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NUCLEAR REGULATION IN THE UNITED STATES AND A POSSIBLE FRAMEWORK FOR AN INTERNATIONAL REGULATORY APPROACH

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Abstract
Effective regulation of nuclear power plants is a key element of reactor safety. It also enhances public trust in the licensing and operation of these facilities. Although significant efforts to improve and enhance nuclear regulation have been made, three major power reactor accidents and a series of incidents have occurred and suggest that changes in the regulatory process are warranted. Various regulatory options are presented and possible directions for the international regulation of nuclear power plants are outlined.

Keywords
Nuclear Regulation, Proposed International Regulations, Fukushima Daiichi Accident Regulatory Impact, US Regulatory Approach

1.0 Overview

An ideal regulatory framework would be proactive, internationally accepted, supported by the public, anticipate accident events, constantly challenge accepted practices, and prevent major accidents that result in the release of fission products. The 1979 Three Mile Island Unit-2 (TMI-2) [1-3], 1986 Chernobyl Unit-4 [3-6], and 2011 Fukushima Daiichi Units 1, 2, 3, and 4 [7-12] accidents suggest that the conventional regulatory framework has not been completely successful in achieving these goals. Three major reactor accidents in a span of 35 years offer a sobering reminder that the current regulatory approach has not produced the desired results, and that change is warranted.

Wang and Chen [13] reviewed cultural and societal aspects of various nuclear regulation models following the Fukushima Daiichi Nuclear Power Station (FDNPS) accident. Their paper focused on regulatory performance with an emphasis on Chinese, Japanese, and US regulations and examined the impacts of the accident on European regulations. Wang and Chen reviewed the relationship between government and the nuclear industry, but did not examine the failure of regulatory agencies to anticipate or preclude specific events. They also concluded that the US approach is currently the best available regulatory model, but it also requires improvement. In particular, Wang and Chen [13] suggest that the relationship between the US nuclear industry and government involves potential conflicts of interest and the revolving door between industry and government should be eliminated. They also suggest that reactor safety be entrusted to an independent agency that has sufficient authority to be an effective regulator and that nuclear safety be handled by international rules without border limitations.

In this paper, a complementary approach is presented. Wang and Chen's conclusion regarding the US regulatory system forms the basis for the present paper. This paper focuses on the Nuclear Regulatory Commission (NRC) that licenses and monitors commercial power reactors in the US. Specific operational events and occurrences are reviewed to determine systemic weaknesses in the US regulatory system. These weaknesses suggest possible regulatory improvements. This review also offers a possible set of elements for international nuclear regulation.

The discussion begins with a brief review of US nuclear operating history and significant events that shaped the direction of nuclear regulation. The current US regulatory approach, its response to the Fukushima Daiichi accident, and proposed enhancements are also presented.

2.0 Brief Summary of US Reactor Regulatory History

A number of regulatory events illustrate the challenges faced by the NRC and its predecessor organization, the Atomic Energy Commission (AEC) [14]. The AEC had dual responsibility for developing and regulating the nuclear power industry. These responsibilities were often in conflict, and these conflicts were a key factor in stimulating the opposition to nuclear power and in public concerns regarding its development.

Unless otherwise noted, the US nuclear regulatory history is derived from the NRC's history summary derived from NUREG/BR-0175 [14]. US nuclear regulations were initially governed by the AEC and more recently by the NRC. These organizations dominated the commercial nuclear power regulatory environment in the US.

An examination of significant events provides an historical perspective regarding the evolution of US nuclear regulation. It also illustrates the strengths and weaknesses in the process and suggests areas where improvements should occur. Following examination of these events, specific recommendations for future regulatory options are provided. These options encompass near-term revisions to US regulations and eventual transition to an international regulatory process that will enhance reactor safety.

Issues of reactor safety regulation have been historically linked to the protection of the three fission product barriers. These barriers are the fuel and associated cladding, the reactor vessel and included piping, and the containment building. Fission product barriers prevent the release of radioactive material to the environment. Preserving these barriers is an important requirement for a successful regulatory system.

2.1 Containment Fission Product Barrier

Prior to the mid 1960s, the AEC considered that the containment building was the final line of defense against the release of fission products [14]. However, it became apparent that under some circumstances the containment building could be breached. Given this possibility, protecting the public from a fission product release relied heavily on a reliable, properly designed, and functional emergency core cooling system (ECCS). The ECCS included passive, low-pressure, and high-pressure injection systems to ensure core cooling which would minimize the release of fission products to the environment. These ECCS systems were supported by spray systems, filters, ventilation systems, and air circulation units.

The containment issue prompted the AEC to conduct tests of the ECCS system. These tests were not conclusive and their results led to additional questions regarding the effectiveness of the ECCS. The test results cast doubt on previous AEC assertions regarding the adequacy of reactor safety systems, and had the potential to undermine public confidence in nuclear power technology. Unfortunately, the AEC attempted to prevent the test results from becoming public and withheld the results from Congress [14].

2.2 Emergency Core Cooling System Contention

As a result of the containment effectiveness issue, the Union of Concerned Scientists (UCS) and some scientists at US national laboratories questioned the reliability of the ECCS [14]. Questions regarding ECCS reliability arose following tests of subscale systems that failed to fully support the design predictions. The regulatory staff concluded that the issues identified in ECCS testing would be corrected in subsequent designs, and used engineering judgment to justify system requirements. However, the resolution of this safety issue was not universally accepted.

The ECCS controversy damaged the credibility of the AEC and strengthened its critics. Rather than acknowledging the ECCS issue and fully evaluating the system's uncertainties, the AEC acted in a manner that it hoped would not undermine public confidence in reactor safety. The AEC's actions added credibility to the reliability allegations and created an atmosphere of distrust that would influence future reactor safety issues. Had the AEC acknowledged the potential significance of the ECCS tests, devoted additional time to evaluate the associated uncertainties, and involved stakeholders such as the UCS in resolution of the issue, the dispute might have been resolved in a manner that would have avoided much of the distrust that arose and continues today.

The ECCS reliability issue continued to generate controversy into the 1970s, and extensive ECCS hearings were conducted in 1972. Howe testimony and media reports that reflected negatively on the AEC's safety program, revealed divisions among AEC experts, and further damaged the agency's credibility.

2.3 Lyons High-Level Waste Disposal Facility

Another issue that undermined public confidence in the AEC and strengthened its opponents was the methodology used by the Commission to select a high-level radioactive waste disposal site [14]. In 1970, following Congressional and scientific pressure regarding the need for a high-level waste disposal facility, the AEC decided to develop a permanent repository in an abandoned salt mine near Lyons, Kansas. This decision was made without conducting comprehensive geologic and hydraulic evaluations. These omissions were revealed when the State of Kansas and scientists challenged the appropriateness of the Lyons site. The resulting dispute intensified when Congress and State officials continued their challenges. Many of the issues raised during the Lyons site debate were again raised 40 years later with attempts to license the Yucca Mountain site [14].

This dispute ended in 1972, when the concerns expressed by the site opponents proved to be valid. The high-level waste and ECCS issues strengthened the opponents of the AEC and added credibility to groups opposed to the development of nuclear power. These issues as well as concerns over reactor design, reactor safety, quality assurance, and the probability of a major reactor accident continued and amplified the debate over nuclear power and further weakened the credibility of the AEC as an effective regulator of the US nuclear industry. The continuing controversies required the AEC to dedicate resources that would have been more productively utilized in resolving these issues and forging a more cooperative relationship directed at improving the safety and reliability of nuclear safety systems.

2.4 Transition from AEC to NRC

As the AEC's credibility was damaged, increased criticism was directed at its dual responsibility for developing and regulating the nuclear power industry. The weakening of the credibility of the AEC, led Congress in 1974 to divide it into two separate organizations with the NRC assigned responsibility for nuclear regulation [14]. Its other functions were transferred to a separate agency that is now the Department of Energy. The NRC inherited the AEC's credibility issues, antinuclear stakeholders, and growing public uncertainty regarding the safety of nuclear power.

Two new events added to the NRC's challenges in its first few months of existence. These were the Browns Ferry Nuclear Plant Fire in March 1975 and publication of WASH-1400 [15], the NRC Reactor Safety Study also known as the Rasmussen report.

2.5 Browns Ferry Fire

One of the first major issues for the NRC following its creation involved a reactor safety issue associated with a major fire at the Tennessee Valley Authority's Browns Ferry Nuclear Plant in Alabama [14]. This fire burned for about seven hours and nearly disabled the safety systems of one of the site's units. The fire was caused by a technician who used a lighted candle to search for air leaks in an area containing electrical cables powering portions of the plant's control room and safety systems. This event further damaged public opinion regarding the ability of the NRC to ensure reactor safety. The event also raised concerns regarding fire protection programs and their adequacy to protect safety systems. In addition, the Browns Ferry event directed attention to common-mode-failures in which a single failure could trigger a sequence of events that damaged defense-in-depth or redundant safety systems.

2.6 Reactor Safety Study

The NRC's WASH-1400 Reactor Safety Study [15] was commissioned by the AEC in 1972. The Massachusetts Institute of Technology developed the study's methodology with the assistance of NRC regulatory staff. This study's purpose was to estimate severe accident probabilities, since these estimates had not previously been determined in a rigorous manner.

WASH-1400 used methodologies including fault-tree analysis to conclude that the severe accident risks from a nuclear power accident were very small when compared to risks from other events such as aircraft crashes, dam failures, earthquakes, explosions, fires, hurricanes, tornadoes, and toxic chemical spills. Although WASH-1400 was a cutting-edge effort that addressed the complex event sequence of a severe accident, both nuclear proponents as well as opponents criticized this study [14]. One common objection was that the study failed to include additional sequences that could lead to a severe accident. Opponents asserted that the analysis and supporting data did not support the report's conclusions regarding the relative risks associated with a severe nuclear power accident. Given the controversy, the NRC withdrew its full endorsement of the report's executive summary in 1979.

Some of the critics of WASH-1400 were concerned about low probability, high consequence events. One of these events (Event V) involved a small break loss-of-coolant accident that occurred at TMI-2. However, Event V was not addressed as a likely operational event in WASH-1400 [15].

2.7 TMI-2 Accident

In 1979, the TMI-2 pressurized water reactor had a small-break loss of coolant accident (LOCA) with associated fuel damage [1-3]. TMI-2 was caused by a combination of operator errors and design weaknesses.

The 1979 TMI-2 accident [1-3, 14] galvanized nuclear power opponents and intensified public debate and criticism of the NRC. Following mechanical failures and human errors, the TMI-2 reactor core was uncovered and about one-half of its fuel melted. In spite of NRC oversight, a primary system's pressure relief valve failed to close following a reactor trip and created a flow path for the loss of primary coolant. Although, the ECCS system functioned as designed, operators significantly reduced its flow rate because available instrumentation suggested that the primary system was filling with water. Operators believed this condition presented a danger of over pressuring the reactor coolant system. When the actual condition of low core water level was discovered and ECCS flow restored, core water inventory was reestablished. However, core uncovery and fuel melting had already occurred.

The TMI-2 accident raised the question of the adequacy of NRC oversight of maintenance activities. In addition, the design of the control room did not provide a clear indication of the event (e.g., easily observable indication of the pressure relief valve failure to close, high water level in the reactor building basement, and actual pressurizer water level) which contributed to the plant operator's failure to determine that a loss of coolant accident was in progress. Moreover, a similar event previously occurred at the Davis-Besse Power Plant in Ohio, but the NRC failed to comprehensively evaluate or disseminate this information to other nuclear plants [1-2, 14]. Given these regulatory failures, the TMI-2 accident clearly demonstrated that a serious reactor accident with fuel melting could occur.

The TMI-2 corrective actions initiated by the NRC included greater emphasis on human factors in plant performance for minimizing the type of operator errors that contributed to the accident. These actions strengthened requirements for operator training, testing, and licensing. The NRC also promoted the use of control room simulators and performed assessments of control rooms and their instrumentation. In addition, the NRC's resident inspector program was expanded to include at least two representatives at each plant site. Focus was placed on the review and dissemination of operating data from nuclear power plants. Emergency preparedness programs were also expanded and more rigorously evaluated with drills and exercises. These actions strengthened the regulatory program and have prevented another major US accident.

In spite of these positive aspects, the TMI-2 accident represented regulatory failures on several levels. First, the event was not identified as a likely severe accident in the WASH-1400 analysis [15]. Second, the control room design and requisite instrumentation did not facilitate operator response or contribute to a timely recognition of the accident conditions. Third, operational experience from other nuclear power plants, that could have prevented the TMI-2 event, was not communicated to the industry. When coupled with the failure to address the high-level waste issue, containment and ECCS safety issues, fire protection effectiveness, and other reactor safety issues, the AEC/NRC record of success and effectiveness of nuclear regulation at the time of the TMI-2 accident certainly could be questioned.

The TMI-2 accident also led to the creation of industry groups to promote improved operational performance. These groups included national as well as international organizations. Two of the more significant groups were the Institute for Nuclear Power Operations (INPO) and the World Association of Nuclear Operators (WANO).

2.8 Salem ATWS Events

Unfortunately, additional reactor events with significant safety significance continued to occur

[16]. These events involved similarities to TMI-2 in that safety related equipment failed to function and maintenance issues contributed to the problem.

In February 1983, inadequate surveillance and testing of reactor shutdown circuitry at the Salem Nuclear Power Plant in New Jersey led to the failure of a reactor to trip when plant conditions warranted an automatic shutdown. Similar events occurred on two separate occasions. This condition known as an Anticipated Transient without Scram (ATWS) event placed the reactor in a condition that was outside its intended design since it did not shutdown as warranted by plant conditions.

The first ATWS event occurred on February 22. There was a lack of thorough and systematic review by the licensee that was necessary to fully understand the February 22 ATWS event. The post trip review was inadequate because plant management did not aggressively investigate the causes of the reactor trip events. In addition, there was a lack of questioning attitude, diligence, and attention to detail in the response to the reactor trips. In spite of the safety significance of these ATWS events, the NRC failed to investigate the failure and prevent plant startup [16].

A second ATWS event occurred on February 25. After consultation with the NRC, the licensee agreed to defer plant startup until a more comprehensive review of the event could be conducted.

The Salem ATWS events are significant because maintenance and testing did not reveal a significant safety issue associated with the reactor trip circuitry. In addition, INPO had previously identified a deficiency in Salem's preventive maintenance program [16], but the NRC failed to further investigate this issue. The NRC systematic appraisal of licensee performance for the period for the period September 1981 to August 1982 did not identify these maintenance problems. However, an NRC Resident Inspector Report in January 1983 noted the need for the licensee to develop a formal preventive maintenance program for reactor trip breakers. Both the NRC and INPO identified a significant deficiency that went uncorrected and contributed to the Salem ATWS events. The Salem event provides another example where the NRC's regulatory approach failed to detect a significant safety issue.

It is the author's view that both national organizations (e.g., INPO) as well as international groups including the International Atomic Energy Agency (IAEA) and WANO have a role in improving nuclear regulation. This role should be apparent in national as well as fully integrated international regulatory structures, and will be addressed in subsequent discussion.

2.9 Chernobyl

Chernobyl Unit-4 had an RBMK design that utilized a graphite-moderated core. Operator errors, an inadequately evaluated test procedure, and an unforgiving reactor design led to the 1986 accident at Chernobyl Unit-4 [3-6]. These factors contributed to a power excursion that resulted in violent reactor disassembly and severe fuel damage. The event ejected a portion of the core and graphite moderator from the reactor pressure vessel and released fission products directly into the environment. Chernobyl-4 released more fission products than the TMI-2 and FDNPS accidents and was exacerbated because no containment fission product barrier was included in the RBMK design [3].

Since the RBMK design was considerably different from US designs, the NRC emphasized that a Chernobyl-type accident could not occur in commercial US plants. In addition, US reactors had redundant safety systems, and a containment fission product barrier that would mitigate the release of fission products into the environment.

Nuclear critics used the Chernobyl-4 accident as a prime example of the hazards associated with nuclear power and the need for a more demanding regulatory approach. The Chernobyl-4 accident was another setback for nuclear power advocates and their desire to garner public support. Chernobyl's environmental impact provided sobering evidence that a major accident could occur, would have severe environmental impacts, and lead to evacuated areas that would remain uninhabitable for an extended time.

2.10 Towers Perrin Report

In 1997, the Towers Perrin consulting firm prepared a report for the Nuclear Energy Institute that was concerned that NRC policies and practices distracted plant management, undermined public trust, and increased operating costs. The report noted that the NRC did not make a significant effort to distinguish safety from non-safety issues and appropriately prioritize these items. It also claimed that the NRC's actions resulted in a diversion and dilution of licensee resources from the most important safety issues when plant problems arose.

The Towers Perrin report illustrated the importance of ensuring that regulators focused on safety significant issues. To achieve this focus, a tool for evaluating hazards in terms of their safety significance was needed.

The report's conclusions were consistent with growing interest within the NRC and the nuclear industry to utilize probabilistic risk assessments (PRA). PRA was viewed as a more effective method to assess hazards and to prioritize resources in order to more effectively address and eliminate their effects. The PRA approach contrasts with the conventional NRC deterministic analysis methodology and the "defense-in-depth" (DID) approach which was instrumental in preventing a significant radiological release during the TMI-2 accident. However, the NRC considered that the PRA approach was secondary to the DID philosophy, and it was used primarily to identify overly conservative regulatory requirements.

During its assessment of the PRA methodology, the NRC adopted a Maintenance Rule that required strong maintenance programs at commercial power plants. This rule was a positive safety development in view of significant weaknesses that included maintenance associated issues with the failure of reactor trip circuitry and failure of a pressure relief valve to close that established a loss-of coolant flow path during the TMI-2 accident.

Although risk-informed regulation offered potential benefits, it was not designed to detect the wide spectrum of safety issues that could occur at an operating nuclear power facility. This situation was demonstrated when a series of problems occurred at the Millstone Power Station in Connecticut. The safety issues at Millstone merited attention, but risk analysis would not necessarily identify them as priority safety issues.

2.11 Millstone Safety Allegations

Millstone safety allegations arose in the early 1990s when several plant employees claimed they were punished for raising safety issues [14]. The NRC investigated the employee concerns, but determined they were not of major safety significance and had been addressed by the licensee. Although the NRC imposed a $100,000 fine on the licensee, this action did not satisfy the critics of the NRC.

Media scrutiny intensified when new Millstone allegations were revealed. In 1993 and 1994, the NRC levied additional fines for procedural violations that were viewed as serious management issues. Millstone employees raised another issue related to outage practices involving offloading the reactor core to the spent fuel pool. This practice was in violation of NRC requirements that precluded a complete core offload for plants of the Millstone type.

The fuel offload issue involved specific plant safety requirements and the ability of the NRC to enforce those requirements. The continuing Millstone controversy was addressed in a 1996 NRC Inspector General Report that faulted the regulator for failing to recognize and impose corrective actions. The numerous Millstone issues illustrated the difficulty that the NRC had with plants that did not perform at the level required by agency standards and in correcting the associated issues in an effectively and timely manner. These issues also demonstrated that once a problem was identified, it would eventually be corrected. However, the regulator had not yet mastered the ability to develop regulations, inspection practices, and management controls that would anticipate problem areas or implement timely corrective actions [14]**.**

2.12 September 11, 2001, Attacks

The inability of the NRC to anticipate safety issues was again illustrated by the September 11, 2001, terrorist attacks on the World Trade Center in New York and the Pentagon near Washington, DC. These attacks raised two additional safety issues, which the existing licensing basis of some nuclear plants had not fully addressed [14].

The first issue involved the nuclear power facility's design basis as related to the effects of an aircraft impact on the integrity of the fission product barriers. A second issue was the plant's vulnerability to a terrorist attack resulting in a release of fission products to the environment. Accordingly, the NRC ordered a series of security measures, and again Congress challenged their rigor and effectiveness [14]. More importantly, the original design basis did not specifically address these challenges or ensure the full spectrum of natural and man-made events were evaluated before a plant license was issued.

The ramifications of the September 11 attacks added to the continuing theme regarding the adequacy of the facility design basis. In particular, the adequacy of the containment building and spent fuel pool designs to withstand the impact of a contemporary commercial aircraft was questioned. In September 2004, the NRC reported that an aircraft strike at a nuclear power plant could cause a radioactive material release [14]. In addition, a 2005 National Academy of Science report concluded that a successful terrorist attack would be difficult to achieve, but would be a credible threat [14]. This report argued that there was no regulatory requirement to protect the facility from this type of hazard. Once again, the NRC was involved in a design basis controversy.

2.13 Davis-Besse Reactor Vessel Head Erosion

During the aircraft attack controversy, a serious operational issue arose at the Davis-Besse plant [17], which is a pressurized water reactor. In 2002, an inspection of the upper reactor vessel head discovered significant material degradation, which created an American football-sized cavity. This degradation was caused by borated water that leaked onto the reactor vessel head through cracks in a control rod drive mechanism nozzle and the weld that attached the nozzle to the reactor pressure vessel head. The erosion of the reactor vessel head structural material involved about 32 kg of steel, which only left the thin (about 1 cm) stainless steel cladding intact as the only pressure boundary preventing a loss-of-coolant accident.

Both the utility and NRC failed to identify the erosion and take timely action to correct the conditions that initiated and continued the erosion process. The NRC's failure is also of concern since the corrosion issue was related to a previous NRC inquiry regarding the cracking of control rod drive mechanism nozzles [14, 17].

In August 2001, the NRC instructed pressurized water reactor owners to inspect these nozzles by December 2001. However, the inspection date could be delayed if the NRC staff judged the specific plant's risks were acceptably small. The operators of Davis-Besse requested a delay in the inspection date until a scheduled first quarter 2002 outage. The NRC staff approved the request and determined that the plant could be safely operated until that date.

The discovery of the significant reactor vessel head erosion suggested that the NRC was in error in granting an extension to perform the requisite inspection after the December 2001 due date. The NRC Inspector General (IG) responded to a Union of Concerned Scientists charge that the NRC failed to adequately regulate the Davis-Besse plant and that a loss-of-coolant accident could have resulted from failure of the reactor pressure boundary. The IG strongly criticized the NRC's safety performance, and found that the agency had considered the financial impact to the licensee rather than making public health and safety its highest priority. Although the NRC disputed the IG's safety conclusion, it did conclude that a break in the cladding could have led to a loss-of-coolant accident and that the corrosion of the reactor vessel head was an enormous failure of both the NRC and operating utility. However, the NRC denied that the cladding failure would have led to a massive release of radioactive material to the environment. Defense-in-depth was emphasized as an effective means to prevent the release of fission products to the environment.

The Davis-Besse event is troubling from a regulatory perspective because many indications of the event were present but were not recognized by operating utility and regulatory personnel. These indications include radiation monitoring system filter systems being clogged by boric acid and corrosion particles, the buildup of boric acid deposits on containment air cooler fins, and boric acid deposits on the reactor vessel head [17]. This event was not prevented because the NRC, plant personnel, and industry groups failed to adequately review and analyze relevant operating experience; plant personnel failed to ensure that safety issues received proper attention; and the NRC failed to include known or available facility information into its assessments of Davis-Besse performance.

Once again, the NRC provided sound corrective actions after the event. However, it failed in its ability to be proactive and preclude another significant event in spite of multiple indications that a significant corrosion issue existed.

2.14 Yucca Mountain High-Level Waste Repository

The Yucca Mountain High-Level Waste Repository Site is approximately 100 miles northwest of Las Vegas, Nevada. This site is intended to store high-level waste, including spent fuel from commercial power reactors, in an underground facility that has stable geologic characteristics [3, 14]. The NRC would license the facility, and the site was selected and designed by the Department of Energy (DOE).

Selection of the Yucca Mountain site by the DOE was reminiscent of the previous Lyons site selection process since significant opposition was expressed by the host state, and litigation followed the site selection. In spite of this opposition, the Nuclear Regulatory Commission received an application from the Department of Energy in 2008, for a license to construct and operate the first US geologic repository for high-level nuclear waste at Yucca Mountain. This submittal was a significant milestone, because it transferred focus from DOE's efforts to select a repository site to the NRC's review of the repository design to determine its suitability as a high-level nuclear waste storage facility.

The NRC's regulatory process involves technical design reviews and hearings that are conducted concurrently. Technical licensing reviews assess the merits of the repository design. Adjudicatory proceedings assess challenges by the public and other stakeholders regarding the technical and legal aspects of the DOE license application. Based on the results of the licensing review and the hearings, the Commission determines the appropriateness to authorize construction of the Yucca Mountain repository.
In 2011, the NRC regulatory process was interrupted by a variety of factors that included many of the previously raised Lyons site objections. The Yucca Mountain issue is before the US Court of Appeals and may eventually reach the US Supreme Court. It appears that the DOE and NRC have repeated the errors of the AEC in its attempt to license the Lyons high-level waste repository.

The examples of this section summarized events that revealed weaknesses in the US regulatory process. In the next section, these weaknesses are reviewed within the context of the FDNPS accident.

3.0 Fukushima Daiichi Accident

In March 2011, the Fukushima Daiichi Nuclear Power Station in Japan, consisting of six boiling water reactors (BWR), was struck by a significant seismic event and subsequent tsunami that led to fuel damage in a number of reactor vessels and possibly in multiple spent fuel pools [7-12].

The Fukushima Daiichi accident occurred when the facility encountered an earthquake and resulting tsunami that exceeded its design basis assumptions [7-12, 18-22]. As s result of this accident sequence, the defense-in-depth safety systems failed to provide the intended margin of safety and rapidly led to the loss of all fission product barriers with releases of radioactive materials from multiple units into the environment. From a regulatory perspective, failures of both the design basis foundation and defense-in-depth philosophy suggest that a reevaluation of the regulatory basis for licensing nuclear plants is warranted. A review is also warranted because the recommended corrective actions for US plants that were derived from analyses of the Fukushima Daiichi event continue to utilize the defense-in-depth philosophy to preclude major events [18].

There were a number of national and international reviews of the Fukushima Daiichi accident [18-22]. Although each has a unique and valuable perspective, this paper focuses on the NRC review [18] and review by the Japanese government [21].

3.1 NRC Review

The NRC review of the Fukushima Daiichi accident is important because the Mark I Design utilized at the FDNPS is also utilized in the US. In its review, the NRC focused on the defense-in-depth actions, training, procedures, and programs. The NRC continued to follow its basic regulatory philosophy and did not propose any new approaches.

Following its review of the FDNPS event, the NRC recommended that licensees take a number of actions [18, 22]. These actions direct licensees to address a number of areas including:

1. *Design basis seismic and flooding systems, structures, and components (SSCs)* – Seismic and flood protection SSCs need to be reevaluated and upgraded as necessary;
2. *Station blackout (SBO) mitigation capability* – The capability of SBO systems to mitigate design basis and beyond design basis events needs to be reevaluated and strengthened as necessary. Emergency preparedness equipment must be capable of addressing multiunit and SBO situations, and facility emer-

gency plans must address prolonged SBO and multiunit events;
3. *Mark I and Mark II BWR containments* - A reliable hardened vent must be provided for these reactor types;
4. *Spent fuel pools* – An installed seismically qualified means to spray water into the SFP must be provided. The enhanced SFP water addition capability must include associated instrumentation and safety related power. Safety related instrumentation must be capable of withstanding design basis natural phenomena to monitor spent fuel parameters including water level, temperature, and radiological conditions; and
5. *Onsite emergency response capabilities* - Emergency operating procedures, severe accident management guidelines, and emergency damage mitigation guidelines must be strengthened and integrated. More realistic training and exercises must be provided for all staff expected to implement these guidelines during an emergency.

These actions again reflect the NRC's history of providing credible corrective actions after an event occurs. In an ideal regulatory approach, these recommendations should have been included in the original licensing basis of nuclear power plants. Consistently failing to have an inclusive design basis and reacting to events by only issuing corrective actions following the event is not a successful, long-term regulatory approach. It further suggests that a change in regulatory approach is warranted, and that previous assumptions and practices require significant revision.

3.2 Japanese Diet Commission Review

The results of the Japanese Diet Commission review of the Fukushima Daiichi accident [21] provides a somewhat different perspective and focuses on a regulatory process perspective. This is in contrast to the NRC recommendations that focused on plant systems, programs, and staffing. The major conclusions of the Japanese Diet Commission review include:

1. The accident's root causes were the organizational and regulatory systems that supported faulty rationales for decisions and actions;
2. The operating utility was too quick to cite the tsunami as the cause of the nuclear accident and deny that the earthquake caused any damage;
3. Organizational problems within the utility (e.g., level of knowledge, training, and equipment inspection) limited accident response;
4. Emergency response issues existed because roles and responsibilities were not well-defined;
5. Regulators failed to implement adequate evacuation plans, and an inadequate crisis management system contributed to public confusion during the evacuation;
6. The government and regulators are not fully committed to protecting public health, safety, and welfare of the evacuees;
7. The safety of nuclear energy in Japan cannot be assured unless the regulatory process is changed by eliminating its insular attitude of ignoring international safety standards;
8. The operating utility did not fulfill its responsibilities as a private corporation, and its relationship with the regulators was used it to weaken proposed safety regulations;
9. The latest technological findings from international sources should be reflected in existing nuclear energy laws and regulations; and
10. Root causes must be addressed and preventive measures implemented to preclude future accidents.

The Diet Commission conclusions are not inconsistent with the analysis of Wang and Chen [13]. It also has a number of regulatory items that are appropriate for consideration of future US and international regulations. In particular, recommendations 1, 7, and 9 will be addressed in subsequent discussion.

4.0 US Regulatory Improvements

A number of improvements in the US Regulatory process are in development. These include a state-of-the-art consequence analysis and proposed risk-informed regulatory management framework. Although these are improvements and offer the potential for increased safety, they do not offer a new regulatory approach or significant departure from current regulatory philosophy.

4.1 State-of-the-Art Reactor Consequence Analyses

The US regulatory history illustrates numerous areas where improvement could have been achieved, and opportunities were missed. Fortunately, the NRC has attempted to learn from previous failures and to improve the US regulatory process. Part of the improvement process is the desire to enhance analysis tools for assessing events in order to focus attention on safety significant issues.

Improvements in the NRC's regulatory process is illustrated by the most recent generation of reactor phenomena and consequence models designed to develop the best estimates of the offsite consequences from potential severe reactor accidents. The NRC's most recent methodology is called the State-of-the-Art Reactor Consequence Analyses (SOARCA) [22]. This project evaluates plant improvements and changes that were not reflected in earlier NRC models. SOARCA includes system improvements, improvements in training and emergency procedures, and emergency response and security upgrades. It also incorporates the effects of plant changes such as power uprates and higher core burnup.

The SOARCA methodology was intended to be comprehensive and was initiated prior to the Fukushima Daiichi accident. As such, the SOARCA and FDNPS accident offer a benchmark to its predictive capability. It is interesting to note that several classes of accidents that were integral to the FDNPS events were not considered as part of the SOARCA. These omissions include:

1. Multiunit Accidents – The FDNPS is a six-unit facility. Hydrogen explosions occurred in multiple units and the operating units (Units 1, 2, and 3) affected Unit 4, which was defueled at the time of the accident. This omission does not provide confidence in the adequacy or completeness of the existing SOARCA since it omitted a key aspect of the FDNPS accident;
2. Low Power and Shutdown Unit Accidents – FDNPS Unit 4 had its entire fuel inventory offloaded to its spent fuel pool. The events in Units 1, 2, and 3 led to a hydrogen explosion in Unit 4 that severely damaged its reactor building and possibly damaged the spent fuel pool and its included fuel. The extent of fuel damage in the Unit 4 spent fuel pool is uncertain. Accounting for the potential for a defueled unit to suffer a severe hydrogen explosion and its impact on a defueled unit's spent fuel pool were not considered in the SOARCA;
3. Extreme Seismic Event that Led Directly to Gross Containment Failure with Subsequent Core Damage – The extent of damage at the FDNPS attributed to the seismic event is unclear [12, 21]. Specific seismic inspections will not be possible until facility radiation levels are significantly reduced. However, the tsunami caused by the massive earthquake facilitated the station blackout condition and subsequent core damage, but this sequence was not considered in the SOARCA. This omission challenges the adequacy of the SOARCA to anticipate serious events that could damage fission product barriers and lead to a significant release of radioactive material to the environment; and
4. Spent Fuel Pool Accidents – The hydrogen explosions in Units 1, 3, and 4 damaged their respective reactor buildings. The extent of the damage to the spent fuel pools is uncertain, but the Unit 4 pool appears to have been the most heavily damaged. Debris from the hydrogen explosions fell into the pools and may have mechanically damaged fuel and associated safety systems. The SOARCA did not predict the sequence of events associated with the FDNPS that is another omission in the NRC's ability to anticipate and preclude events.

4.2 Risk-Informed Regulatory Framework

In 2011, the NRC reviewed the need for modifications to its regulatory framework in order to enhance safety and improve regulatory consistency across the various regulated programs [23]. This effort focused on a more comprehensive, risk-informed, performance-based regulatory approach for power reactors, research reactors, materials, low-level waste, high-level waste, uranium recovery, fuel cycle, spent fuel storage, and transportation. The risk-informed framework envisioned an approach that could be in existence within the next 10 – 15 years.

Although the risk informed approach has merit, it focuses on a variety of programs. One must ask the question, if the NRC or an international regulator would be more effective by solely focusing on nuclear power plants. The NRC's global approach can be contrasted with INPO and WANO that have a singular, nuclear power focus. This focus has been successful in improving nuclear power plant performance, and should be considered in future regulatory models.

The risk-informed study concluded that the defense-in-depth concept remains valuable, but it is not uniformly applied and more guidance is required for its optimization. Optimization would be enhanced using a risk-informed and performance-based regulatory approach. The NRC's risk management approach recognizes that adequate protection of public is not synonymous with absolute plant safety.

The report reaffirms that the concept of design-basis events and design-basis accidents continues to be a sound licensing approach. However, the design basis concept has not been refined to incorporate the operating history of power reactors and a variety of analysis techniques including probabilistic risk assessment. NUREG-2150 [23] also suggested the creation of a design-enhancement regulatory category for the treatment of beyond design-basis accidents. In addition, the methodology used to assess the frequency and magnitude of external hazards should be determined using both deterministic and PRA techniques.

5.0 Nuclear Regulatory Options

The previous discussion provides historical examples that the NRC model has not been completely successful in precluding accidents or off-normal events. Planned NRC improvements follow the conventional regulatory model and do not offer the likelihood of sufficient improvements in predicting or preventing the next major reactor accident. These shortcomings include licensing issues associated with underestimating the design basis (e.g., protection against beyond design basis earthquake events illustrated by the Fukushima Daiichi accident), design issues (e.g., TMI-2), and maintenance issues during operations (e.g., Salem ATWS events). The historical review also illustrates that the agency has incrementally improved its performance and continues to evolve and improve its regulatory model. However, given the nature of the TMI-2 and FDNPS accidents change is needed to better mitigate future events.

If the current NRC regulatory model is not the answer to a proactive approach to prevent or at least significantly mitigate future accidents, then other broad scope options for achieving this goal are possible. The items noted in subsequent discussion are not necessarily complete, but serve to illustrate the types of options that are available for minimizing the probability of a severe reactor accident.

The first option is the Swiss-German approach of abandoning nuclear power. This is essentially a default option that would eventually eliminate nuclear power as an energy source. It could be a consistent worldwide option, but it imposes significant economic penalties.

Option 1 does not guarantee safety since plants would continue to operate for a limited time before being decommissioned. In addition, the reactor's fuel would require disposition and subsequent regulatory attention. The problems associated with licensing the Yucca Mountain repository and the economic loss of electrical generating capacity suggests that this option presents significant challenges. However, it is achievable in the near-term.

The second option is a modification of the first proposal. Option 2 is a gradual elimination of nuclear power plants by reducing their authorized power levels. A power reduction reduces the severity of an accident, but does not necessarily reduce accident frequency. This option could be implemented gradually to permit other power options to be utilized, but an economic impact would result. The issues of decontamination, decommissioning, and spent fuel disposition would remain as they did for the first option. As with Option 1, this option is also achievable in the near-term.

The third option is to continue to operate nuclear power plants using the current US regulatory framework. This approach has prevented additional major US accidents, and could be justified on that basis. Option 3 would continue to incrementally improve safety by adopting the lessons learned from operational events and accidents (e.g., implementing the Fukushima Daiichi Task Force Recommendations [18]) and the improved assessment approaches [22, 23]. It would also include the industry's proposed flexible and diverse or FLEX strategy [24], which provides portable equipment for nuclear plants to maintain cooling capability and power during severe natural events. Following this approach, overall plant safety would improve, but it would not preclude the occurrence of new or previously unidentified events (e.g., events such as the Salem ATWS or the Davis-Besse reactor vessel head erosion).

A fourth approach is similar to the previous option, but replaces the Generation II reactors with Generation III designs [25] after their licenses expire. The Generation III plants would be located in low population density areas. Although Option 4 requires the construction of additional transmission and generation facilities, it would reduce probability of a serious event since the Generation III plants utilize passive safety systems and have a lower core damage frequency than Generation II plants [25]. The impact of a serious event affecting the public would also be reduced since the reactors would be located in low population density areas. As such, the disruption on nearby populations would be minimized. This option would improve safety over time, but in the near-term Option 4 is essentially Option 3.

A fifth approach would restructure the regulatory authority to solely focus on reactor safety. This would mean that regulatory jurisdiction for hospitals, universities, enrichment facilities, and commercial firms using or transporting radioactive material would be transferred to States or other government agencies. Having a regulatory organization solely devoted to reactor safety would have the benefit of not diluting the reactor safety mission with other aspects of radioactive materials and associated events.

A sixth approach involves abandoning the contention that the defense-in-depth methodology is a basis for reactor safety. The defense-in-depth systems would be maintained, but would not be the ultimate justification for reactor safety. Option 6 admits that events outside the original facility design basis can occur, safety systems can fail, and unanticipated and beyond design basis events can occur. Using this approach, the current regulatory system would continue to function to improve performance and safety and manage design basis events. However, beyond design basis events or conditions outside the facility design basis (e.g., a Fukushima Daiichi accident type) would be managed from a new hardened emergency facility collocated with the existing reactor site. The hardened facility would be the ultimate emergency response facility designed to operate and mitigate an accident in conditions encompassing worst-case events.

As envisioned by the French in its hard core concept [26], the hardened facility would provide control, power, and shutdown systems to protect and preserve the three fission product barriers. The new hard core facility would house the requisite power and shutdown systems including pumps, water supplies, and power systems to ensure the core and spent fuel pool did not release radioactive material to the environment. The concept would require a significant expenditure [26], but would add a measure of safety beyond that provided by the currently accepted regulatory philosophy.

It is the author’s view that none of these individual broad scope proposals is the answer to improving US nuclear regulation or forming the basis for international regulations. Future nuclear regulation must consider the current regulatory approach and evolve to a desired end state that includes elements of some of the previous options.

Although there are numerous possible solutions for optimizing reactor safety, the following approach is proposed as an initial model subject to revision. It maintains the most positive aspects of current regulatory philosophy while transitioning to a more robust regulatory framework. This approach also has the potential to form the basis for an international approach since it adopts elements of recommendations proposed by the Japanese [21], French [26], and US regulators [18, 22, 23].

The author proposes the following phased approach to improve safety at commercial power reactors. This approach includes near-term (< 15 years), intermediate-term (15 – 30 years), and long-term transitions (> 30 years). The proposed near-term regulatory changes include:

1. Continue the operation of nuclear power facilities following the NRC model. Improve reactor and regulatory performance by further developing the SOARCA [22] and NUREG-2150 [23] proposals. This is essentially the current NRC model with planned improvements;
2. Limit the NRC regulatory role only to include power reactors. Its other regulatory responsibilities should be transferred to the States or other agencies;
3. Utilize third-party groups such as INPO, WANO, and the IAEA to provide operational assessments of reactor performance and an outside reactor safety perspective. Require that national regulatory agencies evaluate and formally respond to INPO, WANO, and IAEA recommendations. National regulators would have the final decision to accept or reject these recommendations;
4. Facilitate the international exchange of utility personnel to broaden operating and maintenance experience;
5. Implement the FLEX concept [24] as an interim measure to enhance reactor safety;

6. Develop the French hard core [26] concept including appropriate design and cost analyses for all operating reactors. International participation and stakeholder groups should be involved in this process. Stakeholders should include groups that have previously expressed safety concerns (e.g., the Union of Concerned Scientists). The applicability of incorporating the FLEX approach to supplement the hard core facility design should be determined;
7. Develop a consistent international regulatory approach using the NRC as the initial model. All nations with operating nuclear reactors or plans to initiate nuclear plant construction should participate and industry groups (e.g., INPO and WANO) should have a significant role in influencing the regulatory process.

These near-term actions advance existing concepts and form the basis for future international regulations. Developing nations should also be included in the near-term process to ensure that new nuclear power programs have a firm safety foundation.

Near-term Items 1 – 6 are significant steps, but Item 7 is crucial. Since a major accident affects all nuclear plants, it is essential that all nuclear reactors have a robust and sustainable regulatory foundation that provides the public and stakeholders an avenue to participate and influence regulatory issues in a positive manner.

The intermediate-term options assume that the near-term options have progressed and that the basis for international regulations has been achieved. With these assumptions, the intermediate-term actions include:

1. Incorporate the French hard core [26] concept into existing and future reactor designs;
2. Use Generation III designs with hard cores for all new facilities and locate them in low population density areas. The licensing basis of these facilities should include a consistent international approach.
3. Implement the international regulatory approach developed in the near-term.

The intermediate-term actions involve difficult choices, particularly incorporating the hard core concept into new and existing facilities. If properly managed, the hard core concept provides a methodology to build public and stakeholder consensus for nuclear reactors and add a degree of safety beyond that existing today.

If the near-term and intermediate-term actions are accomplished, the long-term goal of developing and operating Generation IV Reactors [25] will move closer to reality. The major long-term action is to locate proposed Generation IV reactors with hard cores and their support facilities in remote, low population areas. This would include the Generation IV plants and their associated reprocessing facilities.

The implementation of Generation IV designs require addressing significant issues (e.g., nuclear proliferation), but it enhances the use of uranium and plutonium resources, and minimizes many of the issues associated with high-level waste storage.

This proposed operating and regulatory concept would require agreement of government, industry groups, the public, and groups typically opposed to the current approach (e.g., the Union of Concerned Scientists and the National Resources Defense Council). Reconciling the viewpoints of these groups would be a very significant challenge. However, it would place nuclear power in a position of consensus that could be advanced using an open dialogue with new safety ideas freely exchanged. It also has the potential to enhance reactor safety performance over the current approach.

The facility operating and regulatory costs would be uncertain and require further analysis. However, given the aftermath of the Fukushima Daiichi accident, the cost issue should be included in discussions with stakeholders. These discussions should also review the merits of nuclear power in terms of global climate change and alternative power generating technologies.

6.0 Conclusions

A review of US regulatory history suggests that major weaknesses have been minimized and improvements have occurred. However, the process has not been completely successful in eliminating significant events and programmatic weaknesses remain. An evaluation of the Fukushima Daiichi accident and subsequent recommendations in terms of the US regulatory history suggest an alternative regulatory model. This model evolves with time and incorporates international input. It also offers the potential for an international regulatory framework and minimizing opposition to the deployment of nuclear generation.

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