The enduring legacy of ACRS: Reviewing safety-licensing to protect the public

Since 1957, the Advisory Committee on Reactor Safeguards has had a continuing statutory responsibility for providing independent reviews of, and advising on, the safety of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards in the United States.

By Hossein Nourbakhsh

The 1957 amendment to the Atomic Energy Act of 1954 established the Advisory Committee On Reactor Safeguards as a statutory committee with an independent advisory role and the responsibility to "review safety studies and facility license applications" and advise the U.S. Atomic Energy Commission "with regard to the hazards of proposed or existing reactor facilities and the adequacy of reactor safety standards." With the enactment of the Energy Reorganization Act of 1974, the ACRS was assigned to the newly established Nuclear Regulatory Commission with its statutory requirements intact.

The ACRS consists of up to 15 members who are well-recognized experts in technical areas that are key to nuclear safety and who have a breadth of experience in all aspects of nuclear enterprise: industry, universities, national laboratories, and government.

Throughout its history, the ACRS review has been an important element of the reactor licensing process. The committee's licensing reviews have led to evolution of many new safety requirements and design changes dealing with a wide range of technical issues. As the NRC is preparing for review of new reactor designs that are radically different from the current fleet of light water reactors, the ACRS, with its diverse technical expertise, will continue to be essential for integrated, multidisciplinary independent review and advice. The following is a brief history of the ACRS, with particular attention given to some of the committee's significant contributions to reactor safety, followed by a historical perspective on ACRS reactor licensing reviews. The essential role of the ACRS in reviewing the new, advanced non-LWR designs is also discussed.

Creation of the ACRS

The history of the ACRS goes back to 1947, when the AEC, soon after its establishment, recognized the need for an independent technical group to review and provide advice on reactor safety matters. Thus, the Reactor Safeguard Committee, chaired by Dr. Edward Teller (from 1947 to 1953) was established. Teller has been quoted saying that the Reactor Safeguard Committee "was about as popular—and as necessary—as a traffic cop."[1] According to former NRC chairman Richard Meserve, the committee "clearly established an enduring characteristic of the ACRS—a willingness to provide candid views on reactor safety issues, even at the risk of taking unpopular positions."[2]


Teller

In 1950, the AEC established a second advisory committee, the Industrial Committee on Reactor Location Problems, charged with advising on what we would today consider to be siting issues, including seismic and hydrological characteristics of proposed sites. In 1953, the Reactor Safeguards Committee and the Industrial Committee on Reactor Location Problems were combined by the AEC into one entity, the ACRS, first chaired by Dr. C. Rogers McCullough (from 1953 to 1960).


McCullough

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The licensing of Fermi-1, the world's first commercial liquid metal fast breeder reactor, was at the center of a controversy that ultimately resulted in the establishment of the ACRS as a statutory committee.

A statutory committee

The establishment of the ACRS in 1957 as a statutory committee advising the AEC was in part the result of a controversy involving licensing of the Fermi-1 liquid metal fast breeder reactor. NRC historians J. Samuel Walker and Thomas R. Wellock discussed the incident in their *Short History of Nuclear Regulation, 1946–2009.*[3]

In 1956, the Power Reactor Development Company (PRDC), a consortium of utilities led by Detroit Edison Company, applied for a construction permit to build a fast breeder reactor on Lake Erie, within 30 miles of both Detroit, Mich., and Toledo, Ohio. The fast breeder reactor that the PRDC was planning was far more advanced in its technological complexity than the LWR designs proposed in earlier applications. After a review of the application and discussions with company representatives, the ACRS concluded in an internal report to the AEC that there was insufficient information available to give assurance that the PRDC reactor could be operated without public hazard. The ACRS also expressed doubt that safety concerns could be resolved within the PRDC's proposed schedule for obtaining an operating license. The committee urged the AEC to expand its experimental programs with fast breeder reactors to seek more complete data on the issues raised in the PRDC application.

During congressional hearings, members of the Joint Committee on Atomic Energy (JCAE, the AEC congressional oversight committee) were troubled by revelations of safety concerns and the AEC chairman's intention to attend the groundbreaking ceremony for a reactor for which the construction permit was still under evaluation by the AEC. They were particularly disturbed by the AEC's failure to inform them about the ACRS's reservations. The AEC was obligated by the Atomic Energy Act to keep the JCAE "fully and currently informed" about its activities, and JCAE members believed that, in the case of the ACRS report, the commission had failed to carry out its responsibility.

The AEC was unwilling to provide a copy of the ACRS report to the JCAE without the condition that it would be kept "administratively confidential" and refused to provide a copy to the state of Michigan on the grounds that "it would be inappropriate to disclose the contents of internal documents." Meanwhile, the commission was completing its review of the PRDC's application. It took a more optimistic view of the safety of the proposed reactor than had the ACRS. Since the PRDC had agreed to perform tests in an effort to answer questions raised by the ACRS, the AEC decided to issue the construction permit. However, it acknowledged the ACRS's concerns by inserting the word "conditional" in the permit to emphasize that the company would have to resolve the uncertainties about safety before an operating license would be issued.

To prevent a recurrence of the AEC's conduct in the PRDC case, the JCAE introduced legislation to establish the ACRS as a statutory body, directing that its reports on licensing cases be made public and requiring public hearings on all reactor applications. These actions were accompanied by a significant expansion of public access to the regulatory and licensing activities of the AEC.

ACRS through the years

The role of the ACRS has evolved over its lifetime. Early licensing reviews were generally based on the engineering experience and judgment of regulatory staff working closely with the ACRS—but without the availability of the regulatory guidance and structure established in later years, during commercial development of LWRs. Most of today's U.S. nuclear power plants were licensed during the 1960s and 1970s when both the technology and its governing regulations were in the formative stages. The committee's licensing reviews led to the evolution of many new safety requirements and design changes dealing with a wide range of technical issues.

Early in the development of commercial nuclear power, the ACRS became concerned with core meltdown accidents, particularly one in which the plant's emergency core cooling system (ECCS) might fail to operate as designed, leading to a breach of containment. In 1966, after prodding by the ACRS, the AEC established a special task force to look into the problem of core meltdown. The task force, chaired by former ACRS member William K. Ergen, issued its report in October 1967. The report offered reassurances about the reliability of ECCS designs and the improbability of a core meltdown, but it also acknowledged that a loss-of-coolant accident (LOCA) could cause a breach of containment if the ECCS failed to perform. In a letter dated February 26, 1968, the committee strongly recommended that a "positive approach be adopted toward studying the workability of protective measures to cope with core meltdown" and also that a "vigorous program be aimed at gaining better understanding of the phenomena and mechanisms important to the course of large-scale core meltdown." The task force report and ACRS recommendations formed the basis of some of the most important research initiatives and regulatory decisions by the AEC and NRC, including the AEC's decision to undertake a study to estimate severe accident probability, which resulted in the publication of the landmark *Reactor Safety Study* (WASH-1400) and the beginning of the science of probabilistic risk assessment (PRA) as applied to nuclear power plant safety.

As the ACRS moved into the 1980s, it shifted much of its attention from plant design and construction to improvements in both the operation and regulation of nuclear power plants—a focus it maintains today. The ACRS has made valuable contributions over a wide range of issues at operating plants, including fire safety, operator training and human performance, digital instrumentation and control upgrades, extended power uprates, plant aging, and license renewal.

The ACRS has made significant contributions toward resolution of many generic safety issues, one example being its role in the resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." The ACRS was first to express concerns about the effects of chemical reaction products and particle/fiber mats that could form on screens. The committee was also the first to recognize that increasing screen area, though it could reduce head loss, might result in more fiber debris passing through the screens and increasing downstream effects.

The ACRS was also at the forefront of the development of quantitative safety goals. In a May 16, 1979, letter on the subject, the ACRS recognized the difficulties and uncertainties in the quantification of risk and acknowledged that in many situations, engineering judgment would be the only or at least the primary basis for making a decision. Nevertheless, the committee believed that the existence of quantitative safety goals and criteria could provide important yardsticks for such judgment. *An Approach to Quantitative Safety Goals for Nuclear Power Plants* (NUREG-0739) was developed by the ACRS in 1980. This first set of trial goals formed the basis for later NRC work on the development of an NRC Safety Goal Policy.

In the early 1990s, the ACRS became concerned about the inconsistent use of PRA in the NRC. In a July 19, 1991, letter, the ACRS acknowledged, "PRA can be a valuable tool for judging the quality of regulation, and for helping to ensure the optimal use of regulatory and industry resources." The committee also stated that it "would have liked to see a deeper and more deliberate integration of the methodology into the NRC activities" and pointed to issues such as the inconsistent use of conservatism and the lack of the treatment of uncertainties. In response to the ACRS, the NRC chartered a PRA Working Group and a Regulatory Review Group to review processes, programs, and practices to identify the feasibility of substituting performance-based requirements and guidance founded on risk insights in place of prescriptive requirements. These efforts led the NRC to issue a policy statement on the use of PRA so that the many potential applications of PRA could be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. The ACRS has been very supportive of the evolution toward a risk-informed, performance-based regulatory system and has taken a leading role in considering some of the challenging issues that have arisen in this effort.

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Since the beginning, an essential activity of the ACRS also has been to review commission-sponsored research. This includes the evaluation of technical and programmatic aspects of the overall reactor safety research program as well as episodic review of particularly important ongoing research.

Prior to the establishment in 1988 of the Advisory Committee on Nuclear Waste and Materials (known from 1988 to 2007 as the ACNW), the ACRS reviewed matters related to the long-term management of radioactive waste produced by the nuclear industry. In 2008, the two committees merged, and since then the ACRS has resumed reviewing many aspects of nuclear waste management, such as the handling, processing, transportation, and storage of nuclear wastes, including spent fuel and nuclear wastes mixed with other hazardous substances.

ACRS reactor licensing reviews: A historical perspective

Early reviews

The passage of the Atomic Energy Act made it possible for private companies to build and operate nuclear reactors under license and also assigned to the AEC the responsibility of protecting the health and safety of the public through a licensing process.

Early power reactors approved for construction included small prototypes developed by the AEC in cooperation with electric utilities and reactor manufacturers. During the ACRS licensing review of early power reactors, there were considerable discussions about site selection, as the proposed plants had to rely more on containment than isolation as a means of protecting the public against the potential consequences of reactor accidents.

In the early years of nuclear power plant development, both the technology and its governing regulations were in the formative stages. In the early 1960s, the AEC began defining a standard regulatory prescription to licensing of nuclear reactors. After publication of the proposed 10 CFR 100, *Reactor Site Criteria*, in 1961, several high-power reactors were proposed for construction, including San Onofre (a 1,347-MWt PWR), Connecticut Yankee (a 1,473-MWt PWR), and Oyster Creek (a 1,600-MWt BWR). The principal focus of the ACRS licensing reviews of early large LWRs appears to have been on the efficacy of the engineered safeguards (containment plus sprays and/or filters) needed to meet the dose guidelines of 10 CFR 100. These early licensing reviews also led to evolution of many new safety requirements dealing with a wide range of technical issues. Below are some examples of the issues raised by the ACRS during this time.

■ **Control rod ejection accidents.** The licensing review report for the Connecticut Yankee plant in 1964 was the first to call out a requirement for a study of control rod ejection accidents. This led to design changes in large LWRs, either to limit the reactivity worth of control rods or to add an additional mechanical restraint to control rod ejection (an approach taken in boiling water reactors).

■ **Design considerations for a tsunami following a major earthquake.** The 1964 ACRS report on the proposed 1,473-MWt Malibu Unit 1 at Corral Canyon, 29 miles west of Los Angeles, was the first to raise the issue of adequate protection against a tsunami following a major seismic event. The following paragraph is particularly noteworthy:

> The ability of the plant to withstand the effects of a tsunami following a major earthquake has been discussed with the applicant. There has not been agreement among consultants about the height of water to be expected should a tsunami occur in this area. The Committee is not prepared to resolve the conflicting opinions and suggests that intensive efforts be made to establish rational and consistent parameters for this phenomenon. The applicant has stated that the containment structure will not be impaired by inundation to a height of fifty feet above mean sea level. The integrity of emergency in-house power supplies should also be assured by location at a suitable height and by using water-proof techniques for the vital power system. The emergency power system should be sized to allow simultaneous operation of the containment building spray system and the recirculation and cooling system, and the ability to remove shutdown core heat under conditions of total loss of normal electrical supply should be assured. If these provisions are made, the Committee believes that the plant will be adequately protected.

The proposed construction of the plant was successfully contested, and the Malibu reactor was never built. The adequacy of seismic design was one of the main points of contention.

■ **Effectiveness of ECCS design.** By the mid-1960s, as proposed plants increased significantly in power level, the ACRS became concerned that a core meltdown accident, particularly one in which the plant's ECCS might fail to operate as designed, could lead to a breach of containment. The committee emphasized the

Dresden-2 (pictured), proposed for construction in 1965, represented a large jump in power level, compared with Oyster Creek, then the largest reactor previously approved for construction, and therefore received much attention from the ACRS. (Photo: Exelon)

need for improved emergency core cooling systems. By August of 1966, General Electric responded in support of Dresden-3 by proposing a redundant core flooding system and an automatic depressurization system, which would reduce the primary system pressure sufficiently to maximize the effectiveness of the low-pressure core spray or core flooding system. Later that year Westinghouse introduced accumulators.

■ **Emergency planning.** As the size of proposed nuclear power plants increased and containment could no longer be regarded as an unchallengeable barrier to the escape of radioactivity, the ACRS paid more attention to emergency planning. Pressed by the ACRS in 1966, the AEC undertook a study of emergency plans and procedures that eventually led to adding a new appendix to 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*—Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

■ **Reactor pressure vessel integrity.** Concerns regarding reactor pressure vessel integrity date back to 1961, when the ACRS first raised the issue of potential damage to the reactor pressure vessel as a result of lifetime exposure to neutron flux. In 1965, concern was growing, in part due to the 1964 failure of a very large heat exchanger at a temperature near the nil ductility temperature during testing by the Foster Wheeler Corporation. During the Dresden-2 licensing review, the ACRS discussed this matter extensively and debated whether this issue should be handled in a generic way or by specifically addressing Dresden-2. The committee finally decided to issue a report favorable to the construction of the reactor and at the same time wrote a general report regarding reactor pressure vessels.[4]

■ **Anticipated transients without scram (ATWS).** The issue of ATWS was first raised by an ACRS consultant in January 1969. Soon after, the ACRS addressed the issue in its report on Hatch-1 and in response to the application for the construction permits for Brunswick Units 1 and 2. In each report, the committee recommended a study be undertaken by the applicant as a further means of preventing common failure modes from negating scram action and of using design features to make tolerable the consequences of failure to scram during anticipated transients.

Review of non-LWR designs

The ACRS has a long history of review and evaluation of non-LWRs. Early safety reviews were highly customized and generally were based on the engineering experience and judgment of the regulatory staff working closely with the ACRS, without the availability of regulatory guidance and structure (which was not established until later, during LWR commercial development). In later reviews, explicit use was made of LWR regulatory guidance where applicable. Early non-LWRs approved for construction up through 1960 included Fermi-1 (a 300-MWt liquid metal fast breeder reactor), Piqua (a 48-MWt organically cooled and moderated reactor), Carolina-VA (a 63-MWt pressurized tube heavy water reactor), Hallam (a 240-MWt sodium-cooled, graphite-moderated reactor), and Peach Bottom-1 (a 115-MWt high-temperature, gas-cooled reactor).

Prior to the 1986 policy statement on the regulation

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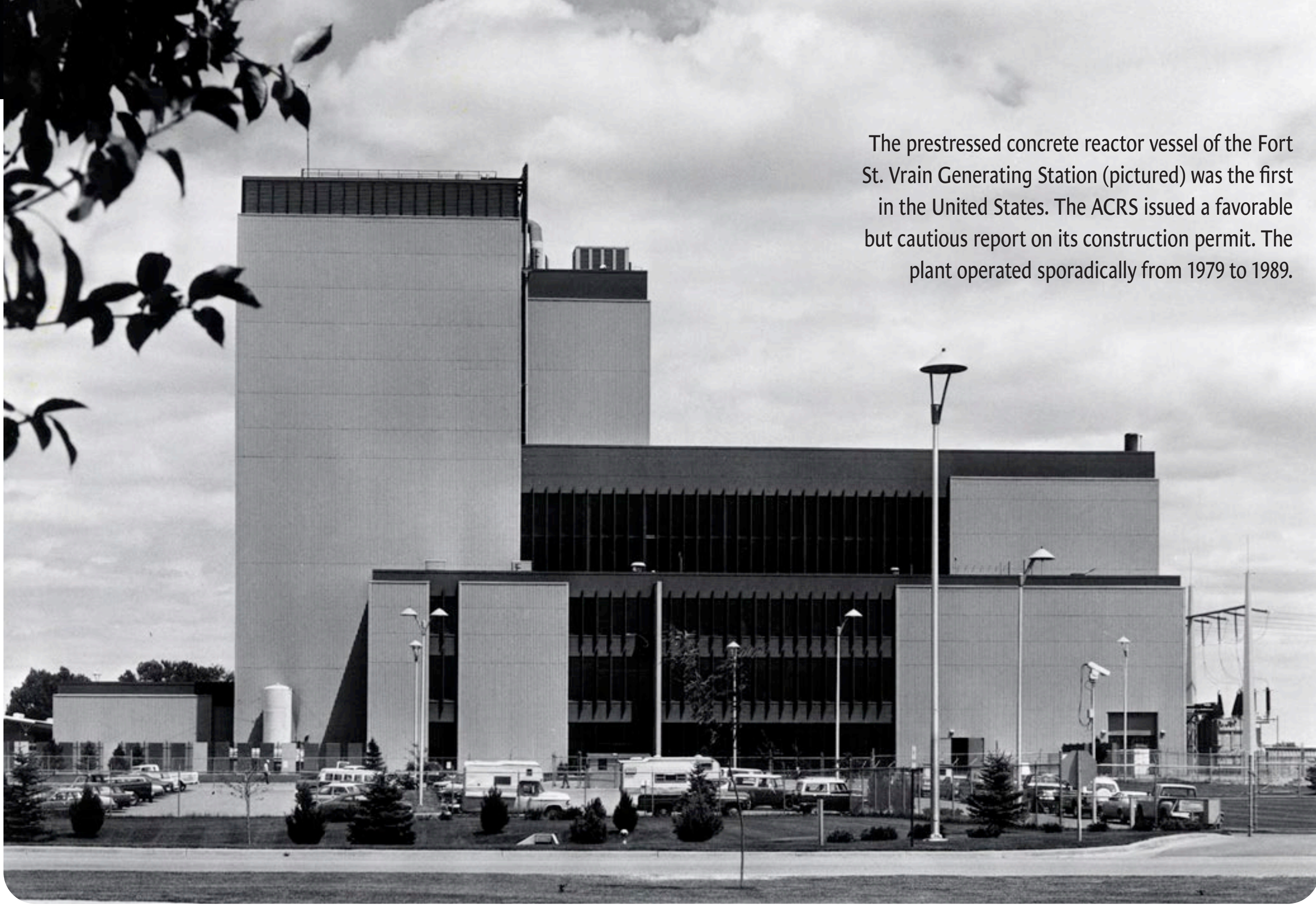


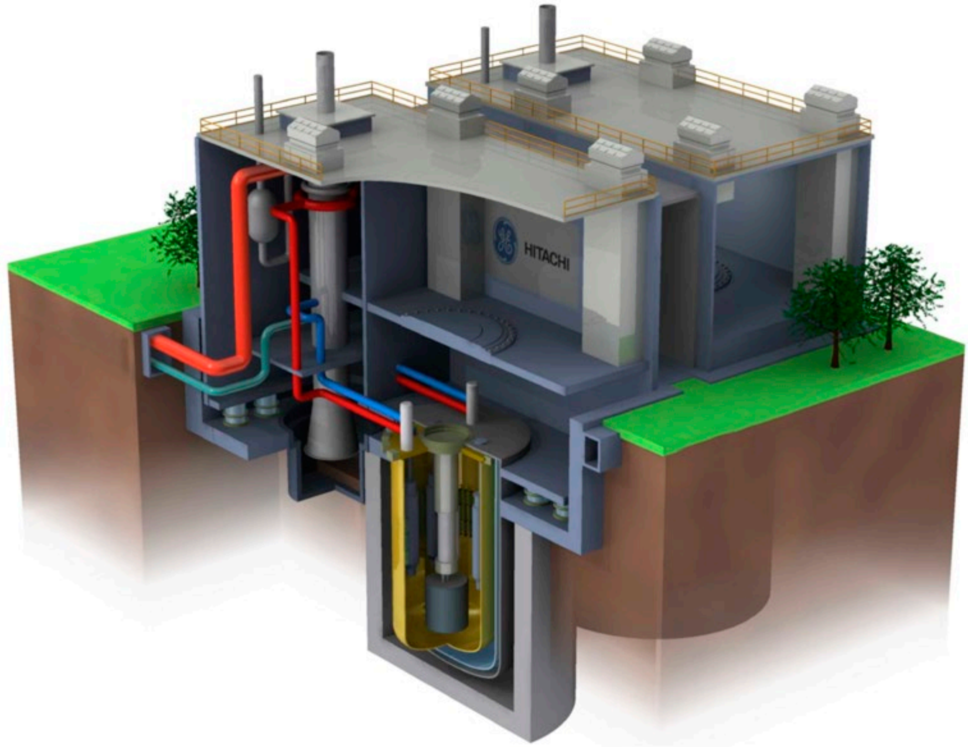
The prestressed concrete reactor vessel of the Fort St. Vrain Generating Station (pictured) was the first in the United States. The ACRS issued a favorable but cautious report on its construction permit. The plant operated sporadically from 1979 to 1989.

of advanced reactors, the principal statement on non-LWR review policy was given in the introduction to 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The introduction states, "The General Design Criteria [GDC] are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units." The AEC's regulatory staff worked closely with the ACRS in developing the GDC, versions of which were publicly available as early as 1965 before being incorporated into 10 CFR 50 in February 1971. Development of the GDC led to the "comparable level of safety" philosophy under which proposed HTGRs and liquid metal reactors were reviewed in later years. Examples of those non-LWRs included Fort St. Vrain (an 842-MWt HTGR) and the Fast Flux Test Facility (a 400-MWt sodium-cooled fast reactor).

In 1986, the Commission issued its Statement on the Regulation of Advanced Nuclear Power Plants. Advanced reactors were to include evolutionary LWRs, non-LWRs, and small modular LWRs. The NRC's policy statement has encouraged early interaction (prior to a license application) between vendors and the NRC "to provide for early identification of regulatory requirements for advanced reactors, and to provide all interested parties, including the public, with a timely, independent assessment of the safety characteristics of advanced reactor designs." The NRC has been particularly interested in any regulatory issues that could lead to the need for commission policy decisions or technical issues unique to the design that could require extensive effort and a long lead time to resolve. Consistent with the NRC policy statement, the ACRS has been holding preapplication meetings and discussions to familiarize itself with the design under consideration and to identify topics for more detailed discussions before the application is submitted. ACRS preapplication reviews of non-LWR designs during the late 1980s and early 1990s included the modular high-temperature, gas-cooled reactor (MHTGR), the sodium advanced fast reactor (SAFR), the Power Reactor Innovative Small Module (PRISM), and the Toshiba 4S ("super-safe, small, and simple").

Review of advanced LWRs

In 1989, the NRC established alternative licensing processes to improve regulatory efficiency and add greater predictability. The new processes included combining a construction permit and operating license—with certain conditions—into a single license. Other licensing alternatives established in 1989 were early site permits, which allowed an applicant to obtain approval for a reactor site for future use, and certified standard plant designs, which could be used as preapproved designs. The ACRS played an important role in design certification processes by providing an independent review of the determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the NRC's



Power Reactor Innovative Small Module (PRISM): The NRC conducted a thorough review of the 475-MWt design between 1986 and 1994. Consistent with the commission's advanced reactor policy, the NRC staff, to the extent feasible, used existing regulations at the time to formulate criteria and procedures for review of this design. The ACRS reviewed the staff's preliminary findings. After the review, several revisions to the conceptual design were made. (Image: NRC)

regulations. To date, the ACRS has reviewed design certification applications for the advanced boiling water reactor (ABWR), the System 80+, the Advanced Passive 600 (AP600), the AP1000, the economic simplified boiling water reactor (ESBWR), the advanced power reactor 1400 (APR1400), and the NuScale small modular reactor, among others.

The ACRS has identified many technical issues during design certification reviews, which have been resolved before the final recommendations for approvals have been provided. For example, the committee's review of NuScale's design certification application led to design and setpoint changes to the NuScale Power Module to mitigate the effects of boron dilution in the downcomer for uncontrolled passive cooling events.

Looking ahead to new advanced reactor designs

ACRS's essential role

Since the inception of the ACRS, the committee's independent reviews have been an essential element of the reactor licensing process. In a 1956 letter from then AEC chairman Willard Libby to Sen. Bourke Hickenlooper of the JCAE, written in response to a letter Hickenlooper sent to the ACRS a month prior on the question of public safety of nuclear reactors, Libby stated:

> The financial incentive of the owners of the reactor to take all steps necessary to protect their investment, as well as to decrease their potential public liability, and the legal and moral responsibilities of the Commission to protect the public from overexposure to radioactivity, are resulting in a system which is characterized by an attitude of caution and thoroughness of evaluation unique in industrial history. Every phase of the reactor design and operating procedure is reviewed separately and as a part of the whole. The inherent nuclear, chemical, metallurgical, physical, and mechanical characteristics of the fuel, moderator, coolant, neutron absorbers, and structural materials are carefully considered . . . to assure that the probability of an operating mishap has by adequate design and operating precautions been brought to an acceptably low level. [5]

As the NRC staff strategizes to assure that the commission is ready to review potential licensing applications for new advanced reactor designs, the role of the ACRS, with its diverse technical expertise, will continue to be essential for an independent, integrated, multidisciplinary review. The committee's role is particularly important since new advanced reactor designs, currently under development, pose new challenges for safety-licensing reviews due to the following:

- First-of-a-kind designs with a variety of coolants, fuel forms, and innovative configurations.
- Designs that do not have the same levels of operating and regulatory experience as that of LWRs.
- Limited experimental database and validation.
- Implementation of a new licensing approach.

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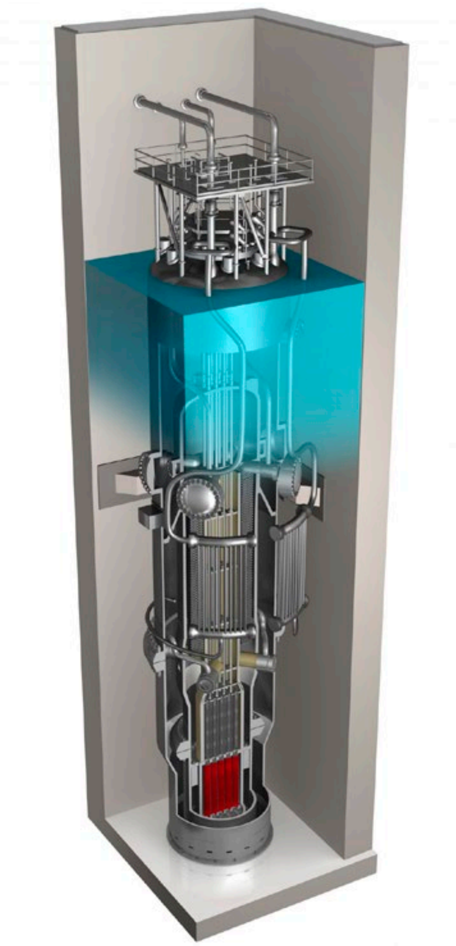
Enhancing efficiency

The ACRS continues to perform introspective evaluations to identify ways to improve its own effectiveness and efficiency, as the NRC accomplishes its mission within a changing regulatory framework and culture and in an expanding industry environment. Over the past several years, the ACRS has also collaborated with NRC program offices to achieve greater efficiencies while maintaining its independence.

As part of a continuing effort to become more effective and assist the NRC in its transformation initiatives, the ACRS recently conducted a self-assessment based on its observations and lessons learned from the recent NuScale design certification and standard design approval application reviews. It was also informed by prior design certification and early site permit reviews and interactions with the NRC staff. Observations and lessons learned from the ACRS self-assessment led to several recommendations by the committee that could improve future NRC reviews of advanced reactor designs.[6] For a more effective and expeditious review, the ACRS has adopted a cross-cutting approach, focusing on key safety-significant design issues. It is expected that this will streamline reviews, resulting in more efficiency and shorter schedules.

Ultimately it is the completeness and quality of a license application and associated supporting documents that significantly impacts the efficiency of the review process (for both the NRC and the ACRS). The following are some of the desired attributes that would improve the quality and completeness of future advanced reactor design applications:

Completeness of the design: Design completeness has a profound impact on the efficiency of the review process. Proposed new reactor designs should be sufficiently complete to demonstrate that all structures, systems, and components important to safety are appropriately identified, designed, and tested to be commensurate with their functions and to provide adequate defense-in-depth. Without an "essentially complete design" and a completed detailed component and system analysis, it may be difficult for the NRC to make a technically sound finding on any requested deviation (exemption) from historical regulatory requirements (e.g., GDC). Design changes during the review process may also adversely impact the efficiency of the review process.



The ACRS identified several technical issues during its certification review of the NuScale SMR (pictured), which led to design and setpoint changes to the NuScale Power Module. (Image: NuScale)

Comprehensiveness of knowledge base: All safety decisions, either explicitly or implicitly, are based on identifying radiological hazards and addressing the "risk triplet" questions: "What can go wrong?" "How likely is it?" and "What are the consequences?" The NRC addresses these three questions through the body of its regulations and guidance. The comprehensiveness of the knowledge base (experimental data, operational experience, relevant analyses, etc.) to support the safety decisions has significant impacts on the review process efficiency.

Both traditional deterministic and probabilistic approaches to safety analyses are based on identification of hazards, initiating events that disturb normal operation, and scenarios (event sequences) that could evolve from the initiating events, as well as their associated consequences. Theoretical and experimental bases are needed to understand the associated phenomenology of possible scenarios. The design maturity and knowledge base of new non-LWRs are not likely to be as comprehensive as they were for evolutionary LWR-based designs; this limited knowledge base may impact the regulatory review. When there is a lack of operating experience or an inability to perform experiments with sufficient similitude to the planned full-scale design, one approach, as suggested by the ACRS, is limitations on power ascension and focused surveillance tests during initial operation.[7]

Proper consideration of uncertainties: Safety-licensing decisions are made in the face of uncertainty and within the boundaries of the state of knowledge of how the proposed reactor design would behave under both normal and accident conditions. Both deterministic and probabilistic safety evaluations must deal with uncertainties, proper consideration of which significantly helps the review process. Addressing uncertainties affects reviewer confidence regarding the results of safety evaluations and the resulting safety margins.

Two major groups of uncertainty that have been recognized are aleatory (or stochastic) and epistemic (or state-of-knowledge). The key distinction between these two types is that aleatory uncertainty is irreducible, where as epistemic uncertainty can be reduced by further study.

There are two classes of epistemic uncertainty: parameter uncertainty and model uncertainty. Parameter uncertainties are associated with the values of the fundamental parameters of a model, such as equipment failure rates that are used in quantifying the accident sequence frequencies in PRAs. Model uncertainties reflect the limited ability to accurately model specific events and phenomena. Completeness, including possible "unknown unknowns," can also be considered one aspect of model uncertainty. Completeness uncertainty arises because not all contributors to risk are addressed in PRA models, and not all phenomena and processes are addressed in deterministic safety evaluation models. The safety philosophy of defense-in-depth and safety margins has been the traditional means of dealing with uncertainties.

The novel aspects of new technologies and first-of-a-kind reactor concepts can make the identification of hazards, initiating events, and scenarios more challenging. To address uncertainties caused by limited information, the ACRS has recommended critical examination of the design, its safety behavior, and all aspects of operations, starting from a "blank sheet of paper" to avoid bias.[6] The committee has also suggested use of several analytical tools, which have been developed to improve the search process and apply equally to traditional and probabilistic safety analyses. Such tools can help formalize and add structure to the safety assessment and reduce completeness uncertainty.

Appropriate submittal timing: When it comes to submitting supporting documents (e.g., licensing topical reports), proper timing may have a significant impact on the efficiency of the review process. Submittal of critical licensing topical reports too late in the review process, or in tandem with related chapters of the design certification application, can reduce efficiency. The proper timing follows the sequential hierarchical order of submittals, wherein licensing topical reports on methodology description, demonstration, and verification and validation precede the applications.

Similarly, proper timing is also vital when submitting critical topical reports for the review of non-LWR concepts, which are likely to have more uncertainty associated with analytical methods and their application, underlying experimental bases, and validation of models. The licensing topical reports that support the design basis and safety analyses should be reviewed as early in the process as possible because new reactor designs, especially non-LWRs, will generally be more dependent on analytical methods for understanding the safety response of the system.

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