MODULE 11A: PERIODIC SAFETY REVIEW

Overview

The practice of assessing the safety of operating nuclear power plants through the use of periodic safety reviews (PSRs) is well established among the International Atomic Energy Agency (IAEA) Member States. It is mandatory in several Member States (Refs. 1 – 4) and is a part of the regulatory framework at some Member States (Refs. 5 – 7). Many articles and guidance documents have been published on PSRs over the years (Refs. 8 – 12). In 2003, IAEA published a safety guide, NS-G.2.10, "Periodic Safety Review of Nuclear Power Plants" (Ref. 13), to provide guidance on how to conduct PSRs. The reviews are conducted by the plant staff and include an assessment of plant design and operation against current safety standards and practices in 14 areas (known as safety factors[1]). The objective of a PSR is to "ensure a high level of safety throughout the plant's operating life" by systematically "assessing the cumulative effects of plant ageing, plant modifications, operating experience, technical development and sitting aspects."

The U.S. Nuclear Regulatory Commission (NRC) agrees with the IAEA premise that vigilant oversight and ongoing reviews are essential in ensuring safety throughout the life of the plants. The United States historically has engaged the international community through the development of the process to conduct PSRs (Ref. 14) to ensure the objective of maintaining safety throughout the entire operating life of a plant. The NRC also accomplishes this objective through its comprehensive set of regulations, inspections and safety review programs.

The U.S. regulatory structure was well established when the PSR approach was being developed. During the formulation of the License Renewal Rule in the early 1990's, the NRC considered the concept of performing a comprehensive review of a plant to bring it closer to the current standards (a goal of the PSR approach). The Commission did not adopt this approach in part because it believed that the NRC's robust and mature programs, including the onsite resident inspector program, generic issue identification, and systematic evaluation process, afforded adequate protection to the public[2]. The Commission has reviewed the PSR and U.S. system since and, as reiterated in several U.S. National Reports to the Convention on Nuclear Safety (Refs. 17 – 19), has maintained that the safety functions of PSRs are achieved by the U.S. system. This paper presents an overview of the U.S. regulatory structure, salient features of the U.S. regulation consistent with the PSR approach, and a comparison between the safety factors in the PSR Safety Guide, and the comparable U.S. activities. The paper presents one

---

1 The fourteen safety factors are: (1) plant design, (2) actual condition of systems, structures, and components, (3) equipment qualification, (4) ageing, (5) deterministic safety analysis, (6) probabilistic safety assessment (PSA), (7) hazard analysis, (8) safety performance, (9) use of experience from other plants and research findings, (10) organization and administration, (11) procedures, (12) human factors, (13) emergency planning, (14) radiological impact on the environment. In addition, a global assessment integrates all the individual assessments together.

2 The License Renewal Rule (Ref. 15) and NUREG-1412, "Foundation for the Adequacy of the Licensing Bases – A Supplement to the Statement of Considerations for the Rule on Nuclear Power Plant License Renewal (10 CFR 54) Final Report," issued 1991 (Ref. 16), provide a more complete discussion of the history of the License Renewal Rule.

recommendation to further inform the U.S. program with the IAEA PSR process. This paper proposes that the recommendation be evaluated in conjunction with the Integrated Regulatory Review Service (IRRS) discussions, and that any potential agency actions be presented to the Commission, as appropriate.

<u>NRC Policy/Program</u>

Background/History

The development of atomic energy for peaceful purposes while strengthening free competition of private enterprise has always been understood as a part of the U. S. nuclear framework since the initial promulgation of the Atomic Energy Act of 1946 (Ref. 20). The U.S. Atomic Energy Act of 1954, as amended (the Act), opened nuclear technology to commercial enterprise and authorized the Atomic Energy Commission (the NRC's predecessor) to establish regulations that provide reasonable assurance of adequate protection for public health and safety in the use of radioactive material (Ref. 20). As a point of discussion, the NRC is not required to continuously improve the level of adequate protection because the Act itself is silent on the concept of continuous improvement. However, as summarized in the paper, the U.S. nuclear industry itself, along with the comprehensive independent oversight by the federal regulator, has and continues to make safety improvements.

The Act limits the initial license period of commercial power reactors to 40 years. The current operating nuclear power plants were licensed under a two-step process in which the agency first issued a construction permit and then an operating license. The applicant submits a preliminary safety analysis as part[3] of the construction permit application. The NRC reviews the application and documents its findings and emergency planning in a safety evaluation report (SER). The Advisory Committee on Reactor Safeguards (ACRS), an independent advisory group of technical experts, reviews every construction permit application and the NRC's safety evaluation. The ACRS then reports its reviews to the NRC's Commission. The second step of the licensing process is the application for an operating license. The application contains a final safety analysis report (FSAR) and an updated environmental report. The FSAR includes the plant's final design, safety evaluation, operational limits, anticipated plant response to postulated accidents, and emergency plans. The NRC documents its review of this application in a final SER. ACRS reviews each operating license application and the NRC's final SER in a public meeting. A *Federal Register* notice is published to give any person whose interests might be affected by the proceeding the opportunity to petition the NRC for a hearing. All documentation, except those parts that are proprietary, is publicly available.

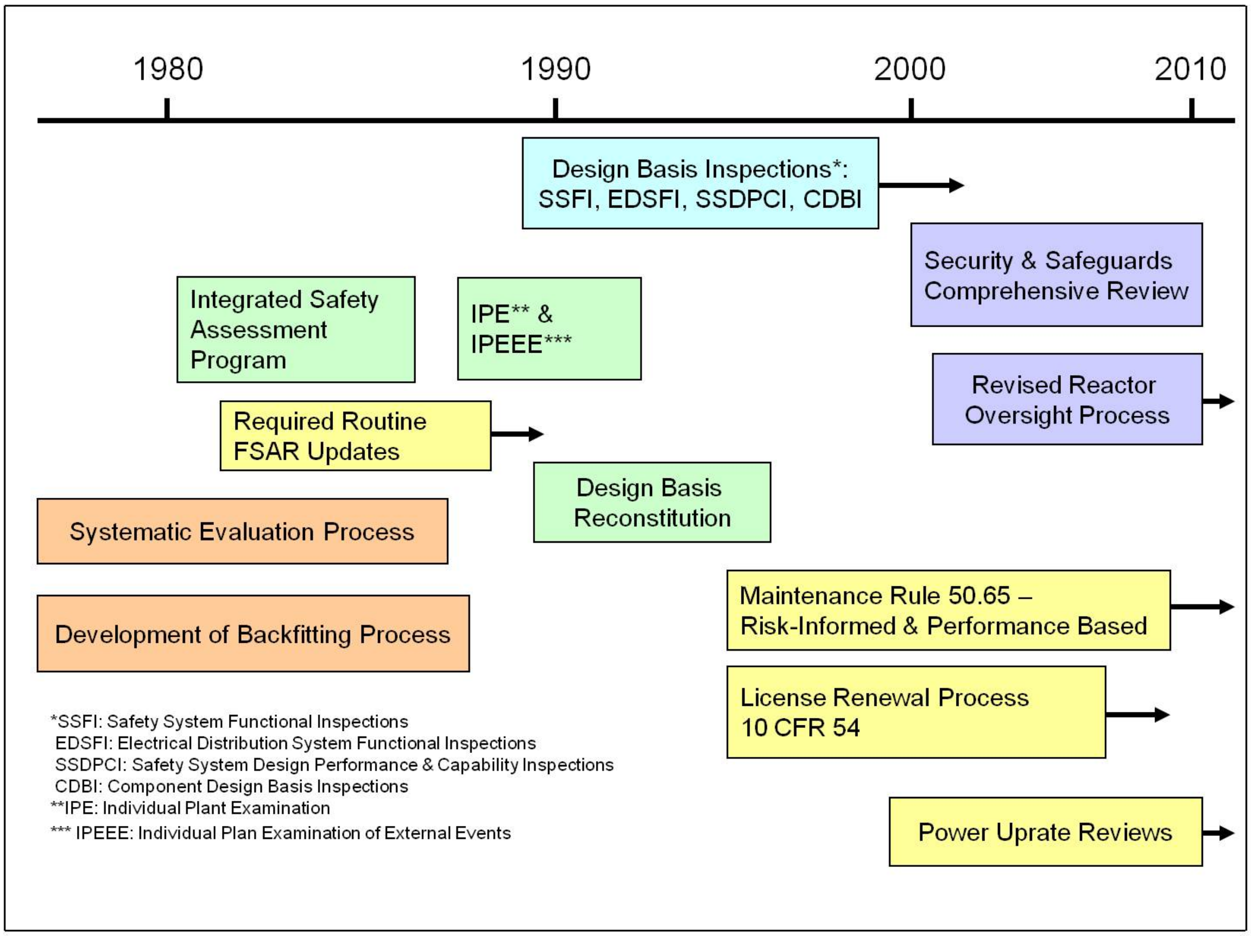
NRC regulations (Ref. 21) allow the plant license to be renewed for up to 20 years. The license renewal process proceeds along two tracks: one for review of safety issues, and another for review of environmental issues. An applicant provides the NRC with an evaluation that addresses the technical aspects of plant ageing and describes how it will manage those effects during the period of extended operation. The applicant also prepares an evaluation of the potential impact on the environment during the period of extended operation. The NRC staff documents its review results in an SER and a Supplemental Environmental Impact Statement. ACRS reviews every license renewal application and related NRC staff evaluations, and forwards its recommendation to the Commission.

---

[3] The construction permit application also requires an environmental review and financial and antitrust statements.

In the mid 1970's, the NRC recognized the importance of assessing the adequacy of design and operation of currently licensed nuclear power plants, motivated by the fact that deviations from applicable current standards that may have been approved after those plants were licensed might be safety significant. Consequently, the NRC initiated the Systematic Evaluation Program (SEP) in 1977. In 1984, the NRC staff presented 27 SEP lessons learned to the Commission as a part of a proposal for an Integrated Safety Assessment Program. The SEP process was subsequently transformed into the Integrated Safety Assessment Program pilot. Later, the NRC transferred many issues to the established Generic Safety Issues Program. In the late 1980s and throughout the 1990s, the NRC continued its efforts to strengthen its regulatory infrastructure and ensure continued safe operations of commercial nuclear power plants through inspections, assessments, and where appropriate, establishment of new generic requirements. Figure 1 shows a brief timeline of major U.S. regulatory reviews.

Figure 1. Timeline of Significant U.S. Regulatory Reviews



Current Policy/Program

Following the issuance of the initial license and during the period of extended operation, the NRC continues providing oversight of plant operations to verify that they are being conducted in accordance with the NRC regulations. The oversight includes daily monitoring by the on-site resident inspectors and periodic regional inspections, operating experience evaluations, generic issue resolution, biennial updates of the licensing basis, and imposition of new requirements. The following NRC programs ensure the safety of the operating plants and correspond to IAEA

PSR safety factors

I. *Operating Experience*

<u>Associated safety factors</u>: *several; including use of experience from other plants and research findings, actual conditions of SSCs,* and *plant design*

The NRC understands that effective use of operating experience from domestic and international plants is important in enhancing safety of plant operations. Therefore, the NRC has established and commits to a robust ongoing Operating Experience Program that collects, evaluates, communicates, and applies the operating experience to prevent significant events and inform NRC decision-making. To that end, the Operating Experience Program processes reactor operating experience in a risk-informed and timely manner to ensure the agency's rulemaking, licensing, oversight, and incident response programs are able to continuously learn from the relevant operating experience and effectively apply the lessons learned. Key features of the Operating Experience Program include: (1) reporting of events, (2) screening of events, (3) investigation of events, (4) in-depth analysis of safety-significant events, (5) consideration of trends, (6) recommended actions, (7) dissemination of information, (8) continuous monitoring programs, and (9) storage, retrieval and documentation of information.

The NRC identifies operating experience in license event notifications, other reporting requirements in the regulations, and through U.S. review of reports from international facilities. The focal point is daily review at each nuclear power plant by two to three resident inspectors whose full-time job is inspecting and assessing plant performance. These individuals, supplemented by other region-based specialists and inspectors, monitor the plant operation and ensure that plant operating experience is promptly fed back to the agency's technical and programmatic offices for future evaluation. One example of how feedback from the operating experience informs the U.S. regulatory activities is in the area of reactor license renewal. The NRC has developed several documents to aid in the effective and efficient evaluation of license renewal applications. One of the reference documents used by the NRC staff is the "Generic Aging Lessons Learned (GALL) Report" NUREG-1801, Volumes 1 and 2 (Ref. 22). The GALL Report provides guidance to staff on how to evaluate the aging management programs (AMP) proposed by plant operators to ensure the safety of the plants during the extended period of operation. Previously the scope of the electric power cables AMP was limited to cables with 2 kV to 35 kV range. Recent industry experience has shown that lower voltage cables, when exposed to adverse environmental conditions for which they were not designed for, experience reduced insulation resistance, thereby leading to increased likelihood of cable failure. Therefore, the staff expanded the scope of this AMP to include cables carrying voltage greater than 480 V in the next update of the GALL Report.

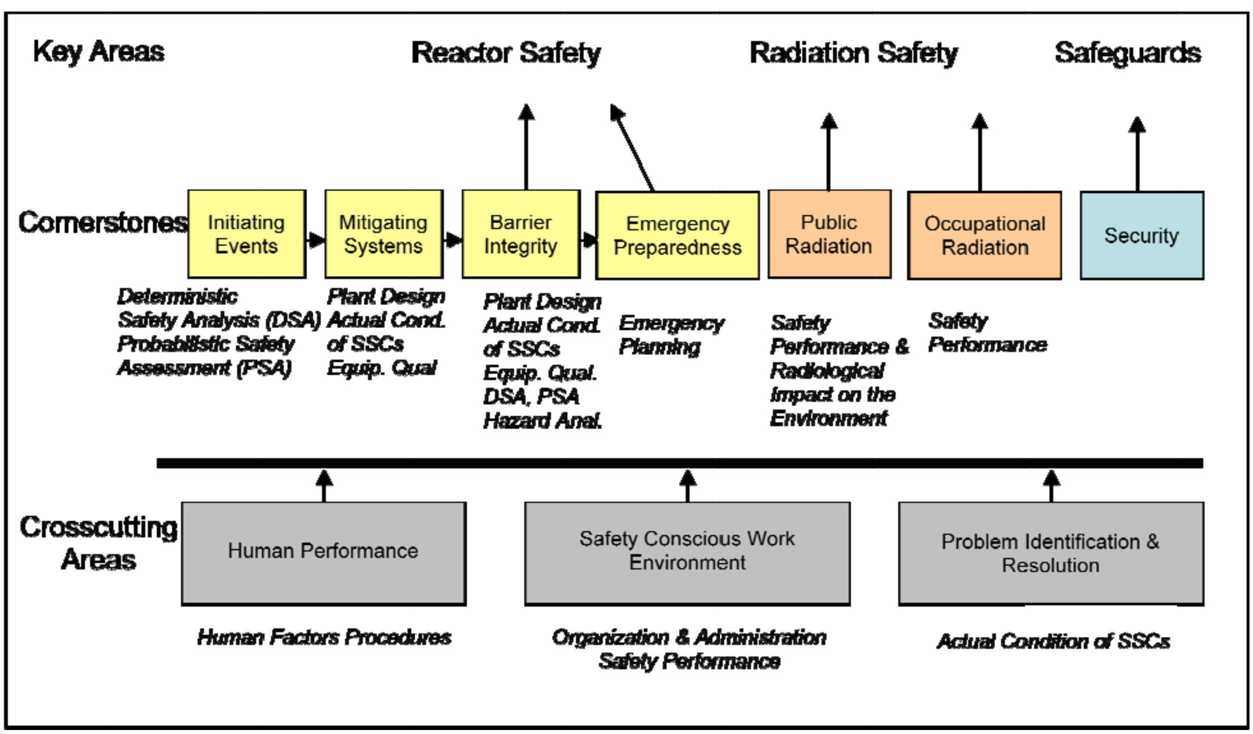
The United States benefits from having 104 operating nuclear units from which to draw operating experience and participates in many international forums to collect and better understand operating experiences from other countries. Sources of international operating experience include information from the International Nuclear Event Scale and the Incident Reporting System. The NRC staff systematically screens all nuclear power plant-related operating experience for safety significance and generic implications. (The IRRS Module 11B *Feedback of Operating Experience* paper explains the NRC operating experience program in more detail.)

II. *Reactor Oversight Process*

> <u>*Associated safety factors*</u>: *plant design, actual condition of SSCs [structures, systems and components], equipment qualification, ageing, deterministic safety analysis, probabilistic safety assessment, hazard analysis, safety performance, use of experience from other plants and research findings, organization and administration, procedures, human factors, emergency planning, and radiological impact on the environment*

The Reactor Oversight Process (ROP) is the NRC's program to inspect, measure, and assess the safety performance of commercial nuclear power plants and to respond to any decline in performance (Ref. 23). The objective of the ROP is to monitor plants' performance in three key areas: (1) reactor safety, (2) radiation safety, and (3) safeguards. The oversight process focuses on seven "cornerstones" which support the safety of plant operations in the three key areas. In addition to the cornerstones, the ROP incorporates three cross-cutting areas that affect and are therefore part of each of the cornerstones. Figure 2 shows the ROP key areas, cornerstones, and the cross-cutting areas, as well as the corresponding IAEA PSR safety factors (noted in italics).

Figure 2. NRC ROP Framework with Corresponding IAEA Safety Factors



Every cornerstone addresses the adequacy of procedures. For instance, the inspection procedure for adverse weather protection requires verification of the procedures associated with summer readiness of offsite and alternate power, seasonal extreme weather conditions, impending adverse weather conditions, and external flooding. As discussed above and shown in Figure 2, the NRC ROP addresses all fourteen IAEA PSR safety factors. Several safety factors are supplemented by other NRC programs. For example, the safety factor related to ageing is managed under the NRC License Renewal Program under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.” Use of experience from other plants and research findings is coordinated under the agency Operating Experience Program (see the previous section for more detail).

In short, the ROP is a risk-informed tool that uses both direct inspections and objective performance indicators to gauge plant performance. The NRC then assesses the results through the agency’s quarterly, mid-cycle and end-of-cycle (EOC) reviews, the EOC summary meetings and the Agency Action Review Meetings. Based on the assessments, the agency may take appropriate actions to ensure the continuous safe operation of the plants.

III. *Ongoing Generic Upgrades and Regulatory Changes (including Licensing Basis Updates and Imposition of New Requirements)*

<u>*Associated safety factors*</u>: *numerous, including safety performance, plant design, actual conditions of SSCs,* and *equipment qualification*

<u>Generic Upgrades</u>.
The NRC evaluates industry-wide issues that are safety significant and may require technical resolution. The agency issues generic communications, such as Generic Letters, Information Notices and Regulatory Issue Summaries, to alert licensees of such issues. The following are five examples of such generic communications, some of which illustrate how the NRC accomplishes generic upgrades on specific topics across the industry:

<u>Generic Letter (GL) 89-10, “Safety-Related Motor Operated Vale Test and Surveillance,” dated June 28, 1989 (Ref. 24)</u>. The purpose of this GL was to inform the license holders that laboratory testing suggested some of the primary system motor-operated valves (MOV) might be subject to previously unaccounted for mechanisms and loads. To ensure safety, the NRC asked the licensees to provide information on a series of questions (e.g., reviewing the design basis of their motor-operated valves and establishing the correct switch settings).

- <u>GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” dated July 18, 1989 (Ref. 25)</u>. The NRC recommended licensees take certain actions to ensure the continuing availability of the service water systems in light of problems such as corrosion or bio-fouling.

- <u>GL 96-06, “Assurance of Equipment Operability and Containment during Design Basis Accident Condition,” dated September 30, 1986 (Ref. 26)</u>. The purpose of this GL was to notify the licensees about safety significant issues (e.g., water hammer of the cooling water systems serving the containment air coolers during a loss-of-coolant accident) that could affect the containment integrity and equipment operability during accident conditions.

- GL 2004-02, "Potential Impacts of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors," dated September 12, 2004 (Ref. 27). In this GL, the NRC notified the licensees about the potential susceptibility of pressurized-water reactor recirculation sump screens to debris blockage during design-basis accidents requiring circulation of the emergency core cooling system or the containment spray system.

- IN 2010-09, "Importance of Understanding Circuit Breaker Control Power Indications," (Ref. 28). The purpose of this IN was to inform the licensees about circuit breaker control power indication issues that could result in degraded circuit breaker protection and control. In this particular case, the incident (an automatic rector trip) occurred on March 28, 2010; and the NRC issued the IN on April 14, 2010. This IN exemplifies how the NRC emphasizes timely communications to the licensees on matters related to the safe operation of their facilities.

Regulatory Changes

The NRC recognizes the need to consider new requirements for plant upgrades during the life of a plant. As new technical information develops, the NRC reviews the potential safety concerns and may conclude that existing programs or regulations need to be revised to assure an acceptable level of safety. Such changes, called backfits, are subject to the requirements of 10 CFR 50.109, "Backfitting." Backfitting is defined as the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position. The backfit rule requires implementation of a backfit when the backfit is necessary to bring a facility into compliance with a license or the rules and/or orders of the Commission, or to ensure adequate protection to the health and safety of public, and is in accord with the common defense and security. The backfit rule also requires implementation of backfits that involve defining or redefining what level of protection to the public and safety or common defense and security is regarded as adequate. Otherwise a backfit analysis must be prepared. A backfit may be imposed if the backfit analysis shows (1) a substantial increase in the overall protection of the public health and safety, or the common defense and security is to be derived from implementing the proposed backfit, and (2) the direct and indirect costs of implementing the backfit are justified in view of the increase protection (Ref. 29). An example of a cost-justified regulatory change is the recently amended 10 CFR Part 73 Power Reactor Security Rulemaking[4]. The Power Reactor Security Rulemaking was a comprehensive rulemaking that amended numerous provisions within 10 CFR Part 73 (where the NRC's security requirements reside) and added new requirements to the NRC's regulations. The majority of the requirements stemmed from the post-September 11, 2001 orders. Since these requirements were already in place and implemented at operating U.S. nuclear power plants, they were not considered as

---

4 The NRC had already undertaken an effort to revise the existing security regulations in 10 CFR Part 73 prior to September 11, 2001 because the security regulations had not been substantially revised for nearly 30 years. The rulemaking effort was delayed as a result of the terrorist attacks on U.S. installations on September 11, 2001. However, the need to improve and update the existing security regulations persists. The rulemaking effort resumed in mid 2000's and the final rule became effective on May 2, 2009.

"backfits" under the NRC's backfit provisions. However the Power Reactor Security Rulemaking did add a significant number of new security provisions which were addressed under the NRC's backfit provisions and were imposed on current licensees as cost-justified, substantial safety enhancements.

A second example of an agency regulatory action involving backfit is the incorporation of new fire protection requirements for nuclear power plants following the 1975 Browns Ferry event (Ref. 30), which highlighted the possibility of common-mode failure. The NRC review of this event found that new regulatory requirements were necessary to reduce the likelihood of fires causing damage to reactor safety systems needed to safely shutdown the plant. To ensure public health and safety, the NRC issued new requirements in 1981: 10 CFR 50.48, "Fire Protection," and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The new requirements significantly enhanced fire protection at U.S. nuclear power plants.

A third example of a rule change is the Anticipated Transient without Scram (ATWS) Rule (10 CFR 50.62, "Requirements of Reduction of Risk from Anticipated Transits without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants"). The ATWS Rule was prompted by two separate incidents in 1983 at Salem Unit 1, where the reactor failed to scram automatically due to failure of both reactor trip breakers to open upon receipt of an actuation signal (Ref. 31). These ATWS events raised significant safety concerns, which led the NRC to promulgate the ATWS Rule. For pressurized water reactors (PWR), the rule requires a diverse scram system to interrupt power to the control rods and a diverse system to initiate the auxiliary feed water system under conditions indicative of an ATWS. For boiling water reactors, the rule requires an alternate rod injection system, a standby liquid control system that will automatically inject, and a system that trips the recirculation pumps under conditions indicative of an ATWS. Other examples of significant rule changes that were imposed on the industry through backfit include the Station Blackout Rule (10 CFR 50.63, "Loss of All Alternating Current Power"), and the Maintenance Rule (10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants").

In addition to generic backfit imposed by rule, the NRC can also impose new requirements by order if the new requirements are necessary to bring a facility into compliance with a license or the rules and/or orders of the Commission, or to ensure adequate protection to the health and safety of public and are in accord with the common defense and security. The Commission may also impose by order new requirements that involve defining and redefining what level of protection to the public health and safety or common defense and security is adequate. An example of the Commission exercising its authority to impose new requirements necessary for adequate protection are the orders issued by the Commission following the terrorist attacks on U.S. infrastructure on September 11, 2001, which required security upgrades at all plants. The NRC can also implement new requirements during the license renewal review in order to effectively manage ageing. Such new requirements are not considered backfits.

Finally, the NRC routinely performs technical evaluations for amendments to plant operating licenses under 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit." These amendments often involve changes to the plant to make designs safer or more reliable. Further, under 10 CFR 50.71(e), the plant's licensing basis must be updated on a biennial basis to ensure that the final safety analysis report contains the latest information. Also, under 10 CFR 50.55(a), "Codes and Standards," the licensees regularly update their licenses to apply newer versions of the codes and standards endorsed by the agency.

IV. *Incorporation of Risk Information into the Regulatory Activities*

> Associated safety factors: *several, including plant design, actual condition of systems, structures and components, probabilistic safety assessment, hazard analysis, safety performance, radiological impact on the environment*

The NRC has embraced the concept of risk since the agency's inception. In 1975, the agency published the WASH-1400 (NUREG-75/014), "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (Ref. 32)[5]. The study used PSA techniques to estimate the probabilities and consequences of severe reactor accidents (core melt) at two commercial nuclear power plants. The study used the concept of event trees to link the system fault trees to the accident initiators and core damage states. The Reactor Safety Study highlighted that human error is a major contributor to severe accidents (Ref. 33).

In GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54 (f)," dated November 23, 1988 (Ref. 34), and associated supplements issued through 1991, the NRC required all nuclear power plant licensees to use a systematic approach to perform an individual plant examination and individual plant examination of external events (commonly referred to as IPE and IPEEE, respectively) to identify severe accident vulnerabilities. In response, licensees performed plant-specific PSAs for internal events, and used various methods (PSA, margins, and screening approaches) to assess plant vulnerabilities to external events. The NRC reviewed all licensee responses to the generic letter to determine the adequacy of the licensees' ability to identify severe accident vulnerabilities at their plants through risk assessment. As a result of the GL and the development of policies and guidance encouraging the use of PSA in regulatory activities, all U.S. nuclear power plants have a plant-specific PSA that, at a minimum, addresses internal events occurring at full power.

An example of NRC using risk insights is its Fire Protection Rule (10 CFR 50.48), which allows licensees to voluntarily adopt and use the fire protection requirements of National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Electric Generating Plant" (Ref. 35). The NRC cooperatively participated in the development of this standard with industry and other. NFPA 805 describes a methodology for existing light-water nuclear power plants to apply performance-based requirements and fundamental fire protection design elements to establish fire protection systems and features for all modes of operation. Historically, deterministic fire protection requirements aimed to establish fire protection engineering margin through the post-fire survival of limited safety systems capable of safe reactor shutdown. The deterministic requirements were developed before the benefits of PSA for fires and before the advent of performance-based methods. Having a voluntary alternative to the Fire Protection Rule is expected to reduce the need for exemptions, and thus the regulatory burden associated with the deterministic approaches, while ensuring plants' ability to maintain safety and providing appropriate flexibility to licensees' fire protection activities.

---

[5] The Reactor Safety Study is commonly known as the Rasmussen Report, named after Professor Norman Rasmussen of Massachusetts Institute of Technology. It was initiated under the auspice of the Atomic Energy Commission, the NRC's predecessor.

A second example of the NRC's use of risk insights is the Pressurized Thermal Shock (PTS) Rule (10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events"), which was recently revised using risk insights to provide alternative requirements for protection against PTS events. The rule increases the realism of calculations used to examine a PWR's susceptibility to PTS. PTS can occur under some scenarios that rapidly cool the internal surface of the reactor vessel while the vessel is pressurized. On the basis of current understanding of material behavior and enhanced ability to realistically model the plant systems and operating characteristics, the NRC concluded that the 20-year-old analyses that supported the development of the PTS Rule were overly conservative (Ref. 36). Therefore, the NRC developed a risk-informed revision of the PTS Rule. The revised rule allows PWR licensees to voluntarily adopt a more realistic technical approach for determining the probability of vessel failure during a PTS event. This revised approach, based on the integrated results from the thermal hydraulic analyses, fracture mechanics analyses, and PSA, was derived using data from research on currently operating PWRs. Analyses based on the revised approach indicate that the U.S. PWRs do not approach the level of embrittlement to make them susceptible to PTS failure until well beyond the originally expected 40-year plant life.

Strengths of Current Policy/Program

The NRC's current policies and program have five main strengths that make them comparable to the PSR process. First, the NRC's regulatory process emphasizes ongoing technical evaluation and oversight of the plant operations. Because the design basis evolves during the entire license period, a continuing oversight process ensures facility safety throughout the life of the plant. Annually, the NRC devotes significant resources to the oversight process at each plant. Through the use of resident inspectors, who provide daily inspections, and regional specialists, each plant receives 6,000 to 10,000 hours of inspection. Focused in-depth inspection teams are routinely scheduled to evaluate the safety of licensees' designs and operations. For example, the NRC spent approximately 20,000 staff hours conducting component design basis inspections (CDBI) at 24 facilities in 2009. The purpose of the CDBI is to verify the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. Additionally, over 1,200 hours are spent evaluating licensing tasks at each plant. This level of effort gives NRC the confidence that its oversight process produces a level of safety comparable to that afforded by the PSR process.

Second, the NRC's regulatory oversight program is comprehensive. It encompasses a wide spectrum of programmatic activities - ranging from the initial licensing and inspection to cross-cutting safety culture issues - that incorporates all of the safety factors evaluated in the PSR process. To assess whether there were any significant gaps between the PSR process and the U.S. regulatory oversight process, NRC staff from several divisions of the Office of Nuclear Reactor Regulation and other program offices studied the IAEA PSR safety factors to verify that the U.S. program elements accomplish the associated functions. Table 1 presents a sample cross-walk between selected PSR safety factors and the corresponding U.S. program elements. Attachment 1 includes a more complete comparison between the PSR's 14 safety factors and global assessment to the comparable U.S. program elements. As indicated in Attachment 1, there are no significant gaps between the IAEA PSR safety factors and the U.S. program. The final section of this paper includes a recommendation to better inform the U.S. program with the IAEA PSR process.

Table 1. A Brief Crosswalk of Selected PSR Safety Factors to U.S. Program Elements

IAEA NS-G-2.10 Safety Factor	U. S. Program Elements
Actual Condition of SSCs	1. In-depth daily inspections by resident inspectors 2. Focused routine inspections by specialist (e.g., Maintenance Rule, Corrective Action Program) 3. ROP Performance Indicators
Equipment Qualification	1. Inspections tied to Environmental Qualification Rule (10 CFR 50.49) • Component Design-Basis Inspections • Permanent plant modifications 2. License Event Reports
Ageing	1. Ageing Management Programs (10 CFR 54) 2. License Renewal inspections (passive components) 3. Maintenance Rule and other ROP inspections (active components)
Deterministic Safety Analysis	1. Evaluation of changes to design and licensing basis 2. Changes to the Final Safety Analysis Report (10 CFR 50.59) 3. Daily inspections that compare everyday operation to design bases
Probabilistic Safety Assessment	1. PSAs used in selecting inspection samples 2. Plant-specific PSAs for internal and some external events 3. PSAs can be used in lieu of deterministic assessments (RG 1.174) 4. Maintenance Rule (10 CFR 50.65) 5. PSAs not required to be updated in the U.S.
Note: See Attachment 1 for a more in-depth comparison of PSR safety factors to U.S. program elements.	

Third, the NRC reviewed several international PSR-related documents to confirm that the outcomes from performing PSRs and conducting the NRC regulatory programs are similar. Because the actual PSRs submitted to the regulatory authorities were not readily available to the NRC for review, the agency reviewed a number of international regulators' PSR evaluations and a PSR summary report submitted by an international plant. A high-level comparison suggested all findings and other recommendations reviewed are in areas that have received similar regulatory attention through the NRC's ongoing regulatory process. Attachment 2 synopsizes the issues identified in the review of the PSR reports. Although this high-level review is not an exhaustive comparison of the U.S. safety program vis-à-vis the PSR process, it adds assurance that the U.S. approach affords a comparable level of safety to the PSR process currently used by many international regulatory authorities.

Fourth, the commercial nuclear utilities regularly assess the safety performance of their nuclear power plants. Following the 1979 Three Mile Island incident, the U.S. nuclear power industry formed the Institute of Nuclear Power Operations (INPO) to promote safety and reliable operation of nuclear power plants. INPO conducts biennial independent assessments at all member stations, using a multidisciplinary team of INPO employees and independent industry peers with supervisory or strong technical expertise in the areas they are assessing. INPO assesses the plants in the following areas: operations, maintenance, work management, configuration management, design engineering, equipment reliability, radiological protection, chemistry, training, organizational effectiveness and safety culture. During the assessments, the evaluation team observes operations, analyzes processes, and shadows personnel. The assessments are preceded by a 3-week preparation process in which the team members review critical data from plant operation (e.g., corrective action information, plant performance data, and self-assessments) collected since the last assessment. The evaluation team uses detailed performance objectives and criteria for each area being assessed. The team briefs the plant senior management on the output of the assessment, which consists of area performance summaries and areas for improvement. NRC staff routinely review these reports as an independent check to ensure that NRC processes are capturing similar performance insights.

Finally, owners groups and equipment vendors have long played the role of providing unified industry approaches to generic nuclear regulatory and technical issues and coordinating interactions with the NRC. The Boiling Water Reactor Owners Group and the Pressurized Water Reactor Owners Group were formed to share industry operating experience. At the request of the owners groups, the NRC may comment and/or review the owners groups' topic reports. The NRC regularly meets with these owners groups to stay abreast of the existing and emerging plant safety issues of mutual interest.

Conclusion and Considerations for the Future

In summary, the objectives of the PSR process are well served by the current U.S. regulatory process. The NRC would agree with three main goals summarized in a recent PSR report:

1. Confirm that the plant is as safe as originally intended.
2. Determine if there are any SSCs that could limit the life of the plant in the foreseeable future.
3. Compare the plant against modern safety standards and identify where improvements would be beneficial at justifiable cost.

As discussed in this paper, the current U.S. regulatory process ensures these goals are met. The U.S. regulatory structure ensures through its licensing process that plants are thoroughly and comprehensively reviewed prior to allowing operation to begin. Further, through an extensive inspection regime, the United Sates monitors plant performance and operating experience on a daily basis. Results from these intrusive, independent inspections are coupled with objective performance indicator data, and then assessed in a comprehensive manner by regional and senior agency management at a mid-year and end-of-year assessment to verify that plants continue to operate safely.

The NRC manages ageing through insights gained from inspections of active components (e.g., maintenance and corrective action inspections, follow-up generic action), and passive components through the formal license renewal process which establishes comprehensive ageing programs for passive long-lived components (e.g., reactor vessels, cabling, buried piping). These processes are informed by a robust operating experience program that screens both domestic and international experience for insights that can be used to improve plant performance.

The U.S. system also requires that plants upgrade to more modern safety standards on an on-going basis through new regulations and orders that impose new requirements. Similar to what is accomplished through the PSR process, the NRC evaluates these changes for safety benefit before requiring implementation. Although the U.S. system does not require its licensees to summarize performance with, for example, a recurring 10-year submittal to the regulator, the U.S. believes that that its day-to-day focus on inspection and assessment ensures that these improvements are evaluated year to year. As part of the preparation for the IRRS Mission visit, the NRC reviewed several international PSR evaluations. Issues identified and documented in these 10-year reviews appear to be very similar to those identified, documented, and evaluated annually in the inspection, licensing, and generic actions under the U.S. process.

The NRC regards the process of standing back and performing a holistic in-depth evaluation of each plant at a regular interval to be beneficial. The NRC follows this practice on a shorter interval, evaluating operating experience, considering upgrades, and performing assessments annually. The NRC then uses its formal License Renewal process to evaluate extending the license expiration date.

On the basis of the most recent comparison of the PSR process to the U.S. program elements, this paper provides one recommendation for consideration. This recommendation relates to particular aspects of the PSR process where further information would be beneficial:

> As a part of the agency's operating experience program and Generic Issues program, the NRC could more systematically review findings from other regulators' assessments of PSRs to continue to verify that international "experience" is fully evaluated for potential applicability to U.S. licensees.

Attachments

Attachment 1 Comparison of PSR Safety Factors to the NRC Program Elements

Attachment 2 Summary of Issues Discussed in the PSR Evaluation and PSR Summary Reports Reviewed by NRC Staff

Attachment 1 - Comparison of PSR Safety Factors to NRC Program Elements

IAEA NS-G-2.10 Safety Factor Objective	Summary of U.S. Activities that Accomplish the Safety Factor Objectives	Potential Gap
(1) Plant Design (IAEA Safety Guide 4.10 – 4.13) – The objective of the review is to determine • the adequacy of the design and its documentation against current international standards and practice.	• UFSAR Periodic Updates and NRC Oversight • Focused Inspections, e.g., the Component Design Bases Inspection (CDBI) – o verifies the initial design and subsequent modifications o monitors the capability of selected components and operator actions to perform their design bases functions o performed at each facility on a triennial basis o 20,000 inspection hours expended at 24 facilities in 2009 • Require licensees to maintain programs (e.g., 10 CFR Appendix B Quality Assurance and 10 CFR Part 21 Reporting Defects and Noncompliance)	The United States relies on its comprehensive regulations and interactions with various international regulators to stay abreast of international standards. Although the United States historically has not reviewed international standards and practices, there have been cases in recent years where NRC actively collaborated with international community to develop new NUREG guidance and standard documents. Example: (1) HRA Best Practices (Ref: NUREG-1792, “Good Practices for Implementing Human Reliability Analysis,” April 2005.); (2) ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” 2009.
(2) Actual Condition of Structures, Systems, and Components (SSCs) (IAEA Safety Guide 4.14 – 4.16) – The objective of the review is to determine • the actual condition of SSCs Important to Safety (ITS) and whether it is adequate to meet their design requirements • Confirm the condition of SSCs is properly documented	• Daily oversight provided by the resident inspectors at each facility • Regional Inspections (e.g., Maintenance Rule) • Reporting Requirements and Operating Experience Review • The average U.S. plant receives approximately 6,000 hours of direct inspection each year.	None identified.

(3) Equipment Qualification (IAEA Safety Guide 4.17 – 4.20) – The objective of the review is to determine • whether equipment ITS is qualified to perform its designated safety function throughout its installed service life	CDBI – • Inspects components for degradation • Verifies component replacement is consistent with in-Service and equipment qualification life, • Ensures the number of cycles is tracked for cycle-sensitive components, • Confirms equipment qualification is suitable for the environment expected under all conditions.	None identified.
(4) Ageing (IAEA Safety Guide 4.21 – 4.25) – The objective of the review is to determine • whether ageing in an NPP is being effectively managed so the required safety function are maintained; • Whether an effective ageing management program is in place for future operation	• The NRC license renewal process safety review focuses on the effects of ageing on SSCs important to safety. Applicant must demonstrate that they have identified and can manage the effects of ageing, and are able to maintain an acceptable level of safety throughout the period of extended operation. • NRC Inspection, Operating Experience, and Generic Correspondence Reviews	None identified.
(5) Deterministic Safety Analysis (DSA) (IAEA Safety Guide 4.26 – 4.28) – The objective of the review is to determine • to what extent the existing DSA remains valid when the following aspects are considered: actual plant design; the actual condition of SSCs, and their predicted state at the end of period covered by the PSR; current deterministic methods; and current safety standards and knowledge • The review should identify any weakness relating to the application of the defense-in-depth concept.	• For the initial siting of current fleet of NPPs licensed under 10 CFR Part 50, the selection, assessment, and evaluation of deterministic safety analyses is governed under 10 CFR Part 100, "Reactor Site Criteria." • The design basis accident (DBA) analyses, summarized in the Final Safety Analysis Report, include dose consequence to the public conservatively calculated based on deterministic safety analyses. • All the deterministic DBA analysis models are maintained in the Updated Final Safety Analysis Report per 10 CFR 50.34. The limiting conditions for operation of the NPP defined in the analyses are maintained in the plant Technical Specifications per 10 CFR 50.36, and are subject to NRC inspections. • Routine licensing reviews and inspections ensure the analyses are up-to-date.	None identified.

(6) Probabilistic Safety Assessment (PSA) – (IAEA Safety Guide 4.29 – 4.32) – The objective of the review is to determine • to what extent the existing PSA remains a representative model of the plant when the following aspects are considered: changes in the design and operation of the plant; new technical information; current methods; and new operational data.	• GL 88-20[6] and supplements required all NPP licensees to use a systematic approach to identify severe accident vulnerabilities. In response, licensees performed plant-specific PSAs for internal events, and various methods (PSA, margins, and screening approaches) to assess plant vulnerabilities to external events. • As a result of the generic letter and development of policies and guidance encouraging the use of PSA in regulatory activities, all U.S. NPPs have a plant-specific PSA that at a minimum addresses internal events occurring at full power. • As stated in RIS 2007-06[7], PSAs supporting risk-informed applications submitted after December 2007 are expected to meet the requirements specified in RG 1.200[8].	PSAs are not specifically required for the current operating plants. However, the PSAs are regularly used as an input to the agency's evaluation of license amendment requests which utilize risk insights. In practice, this is not considered as a significant gap.
(7) Hazard Analysis (IAEA Safety Guide 4.33 – 4.35) – The objective of the review is to determine • the adequacy of protection of the NPP against internal and external hazards w/ account taken of the actual plant design, actual site characteristics, the actual condition of SSCs and their predicted at the end of period covered by the PSR, and current analytical methods, safety standards and knowledge.	• External hazards are assessed as part of the design basis accident evaluation performed during initial plant licensing. • The risk and potential vulnerabilities associated w/ internal and external hazards were assessed on a one-time basis as a part of the IPEEE, and reviewed and accepted by NRC staff. • Potential improvements to reduce the risk from internal and external events (within the scope of IPEEE) are considered within the Severe Accident Mitigation Alternative analysis performed for license renewal. • NRC resident inspectors regularly perform walk-downs to check for internal and external	None identified.

6 NRC Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," 1988.

7 NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation."

8 NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." This RG endorses the industry peer review process used in the licensee's evaluation of the adequacy of their plant-specific PSA. Through this RG, NRC endorsed PSA consensus standards that can be used to determine the technical adequacy and capability of a plant-specific PSA when it's for a specific application.

	hazards. Inspections for adverse weather, flooding (internal and external), and fire protection and plant outages are conducted on a quarterly and annual basis. • To take advantage of certain risk-informed regulations, licensees will upgrade their plant based on risk insights. For example, to adopt the new fire protection standards (NFPA 805), the Harris Plant is adding make-up pumps and an emergency diesel generator.	
(8) Safety Performance (IAEA Safety Guide 4.36 – 4.38) – The objective of the review is to determine • the safety performance of the NPP and its trends from records of operating experience.	• The NRC continuously assesses industry safety performance and licensee's ability to operate within the requirements set forth in the regulations and the technical specifications. The assessment is implemented through its Reactor Oversight Program (ROP). • The ROP Performance Indicator (PI) Program uses a cornerstone approach that covers all areas of licensee performance. PIs are reported to the NRC by the licensees on a quarterly basis. The NRC periodically reviews PI data to verify its accuracy and completeness. • The PI Program supplements the ROP Inspection Program. The inspection content and sample size are determined based on plant-specific performance issues and industry-wide operating experience. • Supplemental inspections are conducted to verify the adequacy of the licensee's corrective actions taken in response to inspection findings. In addition, reactive inspections are conducted in response to an event or degraded conditions.	None identified.
(9) Use of Experience from Other Plants and Research Findings (IAEA Safety Guide 4.39 – 4.40) – The objective of the review is to determine	• As outlined in GL 82-04, compiling and analyzing operating experience within the industry is joint responsibility between INPO and the individual licensees.	None identified.

• whether there is adequate feedback of safety experience from other NPPs and of the findings of research.	• The INPO Significant Event Evaluation and Information Network (SEE-IN) provides a mechanism for central collection and screening of all events from both U.S. and foreign NPPs. • The NRC Operating Experience (OpE) Program is used to evaluate issues that are safety significant and generic. The OpE information is shared with the nuclear industry in a timely manner to ensure safety. • The application of lessons learned from OpE evaluations could involve further communication with internal (e.g., resident inspectors) and external stakeholders (e.g., licensees and public). Examples include coordination and issuance of generic communications, preliminary notifications made available through the web, assessment of ratings for all power reactor events, reporting to the International Nuclear and Radiological Event Scale of significant events, and development and reporting in accordance with the IAEA Incident Reporting System. • Technical issues (e.g., pump performance, etc.) identified in the OpE programs are reviewed by the NRC Technical Review Groups (TRG). Each TRG provides a written report to the OpE staff to summarize the information received, steps taken and recommendations for future action(s), if any. • The effectiveness of licensee operating experience program, use of SEE-IN program, and application of communications from the NRC is subject to NRC inspection.	
(10) Organization and Administration (IAEA Safety Guide 4.41 – 4.42) – The objective of the review is to determine • whether the organization and administration are adequate for the safe operation of the NPP.	• The ROP integrates the NRC's inspection, assessment, and enforcement programs to evaluate the adequacy of the licensee's organization and administration on the safety of plant operation. • Cross-cutting issues (i.e., human performance, problem identification and resolution, and	The NRC reviews performance and administration, but not organization. Organizational issues leading to degrading performance will manifest themselves through the performance indicators or

	safety conscious work environment) are identified and assessed through the ROP. • Licensee performance is evaluated continuously through on-going inspections and monitoring of plant activities. • Safety Culture – In the past, weaknesses in safety culture were identified as contributing factors to several events at NRC-regulated facilities. Currently, NRC has several on-going internal and external safety culture activities. NRC held a public workshop to seek feedback on the definition of safety culture and traits in February 2010. NRC continues to reach out to all stakeholders in the nuclear power and materials areas to develop a Safety Culture Policy Statement applicable to all NRC licensees and certificate holders. A final draft policy statement is due to the Commission in March 2011.	inspection findings. No significant gap identified.
(11) Procedures (IAEA Safety Guide 4.43 – 4.44) – The objective of the review is to determine • whether the procedures are of an adequate standard.	• NRC approach to assuring procedure quality is to approve Owner's Group Generic Guidelines and Site-specific Implementation Plans. • NRC staff reviews procedure programs and processes. • On an on-going basis, the NRC resident inspectors (RIs) evaluate the use of procedures when observing work-in-progress in the field. • The NRC RIs also monitor corrective actions to determine if a licensee has placed appropriate emphasis on solving and preventing problems related to the development, maintenance, and use of procedures.	None identified.
(12) Human Factors - (IAEA Safety Guide 4.45 – 4.46) – The objective of the review is to determine	• Following the TMI accident, all applicants and licensees were required to review the human factors of their control room(s) by performing a Detailed Control Room Design Review with guidance from NUREG-0700[9] and NUREG-	None identified.

[9] NRC NUREG-0700, "Human-System Interface Design Review Guidelines," 2002.

• the status of various human factors that may affect the safe operation of the NPP.	0711[10]. License amendments are reviewed by NRC staff using the same guidance documents and NUREG-1764[11]. • 10 CFR Part 55 establishes procedures and criteria for the issuance of licenses to operators and senior reactor operators. Before the NRC licenses an individual to operate or supervise the controls of an NPP, the applicant must complete an extensive training program approved by NRC, be in sound health, and pass rigorous examinations. • 10 CFR 50.120 requires that the training programs for nine other categories of station personnel be derived from a system approach to training, and incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. • The ROP provides a mechanism for early intervention when human factor-related problems begin to exhibit a negative trend at any plant. • The NRC staff trends human performance problems and their root causes on an industry basis under the Human Factors Information System.	
(13) Emergency Planning (IAEA Safety Guide 4.47 – 4.48) – The objective of the review is to determine • whether the operating organization has adequate plans, staff, facilities and equipment for dealing with emergencies, and	• NRC routinely participates/observes licensee-conducted drills and exercises; • NRC monitors the licensee's reporting of emergency preparedness specific performance indicators. • NRC periodically inspects the licensee's emergency preparedness program.	None identified.

[10] NRC NUREG-0711, "Human Factors Engineering Program Review Model," 2004.
[11] NRC NUREG-1764, "Guidance for the Review of Changes to Human Actions," 2004.

• whether the operating organization’s arrangements have been adequately coordinated with local and national systems and are regularly exercised.		
(14) Radiological Impact on the Environment (IAEA Safety Guide 4.49 – 4.50) – The objective of the review is to determine • whether the operating organization has an adequate program for surveillance of the radiological impact of the plant on the environment.	• The environment surveillance requirements are contained in each plant’s operating license. They include monitoring for direct radiation, airborne radioactivity, waterborne radioactivity, and radioactivity in food. Region-based specialist inspectors perform routine inspections of radiological impacts. • The licensees are required to analyze the monitoring results (including comparison to pre-operational environmental surveillance data) and report the analyses with previous trending data in its Annual Operating Environmental Monitoring Report. • NRC would be notified by the licensee of a deviation if the level of radioactivity in the environment exceeds the reporting thresholds. The licensee reports are publicly available on the NRC website.	None identified.
Global Assessment (IAEA Safety Guide 4.51 – 4.52) – The objective of the global assessment is to present • an assessment of plant safety that takes into account all unresolved shortcomings, all corrective actions and/or safety improvements and the plant strengths identified in the review of all PSR safety factors. • A global assessment report should be prepared that presents significant PSR results (including plant strengths), the integrated	• The NRC uses the ROP to inspect, measure and assess the safety performance of commercial NPPs and respond to any decline in performance. The ROP uses performance indicators (PIs) to ensure compliance with NRC regulations and specific license conditions. Plant design basis, licensing basis, commitments made to NRC, and operating experience are also considered under the ROP to focus inspection effort. • The ROP uses a Significance Determination Process (SDP) to determine the safety significance of most inspection findings identified at commercial NPPs. A licensee is provided the initial NRC assessment of risk associated with an inspection finding, and is	Although a “global assessment, including plant strengths and a global risk judgment” is not written on a global basis (e.g., every 10 years), the U.S. system assesses plant performance annually and regularly makes decisions on continued plant operation. Additionally, license renewal reviews often summarize plant upgrades that have occurred. No significant gap identified.

	implementation plan for corrective actions, and/or safety improvements, and a global "risk judgment" on the acceptability of continued plant operation with any shortcomings remaining after all corrective actions and/or safety improvements have been implemented. The global assessment should show to what extent the safety requirements of the defense-in-depth concept are fulfilled, in particular for the basic safety functions of reactivity control, fuel cooling and the confinement of radioactive material.	asked top provide additional info that might not be available when the initial inspection was conducted. After consideration of the new information, the NRC arrives at a final determination of significance. • On a periodic basis, the NRC conducts a review of each licensee's corrective action program. In addition, the NRC performs periodic inspections on a licensee's ability to address problems. Unresolved items from previous inspections are being tracked for proper disposition. • To provide early warning of potential issues and determine whether the licensees are complying with NRC regulations re: corrective action programs, the NRC reviews and inspects the licensee's problem identification and resolution (PI&R) program. The inspections focus on the identification of problems and effectiveness of corrective actions. • The NRC assessment starts with a continuous review, a quarterly review and a formal mid-year (mid cycle) and a year-end (end-of-cycle) review of licensee performance that are conducted by the NRC regional offices in consultation with the HQ program office. • Both the mid-cycle and the end-of-cycle review provide management an opportunity to review and allocate regional inspection resources. These reviews also consider the conclusions of any independent assessments of licensee performance such as INPO, and IAEA Operational Safety Review Team inspections. Following the reviews, the NRC will either send a mid-cycle assessment letter or an annual assessment letter to the licensee. The letter documents the NRC current assessment of the licensee's performance. A public meeting or event is scheduled following issuance of the	

	annual assessment letter to discuss the results of the NRC annual assessment of the licensee's performance. • Deviations to the nominal ROP inspections are authorized by NRC management for additional inspection or assessment resources, as necessary. • Plant safety assessments are also routinely performed for plant license amendments and generic upgrades.	

Attachment 2 – Summary of Issues Discussed in the PSR Evaluation and PSR Summary Reports Reviewed by NRC Staff

As a part of the PSR module paper preparation, the NRC staff performed a high-level review of four PSR evaluation reports prepared by the international regulators and one PSR summary report prepared by a utility company. Three of the five reports were received in English, while the other two were received in the respective native language. As such, portions of one report were formally translated into English by a contract translation agency, and the second one was informally translated into English at NRC.

The high-level review suggests all findings and recommendations prepared by the regulators and the utility company is in areas that have received similar regulatory attention through the NRC's ongoing regulatory process. The specific issues covered in these reports are as follows:

1. Preparation of ageing management program
2. Failed fuel rods (optimization program; monitor and inspection program)
3. External events and earthquake modeling updates
4. Problems with MOVs
5. Update locations for fatigue monitoring
6. Steam dryer vibrations
7. Evaluate reactor vessel welds
8. Update the brittle failure RTT using Master Curve
9. Update to using environmental fatigue curves
10. Inspection methods for coatings
11. Update PSA
12. Update radiation measuring/monitoring techniques and documentation
13. Effects of new source term specification ANS-18.1-1999
14. Radiological effects of airplane crash
15. Integrate environmental effects in actual description of facilities
16. Major design and operating changes
17. Explosion risks due to internal causes
18. Corrosion and vibration fatigue in hydrogen ducts
19. Analytical approach for explosion risks
20. Cold overpressure risk
21. Instrument to detect any puncture of the reactor vessel
22. Install passive hydrogen autocatalytic recombiners
23. Reduce core meltdown risk
24. Post accident recirculation to limit radiological releases
25. Reduce potential for boron-dilution risk in primary circuit
26. Sulphate attack on containment and other civil-engineering structures
27. Assess behavior of composite material used for building stacks
28. Creep of reactor vessels
29. Review safety, design, manufacturing, in-service monitoring of steam generators
30. Potential for under vessel penetration leaks
31. Plant modifications to ensure quick detection of floods (PRA flood risks)

32. PRA fire risk – updating analysis and upgrading sprinkler system
33. PRA external event risk – oil spillage accidents and line freezing
34. Availability of offsite power – plan for diversification
35. Risk associated with shutdown
36. Replacement of reheaters, 6.6kV switchgear, I&C equipment of the turbine plant and reactor steam dryers
37. Risk-informing non-destructive examination program
38. Compiling a database of pressure and transient monitoring for updating fatigue analyses
39. Replacement of steam dryers, feedwater manifolds and other major equipment
40. Replacement of inner isolation valves in main steam line (based on increased flow rate of steam from power uprate)
41. Replacement of low pressure turbines and main seawater pumps
42. Considering replacement of emergency diesel generators
43. Erosion corrosion in turbine piping, extraction steam piping, etc.
44. Modifications planned: replacement of low-voltage switchgear, improvement of over-voltage protection of electrical drives of RCPs, replacement of cabling inside containment
45. Supply of spare parts
46. Planning to replace analog with digital; during interim, submitting a plan for diversification of programmable equipment
47. Repaired leak in condensate pool lining; replaced expansion joint gasket in containment partition floor; cathode protection of seawater structures
48. Upgrading mechanical supports of safety-classified electrical cabinets and battery banks to improve earthquake resistance
49. Demonstration of electromagnetic compatibility for electrical and I&C components
50. Develop a management of service life plan with regard to electrical and I&C equipment qualified for demanding conditions
51. Progress report on assessment of safety culture