITAAC DEVELOPMENT

ITAAC are largely not design specific. Many, if not almost all, tests, inspections and analyses needed to verify that a reactor will operate properly are common to all projects.

Today, those reactor designs with DC rules or undergoing NRC review for DC rulemaking have >90% of their ITAAC in common. For large reactors, the later reactor vendors with follow-on designs essentially copied much of the ITAAC from the initial DC rules.

While conceptually the ITAAC for SMR designs can be expected to be similar to other designs, it will differ from those ITAAC for large plants to reflect the construction phasing of SMRs. SMRs will have to develop ITAAC that should be complete at the assembly plant, post-transportation ITAAC, and other ITAAC revisions that would be specific to the construction of SMRs compared to large plants constructed in situ. The initial SMR manufacturer will perform the lion's share of effort redrafting large plant ITAAC to suit the special needs of SMR vendors. Other vendors will "free-ride" off this initial work, copying the ITAAC agreed to by the initial vendor and the NRC.

4.0 CONCLUSIONS/RECOMMENDATIONS

Based on the discussion above, the American Nuclear Society President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) recommends that the industry pursue a Petition for Rulemaking to provide additional flexibility and certainty to the ITAAC process for SMR projects. The regulations and procedures currently used to regulate large reactors should be modified in order to facilitate SMRs with respect to

- optimizing the role of ITAAC requirements for SMR that are not constructed and operated pursuant to an NRC-issued COL
- the vendor's ability to gain NRC acceptance of ITAAC during SMR assembly
- the vendor's ability to make the 10 CFR 52.99(c)(1) report as the agent for the ultimate customer in order to minimize administrative inefficiencies if the ultimate customer changes
- the vendor's ability to make the 10 CFR 52.99(c)(1) reports for a type-test–based ITAAC on a multiproject basis
- scheduling issues that SMR vendors can be expected to face concerning 10 CFR 52.99(c)(2) findings
- allowance for closure of the ITAAC by the vendor
- allowance for closure of ITAAC based on a sampling plan or for type testing across multiple modules.

Such revisions will enhance the effectiveness of standardization by better matching the ITAAC process to the commercial needs of SMRs manufactured and assembled for delivery essentially ready to use at a prepared site.

Specifically, the SMR Special Committee recommends the following:

1. **Allow SMR vendors to act as agent of licensees**. In the process of assembling the SMR, the ultimate customer may have changed. The ITAAC acceptance process should provide for the manufacturer to make the 10 CFR 52.99(c)(1) report as the agent for the ultimate customer in order to minimize administrative inefficiencies if the ultimate customer changes.

2. **Optimize recertification process for generic tests**. Some ITAAC are closed based on type testing. Such tests or analysis are not specific to the individual SMR being assembled. The ITAAC acceptance process should provide for the manufacturer to make the 10 CFR 52.99(c)(1) reports for a type-test–based ITAAC on a multiproject basis. Such ITAAC should not require recertification absent a design change or other procurement changes that could impact the type-test performance.

3. **Endorsement/support of DCWG/task force methodology**. A continuing effort to examine ITAAC for SMRs is needed. While 90% of all ITACC will be common among SMRs because basic demonstrations needed to allow a reactor to operate are largely design independent, SMR vendors will need ITAAC that reflect the phasing unique to SMR assembly, e.g., line manufacturing, diverse functional manufacturing methods and their effects, and ultimately consideration of specific issues for "types" of reactors, i.e., light water reactors (LWRs), high temperature gas reactors (HTGRs), and liquid metal reactors (LMRs). A DCWG should further

consider the differences between ITAAC closure processes being developed for large plants currently under licensing review and future SMR modular construction approaches—to specifically consider how these differences will affect the current ITAAC regime and any potential changes to be made via rulemaking. This evaluation should include a review of the phasing of ITAAC requirements for either MLs or OLs and possible solutions toward integration, the vendor's ability to gain NRC acceptance of ITAAC during SMR assembly, allowance for closure of the ITAAC by the vendor, and allowance for closure of ITAAC based on a sampling plan or type testing across multiple modules.

4. **Allow effective parallel working; rulemaking with FOAKE exemptions**. A continuing Technical Working Group/DCWG should also propose establishing interim guidance, to be submitted to the NRC for consideration as rulemaking, to help ensure that first-of-a-kind-engineering (FOAKE) for SMRs is defined early—so that the SMR designs proceed through the regulatory process with transparency and certainty. Additionally, it may be appropriate to provide exemptions to some early movers to facilitate FOAKE. (Rulemaking with FOAKE exemptions would also enable methods that support proper and effective export via new use of 10 CFR 110/10 CFR 810 (Refs. 6 and 7) permits and MLs. These methods could serve as the first steps toward harmonizing foreign regulatory approaches with respect to export of nuclear technology.)

5.0 APPENDIX

DRAFT RULE LANGUAGE—INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC) MAINTENANCE PROVISIONS

Recently, the NRC made available the draft wording of a proposed amendment to requirements in 10 CFR 2.340 (Ref. 8) and 10 CFR 52.99 related to verification of nuclear plant construction activities through ITAAC under a combined license. The NRC's proposed new rules would require (1) licensee reporting of new information raising a reasonable concern that a prescribed inspection, test, or analysis was not performed as required, or that a prescribed acceptance criterion is not met; (2) licensee documentation of the basis for all ITAAC notifications; and (3) licensee notification of completion of all ITAAC activities. The NRC's proposed changes would also correct existing language in 10 CFR 2.340 and 10 CFR 52.99 for consistency with other sections in 10 CFR 52 and with language in the Atomic Energy Act, as amended. With respect to 10 CFR 52.99(c)1 and 10 CFR 52.99(c)(2) specifically, the proposed changes amount to a change in verb tense, so that the licensee notifies the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria *are* met.

6.0 REFERENCES

1. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

2. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 103, "Operation Under a Combined License," U.S. Nuclear Regulatory Commission.

3. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.99, "Inspection During Construction," U.S. Nuclear Regulatory Commission.

4. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.97, "Issuance of Combined Licenses," U.S. Nuclear Regulatory Commission.

5. C*ode of Federal Regulations*, Title 10, "Energy," Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Sec. 52.80, "Contents of Applications; Additional Technical Information," U.S. Nuclear Regulatory Commission.

6. C*ode of Federal Regulations*, Title 10, "Energy," Part 110, "Export and Import of Nuclear Equipment and Material," U.S. Nuclear Regulatory Commission.

7. C*ode of Federal Regulations*, Title 10, "Energy," Part 810, "Assistance to Foreign Atomic Energy Activities," U.S. Department of Energy.

8. C*ode of Federal Regulations*, Title 10, "Energy," Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," Sec. 2.340, "Initial Decision in Certain Contested Proceedings; Immediate Effectiveness of Initial Decisions; Issuance of Authorizations, Permits, and Licenses," U.S. Nuclear Regulatory Commission.

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