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NUCLEAR SAFETY

Sources of Nuclear Regulatory Requirements

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By C. Komanoff*

[Editor's Note: The following article on the sources of regulatory requirements is a fluent and well-referenced review of the subject. The author, who is known to the nuclear community primarily because of his criticism of it, has drawn on sources both within and without the nuclear community in support of his thesis that future expansion of the nuclear industry will lead to the imposition of new regulatory requirements. This interpretation, although it may appear valid on the basis of a linear extrapolation of past experience, may be challenged by many who can logically question this extrapolation. In particular, experience has demonstrated the low health risk of nuclear power—on both an absolute and a relative basis. Moreover, the present administration is aware of the problems of overregulation and appears to be disposed to proceed more prudently in this regard.]

Abstract: *This article reviews the evolution of regulatory requirements pertaining to the design and construction of commercial nuclear power plants in the United States. It identifies three main factors that have caused regulatory requirements to become more stringent: (1) the increasing reactor population has necessitated reducing the per-reactor accident risk to maintain a high probability that a serious accident will not occur; (2) licensing reviews and operating experience have demonstrated that the desired levels of safety for the nuclear sector as a whole were not being achieved; and (3) the increased regulatory effort required to license and oversee an expanding nuclear sector has caused regulatory standards to be made more uniform throughout the sector, generally at a higher common denominator. The close link between regulatory stringency and the size of the nuclear sector suggests that future expansion of the nuclear sector will lead to the imposition of new regulatory requirements.*

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What is the source of new nuclear safety requirements? This question is critical to the future of nuclear power in the United States. Reactor construction costs have risen rapidly, even after adjusting for the effects of inflation. Much of this real increase in costs resulted from the promulgation of more stringent regulatory standards. Yet future standards and their costs have proven difficult to predict. An analysis of the processes involved in developing safety requirements may provide insight into the future rate of their imposition and thus into future nuclear costs.

This review of the evolution of commercial nuclear power regulation and licensing in the United States indicates that increased regulatory stringency arises primarily from three phenomena: (1) the attempt to reduce the permissible risk to public health and safety per reactor; (2) new information indicating that current standards are insufficient to reduce risks to the desired levels; and (3) the greater documentation and standardization of regulatory requirements that accompany expansion of the regulatory effort. These phenomena, in turn, are fed primarily by six distinct motivating sources:

1. Increases in reactor population necessitating a reduction in the risk per plant in order to limit the overall accident probability.
2. Increases in reactor size, which in turn lead to increases in the potential consequences and probabilities of accidents.
3. The discovery of new safety issues through government and industry design and licensing reviews.
4. Reactor operating experience uncovering previously undetected safety problems and underscoring the severity of known, unresolved issues.
5. An increase in public concern contributing both to reducing permissible risks and to the unearthing of new safety problems.
6. Increases in the size and purview of the regulatory staff resulting from the increase in reactor population and the foregoing five factors.

This article begins with a brief history of nuclear power commercialization in the United States and then explores the six sources of increased regulatory stringency listed above. It concludes by considering the relationship between increased stringency and nuclear power expansion and by assessing the prospects for stabilizing regulatory requirements.

THE EXPANSION OF NUCLEAR POWER

The modern era of nuclear power operation in the United States began 13 yr ago with declarations on New Year's Day 1968 that "commercial operation" status had been achieved at America's first 500-MW reactors—the Haddam Neck (Connecticut) and San Onofre (California) plants. Previously, U. S. power reactors comprised only the dozen 10- to 75-MW reactors that started entering service in the late 1950s and three 200-MW plants that began operating in the early 1960s (Ref. 1). All were subsidized under the Power Reactor Demonstration Program managed by the Atomic Energy Commission (AEC) from 1955 to 1963 to promote commercialization of nuclear power.

These reactors were succeeded by a dozen so-called turnkey plants provided at a fixed price (and at a loss) by Westinghouse Electric Corporation and General Electric Company (GE). The order for the first of these (the 650-MW Oyster Creek plant) in 1963 was heralded as proof that nuclear power could compete economically with fossil-fuel generation of electricity without direct government support of plant construction or the fuel cycle. The nearly identical Nine Mile Point 1 was ordered several months later and built on a commercial (non-turnkey) basis. After a 2-yr pause, the other turnkey plants, also in the 500 to 800-MW range, were ordered in 1965 and early 1966. Most entered service between late 1969 and 1972.

Following (and occasionally overlapping with) the turnkey contracts came a second and much larger wave of reactor orders. Fifteen additional units were ordered in 1966 and twice that number in the following year. These reactors averaged several hundred megawatts more than their immediate predecessors. Some surpassed 1000 MW—up to five times the capacity of the largest operating reactors. This not only violated the power industry's precept against large jumps in unit sizes but also raised significant new safety issues, as discussed below.

Nevertheless, the nuclear rush was on. After over a decade of striving, with limited success, to advance nuclear power as a commercial power source, the AEC was suddenly being required in the mid-to-late sixties to license and to ensure the safety of dozens of large reactors, with many more expected to follow shortly.[2]

REDUCING THE ACCIDENT PROBABILITY PER PLANT

Following the Three Mile Island (TMI) accident in March 1979, nuclear pioneer Alvin Weinberg wrote: "For nuclear energy to grow in usefulness, the accident probability *per reactor* will simply have to diminish."[3] Otherwise, nuclear expansion could lead to such a high rate of accidents *per year* that the public's confidence in nuclear power would collapse and plants would be forced to close.

Weinberg's prescription appears to have been followed historically. As nuclear power has expanded, nuclear regulators have tried even harder to reduce the risks associated with the new reactors to prevent the overall nuclear accident probability from increasing as fast as the reactor population.

Although it is not codified in regulations, this effort informs much of the advocacy of improvements in nuclear safety within the regulatory community over the past decade and a half. It is particularly pronounced in the recommendations of the Advisory Committee on Reactor Safeguards (ACRS), an influential body of senior nuclear safety experts that advises the Nuclear Regulatory Commission (NRC), formerly the AEC, on safety matters and individual reactor licensing applications.[4] In November 1965, for example, the ACRS called on the AEC to upgrade standards for reactor pressure vessels on the following grounds:

> [T]he orderly growth of the industry, with concomitant increase in number, size, power level, and proximity of nuclear power reactors to large population centers will, in the future, make desirable, even prudent, incorporating stricter design standards in many reactors.[5]

The statement stimulated major efforts by the AEC and the nuclear industry to improve the design, fabrication, and "in-service inspection" of reactor vessels.[6] It also led to a 1967 AEC report on emergency core-cooling systems (ECCSs) that recommended improvements in the manufacture and inspection of nuclear piping, valves, and pumps because "the large number of plants now being constructed and planned for the future makes it prudent that even greater assurance be provided henceforth."[7] Also in that year, the ACRS urged that greater attention be paid to reactor safety problems because "large increases in the number of reactors lead to the desire to make still smaller the already small probability per reactor that an accident of any significance will occur."[8]

Similar sentiments were expressed by the ACRS during the 1970s to support more stringent standards. Two ACRS chairmen, one of whom subsequently chaired the NRC, told the Congressional Joint Committee on Atomic Energy in 1971 that "the high degree of conservatism used in both nuclear plant designs and in safety reviews" was justified by "the increased number of reactors soon to be operating and . . . the trend toward large reactors of higher power densities."[9] Two years later, the ACRS asked the AEC chairwoman to seek improvements in the ECCSs on the grounds that ". . . for an expanding nuclear industry, the cumulative effects of the added improvements represent prudent goals."[10]

The AEC regulatory staff also appears to have been guided by considerations of the accident frequency of the total nuclear sector. In 1973, when recommending backup shutdown systems to prevent events in which the control rods fail to *scram* (shut down) the reactor during sudden interruptions in its normal operation (events known as anticipated transients without scram, or ATWS), the staff wrote that "since larger safety margins are appropriate as increasing numbers of power reactors are built and operated, design improvements should be made to reduce the probability of ATWS in new plants to a negligible level . . . ."[11] The staff further wrote:

> The present likelihood of a severe ATWS event is considered by the staff to be acceptably small, *in view of the limited number of plants now in operation*, the reliability of current protection system designs, and the expected occurrence rate of anticipated transients of potential safety significance. As *more plants are built*, however, *the overall chance of ATWS will increase*, and the staff believes that design improvements are appropriate to maintain and to improve further the safety margins provided for the protection of the public.[12]

In other words, as more reactors come into operation, the per-reactor probability of an accident must be reduced in order to control the overall accident frequency. Similarly, in 1975 the staff mandated improvements in leakage control systems for main steam isolation valves because "there is a need for design improvements to provide appropriate safety margins for the large number of plants now planned."[13] Conversely but consistently, "the limited number of operating nuclear power reactors" in 1980 following the slowdown in reactor licensing and construction in the late 1970s was cited by the staff as a reason to grant utilities several years to phase in ATWS-mitigating design changes rather than requiring them immediately.[14]

INCREASES IN REACTOR SIZE

Many of the statements quoted above cite increases in reactor *power level* as well as in the *number* of reactors as a source of concern, and, indeed, the rapid increases in reactor generating capacities—from 200 MW for plants licensed in the mid-1950s to 600 MW in the mid-1960s and 1000 MW shortly thereafter—induced regulators to seek more stringent safety measures. Accidents at larger reactors could have more serious consequences, since they carry more fuel with a proportionately greater fission-product inventory which is subject to release. In addition, greater preventive measures were required to constrain the accident probabilities of large units. As AEC regulatory staff stated in 1967:

> The increase in this potential hazard [from larger reactors] must be matched by corresponding improvement in the safety precautions and requirements if the safety status is to keep pace with advancing technology. The protective systems must have shorter response times, larger capacities, and greater reliability to cope with the more rigorous demands presented by the large reactors.[15]

Although the need for shorter emergency response times has receded somewhat since thinner fuel rods with reduced fuel-centerline temperatures began to be introduced in the early to mid-1970s, larger plants do generate proportionately more decay heat following reactor shutdown. Removal of decay heat is a particular concern in many postulated accidents, including small-break loss-of-coolant accidents (LOCAs). As the ACRS noted in 1967, "the *decay heat* production from a large reactor such as . . . Browns Ferry [1098 MW] begins to approach a level compared to the original *full load* power level of the Shippingport reactor [70 MW]."[16]

Moreover, starting with the commercial-size plants first licensed in the early 1960s, new reactors required concrete shielding for the containment walls to reduce the exposure of nearby persons to radiation in the event of accidents. This further cut down the rate of heat dissipation through the containment. As a result, whereas reactor vessels were believed capable of containing a 100-MW molten core, a 1000-MW core "would eat its way right through the pressure vessel" and perhaps through the containment as well.[17] The result was that, as plant sizes grew, increased consideration was given to the ECCS, to systems for removing decay heat from the containment, and to systems for removing radioactivity from the containment atmosphere to reduce leakage.[18]

Concerns over the increases in reactor sizes and the increasing number of reactors combined to produce more stringent regulatory standards for ECCSs and primary reactor piping in the late 1960s. The ACRS was particularly troubled by uncertainties in ECCS performance for the first 800-MW reactors and in 1966 drafted a letter to the AEC chairman in which the following statement was made:

> [A]s more and more reactors come into existence, particularly reactors of larger size and higher power density, the consequences of failure of emergency core cooling systems take on increased importance.[19]

Although it was never formally sent, the letter apparently contributed to GE's decision to expand the ECCS at the Dresden reactors then under construction by adding a separate core-flooding system to the two ECCS core spray systems.[20] Perhaps more importantly for the long term, the letter led the AEC to establish the study group on ECCS referred to earlier. The emphasis in the group's report on the difficulty of delivering cooling water to a partially melted and reshaped core highlighted the importance of equipment to prevent or promptly halt any LOCA.

This and other recommendations in the report prompted the reactor vendors to increase in-service inspection and leak-detection tests for primary system piping and to further expand the diversity and reliability of the ECCS to ensure that it could respond to a wider range of system pressures corresponding to a greater number of potential accidents.[21] General Electric substantially increased the capacities of the ECCS core-flooding and core spray systems for Browns Ferry (1100-MW class) compared to those for Dresden or Quad Cities (800-MW class).[22] Finally, the report's implicit finding that current containment designs might not suffice to contain a melted core sparked concern over ECCS performance, both within and outside the AEC, and helped lay the basis for the tumultuous ECCS rule-making hearings in the early 1970s.

GROWTH OF THE NRC LICENSING EFFORT

The increase in the number of applications for reactor construction permits and operating licenses has also contributed to growth in regulatory requirements. The increase has necessitated a larger NRC staff, which in turn permitted a broader range of safety issues to be examined. It also has led to standardized review procedures that have tended to raise the stringency of standards applied to all plants. Many specific applications have raised new safety concerns, provoking the development of new criteria that have been applied to other reactors.

AEC staff reviews of the first commercial-size reactors were generally limited and haphazard. Each construction-permit application was reviewed by several engineers from the Hazards Evaluation Branch within AEC's licensing division. Experts outside the division had to be called on for technical support, and staff positions on specific design issues were frequently casually codified and documented. Although the first draft reactor *general design criteria* were issued for comment[23] in 1965, few standards had been developed to determine whether the proposed designs and equipment satisfied the criteria. According to licensing specialist B. N. Naft of NUS Corporation, "In the days of the earliest commercial plants, guidance from AEC was based on direct communications and what 'the last applicant had been through.' "[24]

By early 1967, with a dozen large reactors under construction and over a dozen more construction-permit applications docketed, the AEC began to significantly expand its licensing division to cope with the growing case load. The larger staff included reactor specialists who could question applicants more thoroughly on plant designs and construction methods. Word-of-mouth approvals of design approaches were superseded by detailed examinations requiring documentation of engineering assumptions, analyses, and tests.

The AEC also moved to formalize licensing reviews in order to equalize the scrutiny applied to different reactors, to expedite applicants' responses, and to establish uniform procedures to be followed by the growing staff. Licensing positions on specific safety issues were detailed in "Branch Technical Positions." In 1970 the staff inaugurated a series of Regulatory Guides, initially called Safety Guides, specifying acceptable approaches to problematic design and construction issues. Early guides often did not contain new approaches but, rather, codified previously developed positions to provide documentation needed by both staff and applicants. Subsequent guides were published so that guidelines that had been formulated and used in individual reactor reviews could be used for other applications. As the staff and the licensing effort grew, the practice of pegging all construction-permit reviews to the highest common denominator began. This practice elicited greater conservatism in areas such as design of engineered safeguards and quality-assurance (QA) programs.

The number of Regulatory Guides grew from 3 in 1970 to 21 in 1971 and 33 in 1972. Some were innocuous from a cost standpoint, but others—those pertaining to construction methods, seismic criteria, and engineered safeguards—engendered design changes and cost increases. The status of the guides also evolved from guidelines to requirements. Staff usually insisted on close adherence to the practices outlined in the guides, and applicants "volunteered" to conform rather than engage in time-consuming negotiations. As a consultant report to NRC noted, "Utilities often concluded that proposing alternatives to solutions and approaches identified in NRC guidance would be too costly. In these cases the NRC guidance serves as *de facto* regulation."[25]

The *Standard Review Plan,* a compilation of internal review procedures begun by the AEC in 1972 and initially issued by the NRC in 1975, has also tended to raise regulatory practice to the highest common denominator. The plan contains the criteria that new plants must satisfy and staff procedures for assessing whether the criteria are met. It was developed to provide a handbook of requirements for the growing staff, to serve as a benchmark for evaluating changes in regulatory practice, and to standardize licensing criteria. It has had to be updated continuously to keep up with proliferating new requirements. The *Standard Review Plan* now references approximately 150 Regulatory Guides (many in their second or third edition) and the number is still growing, fed by new staff reviews and expanding reactor operating experience.

Many Regulatory Guides have been prompted by particular license applications. As the AEC deputy director of regulation stated in 1967:

> [W]hen any safety problem is first encountered by our regulatory staff . . . we first encounter it on a case-by-case basis. In that process . . . we might come to understand the issues involved, the effects that might result with respect to reactor safety, and what the protective mechanisms might be.[26]

Seismic issues, for example, were first raised in the early 1960s in applications to build reactors near earthquake faults in California—at Bodega Bay on the northern coast and at Malibu near Los Angeles. Prior to these applications, the AEC had not considered seismic phenomena in licensing and had no familiarity with them. (Two small California reactors licensed in the 1950s, GE's Vallecitos test facility and the Humboldt Bay plant, received no detailed seismic review at that time and subsequently shut down in the late 1970s rather than upgrade seismic safeguards at high cost.) Shortly thereafter, the AEC commissioned seismological and geological research which demonstrated that the eastern United States also has considerable seismic potential. In addition, new research in soil mechanics and in structural seismic response led to improved understanding of the transmittal of seismic loadings to reactor equipment. This information led to the publication of eight Regulatory Guides delineating methods of calculating earthquake forces and specifying the instrumentation, structural reinforcement, and component reliability necessary to reduce susceptibility to damage and accidents, which apply, in varying degrees, to all U. S. reactors.

Similarly, concern over intermingling instrumentation for reactor control (operation) and safety (shutdown) first emerged as a significant issue in the ACRS review of the pressurized-water reactor (PWR) Diablo Canyon 1 in 1967. This Westinghouse-designed unit "was to be one of the first of the high-power PWRs built . . . which made it a logical reactor on which not only to look for new, previously unanticipated issues, but to resolve some that had been ongoing," according to one long-time ACRS member.[27] The questions raised in the Diablo Canyon review were also directed at the next Westinghouse reactor in line for a construction permit, Prairie Island, even though the much smaller Prairie Island design had already been accepted by the AEC staff.[28] These and all succeeding Westinghouse plants were ultimately required to increase the separation of control and safety circuits, although not to the extent desired by some ACRS members.[29]

Specific license applications have brought other safety issues to the fore. Hurricanes were first considered in the construction-permit review for Turkey Point in Florida in 1966 and subsequently were factored into the reviews of all East and Gulf Coast sites, and even many flood-prone inland sites.[30] Tornado-protection requirements were initially applied to new reactors in the late 1960s after the first review for a reactor in a high-tornado area—GE's Southwest Experimental Fast Oxide Reactor (SEFOR) in Arkansas—established that tornadoes occurred sufficiently frequently in most parts of the country to warrant uniform defenses in design.[31] Similarly, some sites with relatively high population densities appear to have acted as "magnets" for greater regulatory emphasis on engineered safeguards, which then spread to other reactors.[32]

Operating license reviews have also uncovered generic issues leading to changes in regulations. During late construction at Oyster Creek in 1967, cracks resulting from a combined design and welding deficiency were discovered in most of the control-rod housings. When the AEC staff examined the utility's QA program, it found widespread deficiencies in field construction, installation of instrumentation and power circuits, and equipment procurement, including installation of secondhand valves of unknown condition. These findings provided greater impetus for increased AEC inspections and audits and for promulgation of QA regulations.[33]

INDUSTRY REVIEWS

Reviews of new reactors by utilities, reactor vendors, and architect–engineers have also developed information that has contributed to the changing of standards. These reviews are usually considerably more detailed than those performed by the NRC, which primarily audits industry's analyses. Accordingly, industry analyses of new designs of new reactors using previously approved designs sometimes uncover safety problems that the NRC staff failed to unearth independently.

An example is the "pressure-suppression" issue for boiling-water-reactor (BWR) containment structures. The BWR containments have progressed through three stages: Mark I containments are used at most operating BWRs and at several nearing completion; most BWRs now in advanced construction use Mark II designs, whereas most in design or early construction will use Mark III. All three containments use a pool of water as a heat sink located in, below, or around the primary containment wall. They differ with regard to the materials used, the geometry of the pool, and the configuration for venting air or steam during accidents.

As part of its development of engineering data required by the NRC to approve the Mark III, GE constructed a test facility in 1975 to measure the pressures that would be exerted on structures within containment during postulated accidents. These tests showed that very large vibratory pressures could result from the rush of air and steam from the reactor into the surrounding pool during a LOCA. This led GE and the NRC to question the adequacy of Mark I and II designs along with that of the less-developed Mark III and to bolster all of the containments to reduce the chances of equipment failure during possible accidents. At plants under construction or in design, extra steel has been added to reinforced-concrete containment walls, all-steel containment walls have been thickened or further ribbed, and supports have been strengthened for equipment located within the pool area.[34]

An analogous example affecting PWRs concerns possible "asymmetric loading" on reactor vessel supports. In 1975, Westinghouse and the Virginia Electric & Power Company notified the NRC of findings from improved analytical models being applied to the North Anna plant then under construction: certain postulated LOCAs could create "pressure transients" in the reactor vessel that could overstress the vessel supports. The resulting displacement of the vessel could compress the fuel assemblies and prevent control-rod insertion, disable the ECCS, and damage supports for the reactor coolant pumps and steam generators.[35] This previously unidentified scenario was subsequently established as an *unresolved safety issue*—a pending generic problem whose resolution may require formulation of new regulatory requirements.

These examples illustrate NRC's involvement in industry's contribution to the upgrading of standards. Also relevant are the standards and codes developed by technical societies, such as the American Society of Mechanical Engineers (ASME) or the Institute of Electrical and Electronics Engineers (IEEE), whose members work in various facets of reactor technology and safety. Over a hundred of these documents issued since the late 1960s have led to more stringent manufacturing, testing, and performance criteria for structural materials such as concrete and steel and for basic components such as valves, pumps, and cables. Most of the codes have been "endorsed" by NRC Regulatory Guides; indeed, in some instances the industry-dominated committees drafting new, stringent standards have "justified their stance by stating that unless industry addressed these concerns the NRC would in regulatory guides."[36]

REACTOR OPERATING EXPERIENCE

Reactor technology was initially developed with the expectation that the design, construction, and operation of reactors could be rigorously controlled and managed. Nuclear pioneers concentrated their analytical efforts on physics matters such as reactivity accidents and devoted less attention to the difficult engineering problems of integrating the nuclear steam system with the balance of the power plant, keeping coolant water circulating in the core, and providing safeguards to prevent or mitigate accidents.

These problems began receiving attention in the reviews of the first 500-MW reactors in the early 1960s. Criteria considered excessively conservative were specified in some instances because of the paucity of engineering experience. It was anticipated that favorable operating data would ultimately allow some standards to be relaxed. In actuality, reactor components and equipment have frequently failed to achieve intended levels of reliability and performance. Although operating experience has sometimes justified reduced design margins (in fuel performance, for example), it has more often warranted corrective standards and engendered a more conservative overall regulatory approach.

Nuclear operating experience has come in two waves. The first consisted of the three 200-MW reactors and the dozen units under 100 MW licensed in the late 1950s and the 1960s. Although these reactors had accrued only 60 unit-years of operation by early 1967, this experience provided sufficient evidence for the ACRS to conclude that, "based on reactor operating experience . . . a variety of reactor transients have occurred, a variety of protective features have malfunctioned or been unavailable on occasion, and a variety of defects have been found in operation."[37]

The ACRS cited these specific failures: (1) loss of normal and emergency power in the same incident; (2) simultaneous loss of all (as many as five) incoming power lines; (3) blowdown of a primary coolant system; (4) loss of all protection provided by the capability for automatic scramming of control rods; (5) sticking and breaking of control rods; (6) rupture of a poison sparger ring; (7) failure of structural members within the pressure vessel; (8) faulty design of a steam-generator support; (9) cracks in large pipes and studs; (10) poor choice of material for vital components; (11) melting of some fuel elements; (12) consecutive procedural errors; and (13) safety systems not wired in accordance with design criteria even after extensive test programs. Although it is not possible here to trace specific upgrading of standards to these failures, the "lack of perfection in design, construction, and operation"[38] was a major reason for the ACRS's advocacy in the mid-to-late 1960s of conservative design practices and improved safeguards. For example, the ACRS's portentous 1965 letter to the AEC on pressure vessel integrity was prompted by one member's concern over failures such as broken stud bolts at the vessel head closure and cracked main control-rod shafts. These were "incipient failures which, had complete failure occurred, would have resulted in more serious accidents than any thus far experienced."[39]

Commercial-size (400+ MW) reactors have registered a far larger body of operating experience, beginning with the 1968 startup of Connecticut Yankee and San Onofre. Experience with these larger reactors accumulated slowly at first, reaching only 11 unit-years at the end of 1970. But it appreciated rapidly as more reactors were completed, reaching 36 unit-years at the end of 1972 and 94 after 1974. This operating experience included incidents such as fuel leakage, pipe cracks, faulty installation of control rods, disabling of shutdown systems by the operator, and malfunctioning valves, pumps, and cables. The NRC staff later characterized this as a "considerable body of operating reactor experience [which by 1972] indicated the need for expanded technical review in areas previously thought to be not sufficiently important to warrant much attention."[40]

Utilities report operating problems and deviations in Licensee Event Reports (originally called Abnormal Occurrence Reports) to NRC's Office of Inspection and Enforcement (originally the AEC's Division of Operating Reactors). Not all reports have stimulated corrective action, however. For example, the stuck pressurizer relief valve that caused a substantial loss of primary coolant at Davis–Besse in 1977 was not corrected at other Babcock & Wilcox reactors and contributed to the TMI accident. The NRC's Special Inquiry Group on TMI inferred from this and other disregarded events (which were precursors to the TMI accident) that "the NRC and the industry have done almost nothing to evaluate systematically the operation of existing reactors, pinpoint potential safety problems, and eliminate them by requiring changes in design, operator procedures, or control logic."[41]

Notwithstanding the lack at that time of *systematic* evaluation procedures, many adverse operating events have been "incorporated into the safety reviews of new plants,"[42] as NRC Chairman Joseph Hendrie noted prior to TMI. This process antedates the present-day commercial reactors. For example, a tornado that knocked out all off-site power lines to the Dresden 1 reactor in the early 1960s led to the use of small diesel generators to provide on-site emergency power—a requirement that evolved into much larger diesels to drive safety systems such as the ECCS.[43] More recently, operating experience has arguably become the largest single source of new regulatory requirements. For example, "fire had been recognized as a potential safety concern of considerable importance for at least a decade before occurrence of the Browns Ferry fire in

Table 1 Regulatory Guides Citing Operating Experience (Partial Listing)

Number	Date*	Title
1.6	3/71	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems
1.31	8/72	Control of Ferrite Content in Stainless Steel Weld Metal
1.43	5/73	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components
1.44	5/73	Control of the Use of Sensitized Stainless Steel
1.47	5/73	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems
1.55	6/73	Concrete Placement in Category I Structures
1.67	10/73	Installation of Overpressure Protection Devices
1.68.2	1/77	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants
1.96	5/75	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants
1.115	3/76	Protection Against Low-Trajectory Turbine Missiles
1.120	6/76	Fire Protection Guidelines for Nuclear Power Plants

*Most Regulatory Guides are effectively incorporated into the regulatory review process before their official publication.

1975,"[44] but requirements for cable loading and fire retardancy were upgraded only after serious fires at San Onofre (1967) and Indian Point (1971), and the Browns Ferry fire was the catalyst for major improvements in cable separation and ventilation systems at new plants.

Operating experience has also been cited as the impetus, at least in part, for many Regulatory Guides (see Table 1). Many other guides do not specifically mention operating experience but have also originated from adverse occurrences. They include guides relating to reactor coolant pump flywheel integrity (1.14), protection against pipe whip inside containment (1.46), loose-part detection systems for the reactor primary system (1.133), and many guides concerning quality assurance in component fabrication and plant construction.

Commercial reactor operating experience continues to accumulate rapidly. The total more than tripled from 140 unit-years at the end of 1975 to 430 at the end of 1980, providing much new fodder for more stringent standards (see Table 2). Some of these problem areas have been incorporated into NRC's roster of unresolved safety issues now receiving increased regulatory attention as areas of generic safety concern (e.g., system interaction, water hammer, and residual heat removal). Others have been the subject of NRC

Table 2 Adverse Events Compiled from Reactor Operating Experience, 1976–1979*

Serious Events at Individual Reactors

Faulty test procedures, eliminating the capability to detect a loss-of-coolant accident (Zion, 1977).

Deep circumferential crack in primary system piping (Duane Arnold, 1978).

Loss-of-coolant accident (Three Mile Island, 1979).

Classes of Events with Multiple Occurrences

Separation of control rods from drive mechanisms at BWRs.

D-C electrical failures, degrading the capability of residual heat-removal systems yet requiring their operation.

Water hammer and flow-induced vibration, causing equipment damage—in some cases to engineered safety features.

Systems interaction events, compromising the independence of presumed redundant safety systems (e.g., Zion, 1977).

Loss of high-pressure coolant injection capability due to valve leakage, improper valve lineup, or electrical failure.

Leakage between interconnected fluid systems, causing loss of residual heat-removal systems.

Failure to maintain containment isolation.

Cracks in welds connecting feedwater piping to steam generators at PWRs.

Continued degradation of steam-generator tubes and cracking in steam-generator supports at PWRs.

Overpressurization of pressure vessels at PWRs.

*Compiled from the ACRS report *Review of Licensee Event Reports 1976–1978*, NUREG-0572, issued in 1979.

bulletins and circulars requiring analysis or remedial action by licensees (e.g., loss of high-pressure coolant injection and feedwater weld cracks). The TMI accident, of course, has "introduced a large number of new or previously nonemphasized generic safety issues"[45] while provoking a sweeping reappraisal of safety regulation transcending the specific design and equipment inadequacies that contributed to the accident.

ROLE OF THE PUBLIC IN FORCING MOST STRINGENT SAFETY STANDARDS

Public concern over the hazards of nuclear power has grown as the nuclear sector has expanded. Increased operating experience has brought to light additional evidence of actual hazards, while the construction of more nuclear plants at more sites has increased the number of people exposed to the risks associated with nuclear power. In turn, rising public apprehension has affected nuclear regulation.

Although citizen interventions have been blamed for causing delays leading to higher costs, that effect was statistically negligible in terms of *real* (inflation-adjusted) dollars for reactors completed in the 1970s (Ref. 46). Most delays caused by citizen challenges have affected reactor licensing rather than construction, and plants that took longer to license did not have inordinately higher capital costs than plants on which construction started at the same time. Far more importantly, public concern has spawned expert critics who have identified deficiencies in reactor design, construction, and regulation. Public involvement in nuclear regulation has also reinforced conservative tendencies in the regulators.

The foremost technically skilled critic of reactor regulations has been the Union of Concerned Scientists (UCS). This organization was founded in 1969 to examine science and technology policy but soon turned primarily to reactor safety in response to prompting by its members and funders. UCS was in the forefront of intervenors at licensing hearings who attacked the AEC's 1971 ECCS interim criteria. These challenges helped to force the lengthy rule making that led the AEC in late 1973 to reduce permissible fuel temperatures and to prod the industry to improve ECCS reliability. At least as importantly, the hearings revealed the presence of dissent within the AEC and thereby conferred both publicity and legitimacy on nuclear critics.

UCS's continued critiques of nuclear regulations have both affected specific issue areas and colored the overall tone of the reactor safety debate. Criticism by UCS was cited for its "important contribution" to a 1980 NRC order upgrading standards governing fire protection for electrical cables and environmental qualification of electrical components.[47] Their critiques of the *Reactor Safety Study* (WASH-1400)[48] also helped induce Congress to direct the NRC to convene a committee to review the study. That review led the Commission to retract some of its prior support for the study, an action with important consequences for safety regulation, as discussed below.

Intervenors have sometimes brought about design changes in individual licensing hearings. At North Anna, for example, when a local environmental group showed that building supports were settling into the ground, the NRC ordered tests and design changes (primarily flexible expansion coupling for piping) that added to costs and delayed plant completion.[49] More generally, intervenor participation in licensing hearings has tended to make the regulatory staff "considerably more cautious and conservative,"[50] according to one observer, by fostering a climate conducive to detailed design review. The prospect of cross-examination by intervenors has encouraged applicant and staff witnesses to conduct thorough safety analyses. Similarly, the presence of intervenors tends to reinforce staff concerns with safety and help counteract pressure from the applicant for a speedy review.

Public concern about reactor safety has also reinforced the tendency of regulators to heed the potential for nuclear hazards and thus to add safety requirements designed to limit the overall accident probability. For example:

> The ACRS believes that it is proper that nuclear power be safer than other comparable technologies. The Committee has sought this goal. It believes that the country wants a higher level of safety for nuclear reactors and is willing to pay for it. The ACRS also believes that the country wants a higher degree of assurance as to the level of safety which is being attained.[51]

Statements such as these are a powerful counterweight to the view that different energy sources should have comparable risks and that nuclear power, with a lower calculated public health impact than some alternatives, should therefore not be subject to further major regulatory requirements. The statement arguably would not have been made in the absence of considerable public apprehension over reactor hazards.

Public concern has also affected nuclear regulation through Congress. Although few members of Congress are strongly antinuclear, the concerns of their constituents have led to closer congressional oversight and thence to stricter regulation. In 1977, for example, the chairman of the House Subcommittee on Energy and Environment, Rep. Morris K. Udall, succeeded in attaching a rider to an NRC appropriations bill creating a panel of reactor safety experts to review the *Reactor Safety Study.*[48] The experts' critical review[52] led to an NRC policy statement in early 1979 withdrawing support for the study's executive summary and restricting staff's use of the study's accident probabilities.[53] The first move has bolstered arguments for stronger regulatory standards, while the second may lead to more conservative design bases in specific licensing issues.

Congress has also required the NRC to publish digests of reactor "abnormal occurrences," lists of unresolved safety issues, and Task Action Plans to address key safety concerns. These requirements have raised NRC's priorities for resolving safety problems and thereby enhanced stricter regulation. Moreover, publication of the information has deepened the sense among a large segment of the public that reactors present many potential safety hazards warranting greater attention. In turn, this heightened concern has affected the regulatory process through the conduits previously described.

The entire nuclear enterprise, in fact, has been conducted increasingly in a "fishbowl" environment that admits scrutiny of every aspect of nuclear regulation and operation. Operating anomalies are fed back into design and operating reviews; designs are examined in NRC staff reviews and licensing hearings; construction is scrutinized by activists, the press, the NRC and, increasingly, by the workers at nuclear facilities.

The latter development was first evidenced in 1971 when a welding supervisor at the Surry plant reported that primary coolant piping contained numerous defective welds. The following year, an anonymous letter to the ACRS—perhaps sent by a reactor design engineer—disclosed that postulated steam-line breaks at the Prairie Island reactors could cause pressure to rise in the auxiliary building to the point that vital electrical and mechanical equipment might fail and impede plant shutdown. Neither the applicant nor the AEC staff had evaluated this issue in their reviews.[54] The AEC responded by requiring many plants under construction and in design to conduct further accident analyses, reroute pipes, and modify their auxiliary buildings to provide pressure relief in the event of steam-line failure.[55]

More recently, "heightened public awareness and interest in nuclear power [have] resulted in an increase in the number of allegations received by NRC"[56] of irregularities in plant construction. Since 1977, construction personnel at the Callaway, Wolf Creek, South Texas, and Marble Hill reactors have charged that QA requirements were being bypassed and that designs were being amended in the field by unqualified personnel. These allegations and a critique of NRC construction inspection procedures by the General Accounting Office[57] have led the agency to toughen its supervision of construction. At Marble Hill, for example, workers' affidavits led to the discovery of 170 inadequate patching jobs in concrete walls—including voids up to 180 ft^3 in size[58]—and to the suspension by the NRC of safety-related construction lasting more than a year and a half. The NRC has subsequently announced its intention to consider new rules to enhance the independence of QA auditors and to expand its own inspections of reactor construction activities.[59]

OUTLOOK

The preceding discussion indicates that considerable impetus for new reactor safety requirements in the United States has come from expansion of the *nuclear sector,* i.e., from increases in the total capacity of reactors operating or under construction. Growth in the population of reactors has required new licensing reviews in which additional safety problems were first discovered or addressed. It has led to a more rigid administration of licensing standards and procedures which has raised the stringency and specificity of safety requirements applied in staff reviews. It has also induced regulators generally to endeavor to reduce per-plant risks to contain the industry-wide probability of a serious accident.

Similarly, increases in reactor generating capacities—which, together with reactor population, determine sector size—have necessitated greater safeguards in order to maintain desired safety margins. With the increase in operating experience, new safety deficiencies requiring remedial regulatory actions have been unearthed.

Nuclear sector expansion has also broadened and intensified public concern. With the increase in the number of plants, there have been more frequent mishaps which, with more persons living near more reactors, are accorded wider publicity that in turn adds to pressure on the regulators to abate the perceived hazards. The "fishbowl" environment in which nuclear power must function was not present when only a dozen reactors were operating or planned; it materialized as the concomitant of a large-scale nuclear program that was undertaken before sufficient technical and managerial maturity was achieved.

This is not to say that all of the ingredients that have contributed to increased regulatory stringency are subsumed under nuclear sector expansion. Some information leading to more stringent requirements has come from accident-related research by the national laboratories; for example, work at the National Reactor Testing Station* in Idaho in the early 1970s indicated that ECCS cooling water might not reach the reactor core in some circumstances. (Note, however, that much accident research has been ordered by AEC/NRC in response to new pressures or information originating from the various sources described earlier.)

Separately, although it is likely that increases in the prices of competing energy forms such as oil help create a context in which cost-engendering new requirements are more palatable to nuclear regulators, this factor is not tied to expansion of the nuclear sector, nor are the regulators' own opinions as to the importance of nuclear power, or those of the President, who appoints the NRC commissioners and can seek to influence their conduct of regulatory policy. Conversely, those responsible for nuclear regulation have generally understood that a serious reactor accident—an event whose probability of occurrence must, in the absence of safety improvements, be proportional to nuclear sector size—could spell the end of nuclear power in the United States. This consideration has been reinforced by the TMI accident.

Accordingly, the linkage of increased regulatory stringency to nuclear sector expansion seems firmly based in both regulatory history and logic. Most of the drive toward greater regulatory requirements appears to be accounted for by (1) the need to improve the safety of new plants to keep the sector-wide accident frequency at a low level, (2) information concerning safety problems which arises from licensing reviews and plant operation, and (3) the more rigid administration of regulation necessary to license and monitor a large nuclear sector that draws public concern and scrutiny.

This linkage implies that the nuclear plants under construction (~90 GW capacity) face a significant further increase in regulatory standards. These plants long ago provided the impetus for new regulations when they were awarded construction permits in the 1970s, signaling to the regulators the need for new remedial measures to prevent increases in the sector-wide probability of a serious accident. In theory, the impetus could be defused if public attitudes toward nuclear risks change substantially or if it proves possible to dismiss outstanding regulatory issues without affecting accident risks. Neither event seems plausible, however. The growing number of genuine safety issues[60] and the continued widespread mistrust of nuclear power strengthen the presumption that nuclear regulation will grow more stringent.

The one eventuality that might be expected to slow the rate of increase in nuclear regulations is cancellation of a large number of reactors for which construction permits have been granted. Large-scale cancellations of plants being built would ease public concern and also enable regulators to restrain the growth in safety requirements somewhat without forcing up the sector-wide accident probability. Adjusting the growth of safety requirements to fit the reduced future capacity would be constrained, however, by continued detection of safety problems through operating experience at existing plants. Indeed, judging from the rate of issuance of NRC bulletins and circulars on generic problems, the detection rate *per reactor* apparently reached an all-time high during 1979–1980 (see Fig. 1). Major problems detected in 1979–1980 include the many systematic deficiencies in design and operation revealed at TMI, weaknesses in BWR scram systems, inadequate separation of nonsafety-grade from safety-grade instrumentation and control systems at Babcock & Wilcox reactors, substandard seismic design and construction procedures, and faulty PWR containment water-level controls and indicators, among many others.

At some point, the per-reactor rate of detection of safety problems will almost certainly decline. But even then, the *per-year* rate would fall less rapidly—and might even continue to increase for some time—because of growth in the number of operating plants. New safety issues will thus continually emerge while old ones will be reemphasized, inhibiting efforts to stabilize reactor design criteria and to standardize plants. Moreover, apart from prospective new standards, plants under construction are subject to many existing requirements from which recently completed plants were exempt due to "regulatory lag."

Accordingly, the "environment of constant change"[61] that so pervasively complicates nuclear design and construction should not be expected to

*Renamed Idaho National Engineering Laboratory.

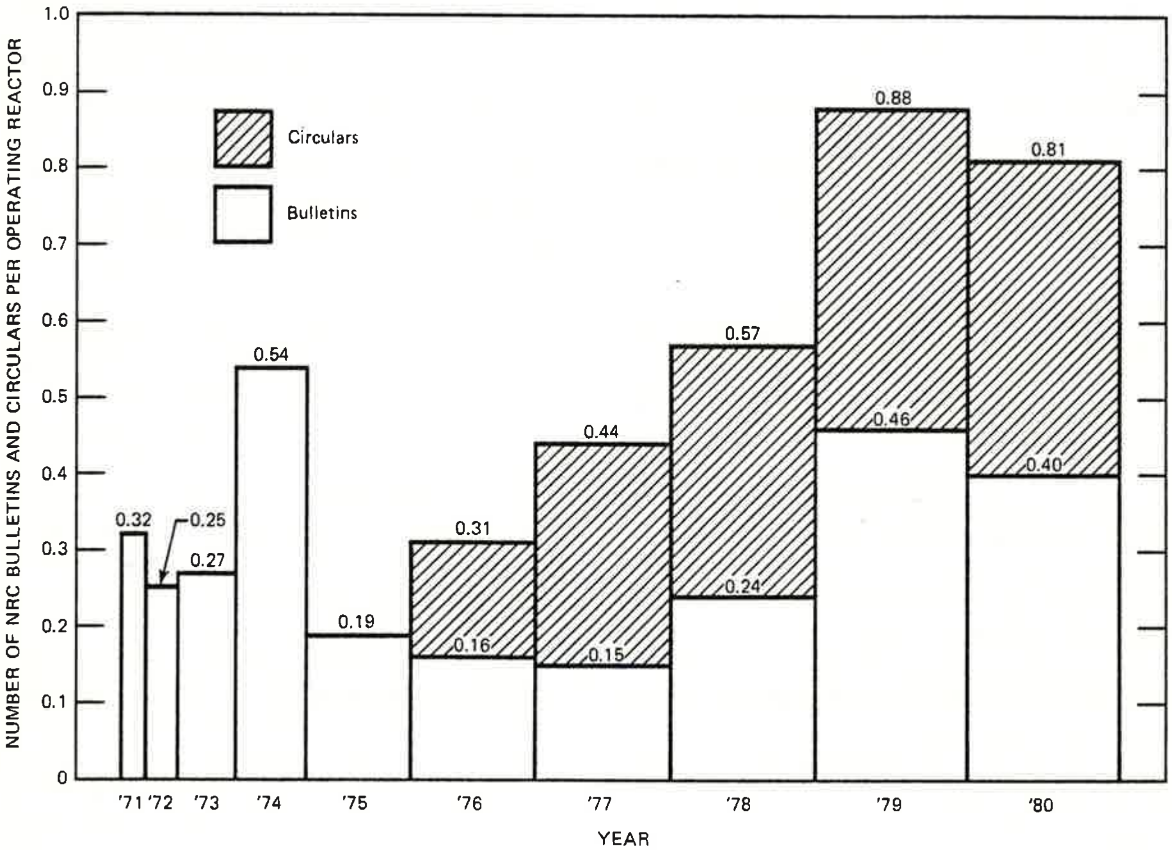

Fig. 1 Number of NRC bulletins and circulars used per operating reactor for 1971–1980. Bar widths represent numbers of licensed operating reactors (including partial years from commercial start, excluding units under 400 MW). Supplements to bulletins and circulars are not included. Circulars were first issued in 1976. Uppermost figures denote the total number of bulletins and circulars issued per operating reactor in that year.

improve significantly, short of a marked reduction in the currently projected growth of nuclear power. Such a slowdown would ease, but by no means completely dispel, the pressures that lead to new regulatory requirements.

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