ANS PSA 2008 Topical Meeting - Challenges to PSA during the nuclear renaissance
Knoxville, Tennessee, September 7–11, 2008, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2008)

COMPARISON OF NEW LIGHT-WATER REACTOR RISK PROFILES

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ABSTRACT

This paper compares the risk profiles of the four light-water reactor (LWR) standard designs that have been certified by the U.S. Nuclear Regulatory Commission (NRC) and three other designs that are in various stages of review by the NRC staff. To the extent possible, the author also compares new reactor risk profiles to those of currently operating plants. The author demonstrates that the core damage frequencies (CDF) from internally initiated events for new reactor designs are about one to three orders of magnitude lower than the mean of operating reactors based on comparison to the Individual Plant Examination (IPE) results compiled in NUREG-1560 and more recent results tabulated by the NRC staff from Mitigating Systems Performance Index (MSPI) basis documents. It is also shown that the frequencies of large radiological release from new reactors are approximately one to four orders of magnitude lower than the mean of operating plants. For new reactors, the author observes a clear distinction in risk profiles between the passive system designs (e.g., AP1000 and ESBWR), and those employing more conventional active mitigation systems (e.g., ABWR, U.S. EPR, US-APWR). While both types of new plants generally have lower risks than operating plants in *absolute* terms (i.e., per reactor-year), on a *relative* basis (e.g., percentage contribution to CDF by initiator type, and the importance measures for key equipment and human actions) those with active safety systems tend to resemble operating plants.

Key Words: new reactors, risk profiles

1 INTRODUCTION

The Reactor Safety Study [1] was completed in 1975. It was the first comprehensive study of commercial nuclear power plant risks using probabilistic risk assessment (PRA) techniques. In 1990, the NRC published NUREG-1150 [2], which built upon over a decade of advancements in PRA methods and in the understanding of severe accident phenomenology. NUREG-1150 included quantitative estimates of risk uncertainty and identified plant-specific risk vulnerabilities for the five plants studied.

In Generic Letter 88-20 [3], the NRC staff requested information on the assessment of severe accident vulnerabilities by each licensed nuclear power plant. The staff summarized the insights of these *Individual Plant Examinations* (IPE) for over 100 nuclear power plant units in NUREG-1560 [4], and for external events (IPEEE) in NUREG-1742 [5]. These two NUREGs provided a comprehensive snapshot of the risk profiles for operating power plants from the 1990s.

The Commission issued a series of policy statements through the mid-1980s and into the mid-1990s regarding safety goals for operating reactors and expectations for advanced reactor safety margins [6-8]. For example, in the August 1985 policy statement the Commission stated that it "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." Similarly, the July 1994 policy statement emphasizes that "The Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety function."

A series of Commission papers and Staff Requirements Memoranda (SRMs) established quantitative goals for core damage frequency (CDF), large release frequency (LRF), and conditional containment failure probability (CCFP) for advanced LWRs, and also proposed the resolution of a number of issues important to safety for these new reactor designs [9,10]. The safety goals and subsidiary objectives for new LWRs are:

- CDF < 10^{-4} /yr
- LRF < 10^{-6} /yr
- CCFP less than approximately 0.1

The more significant severe accident prevention and mitigation design features to be addressed include:

- Measures to minimize the risk from anticipated transient without scram (ATWS)
- Measures to minimize the risk from mid-loop operation
- Measures to minimize risk from station blackout (SBO) events
- Additional protection against interfacing systems loss-of-coolant accident (ISLOCA)
- Measures to minimize fire as a significant contributor to the likelihood of severe accidents
- Measures to control hydrogen in containment
- Measures regarding ex-vessel core-concrete interaction and core debris coolability
- Measures to limit the probability and consequences of high pressure melt ejection leading to direct containment heating
- Containment venting capability (ABWR design)
- Equipment survivability during severe accidents
- A deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges.

2 NEW LWR RISK PROFILES

2.1 Definition of Risk Profiles

Reactor risk profiles consist of numerous measures of risk including:

- CDF for internally and externally initiated events during various modes of operation
- Percentage contributions to CDF by initiating event and accident sequence type
- Top cut sets for core damage
- Contributions to LRF by various hazards and modes of operation
- CCFP
- Highest importance measures (e.g., Fussell-Vesely and risk achievement worth) by equipment and significant human actions
- Qualitative risk insights on plant design features.

2.2 New LWR Designs

Four LWR standard designs have been certified by the NRC and another three designs are in various stages of review by the staff. PRA results for certified designs can be found in several documents including the final safety evaluation report (FSER), while PRA summaries for new design applications are found on the NRC public web site. The major characteristics of each design can be found in Table I. While some basic data such as CDF will be compared for all seven designs, the paper focuses mainly on the five designs with actual or expected Combined License applications (COLA). Moreover, the objective of this paper is to compare at a high-level the risk characteristics of the new designs, citing examples of how a particular design feature affects overall safety. The reader is encouraged to refer to the cited references for detailed PRA information on a given design.

2.3 Quantitative Comparisons

The CDFs for the seven advanced LWR designs are shown in Figure 1. These values include only the conventional *internal events* contribution (no internal flooding or fire). The designs with passive safety features appear to have somewhat lower CDFs than those with active safety systems, although there is some overlap. Also provided are relatively recent industry values for operating plants tabulated from the Mitigating Systems Performance Index (MSPI) basis documents of April 2006. The average CDF for BWRs is $8x10^{-6}$ /yr, while for PWRs it is $2x10^{-5}$ /yr. (These values from the MSPI are about a factor of 3 lower than the IPE results [4], reflecting enhancements in design and operation as well as improvements in PRA modeling.) Overall, the internal events CDFs from the advanced LWRs are about one to three orders of magnitude lower than the mean of operating reactors.

Figure 2 provides the fire PRA CDFs for those advanced LWRs with a realistic assessment. (Some new designs have only a bounding type analysis and these results are not included.) For reference purposes only, the results from the IPEEE [5] are also shown. Caution must be used regarding the IPEEE fire results since the fire PRA methodology has undergone significant

change in recent years. Nevertheless, it is fair to conclude that fire risks for new LWRs are significantly lower than for operating plants owing to greater degree of redundancy and physical separation of safety systems.

Table I. Characteristics of new LWR designs

Design	Nominal Power (net MWe)	Safety Systems	Actual or Expected COLA?	PRA Ref.
Certified designs				
ABWR (GE-Hitachi)	1350	active	x	11,12
System 80+ (Westinghouse)	1300	active		13
AP600 (Westinghouse)	610	passive		14
AP1000 (Westinghouse)	1117	passive	x	15,16
Designs under review by NRC				
ESBWR (GE-Hitachi)	1535	passive	x	17
U.S. EPR (AREVA NP)	1600	active	x	18
US-APWR (Mitsubishi)	1600	active	x	19

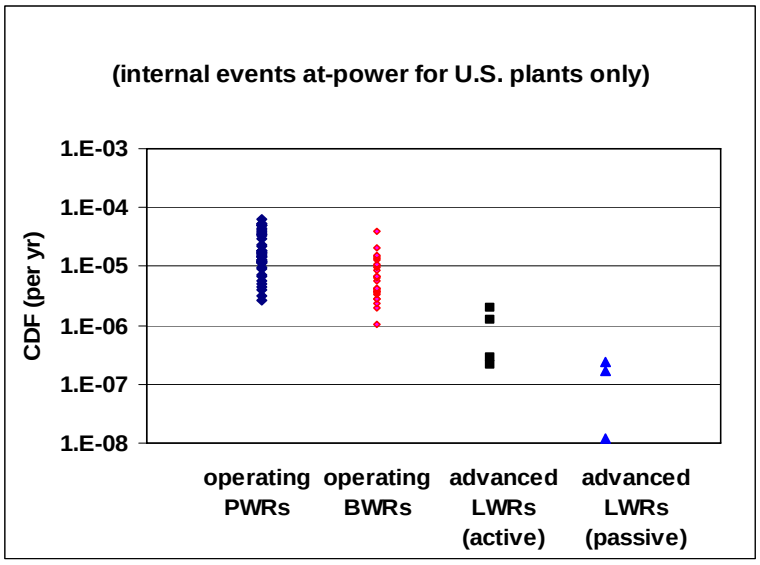

Figure 1. CDF by plant type

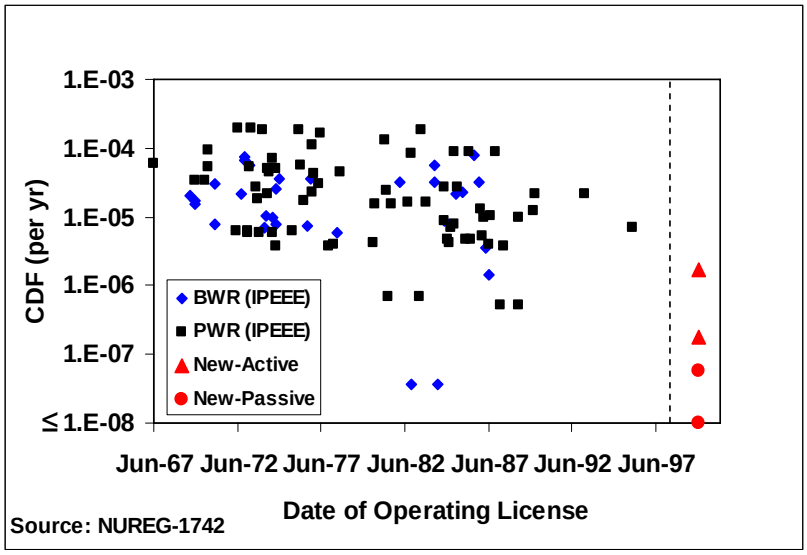

Figure 2. Fire CDF (at-power only)

Figure 3 compares the CDFs by hazard type or operating state for two active and two passive advanced LWR designs. The percentage contribution from at-power internal events, internal flooding, and fire are compared to low power/shutdown (LPSD) states from all hazards. (Other external hazards such as seismic and external floods are not shown because they are site-specific.) For new designs with active mitigation systems, fire CDF is comparable to internal events. For the passive plant designs, the CDFs for LPSD states are comparable if not greater than for at-power events. In the ESBWR design, the greatest contribution to shutdown risk comes from breaks in lines connected to the reactor vessel below the top of active fuel, with the lower drywell equipment hatch or personnel hatch open to facilitate work in the lower drywell. Losses of preferred power or loss of all service water during Mode 6 Unflooded are also large contributors [17].

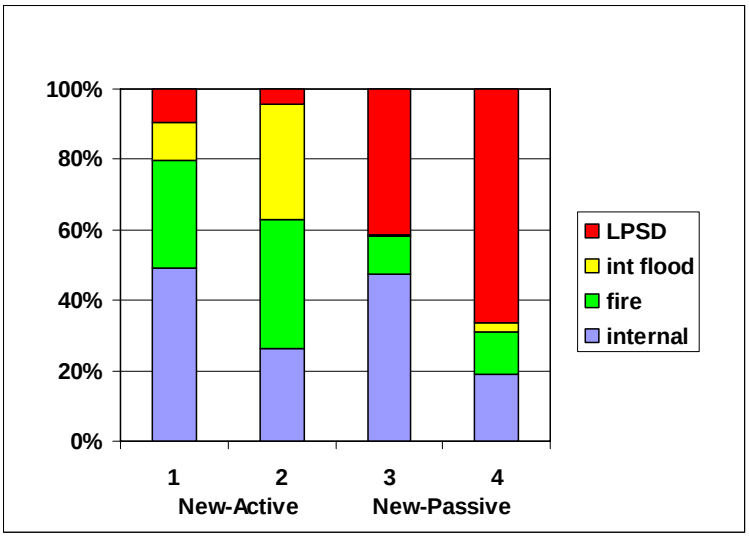

Figure 3. CDF contribution by hazard or operating state

Figure 4 shows the internal events LRF for advanced LWR designs. A comparison of LRF to operating plants is problematic in that there is no consistent definition of "large release," and operating plants have used large, early release frequency (LERF) as the primary level-2 PRA metric. Moreover, unlike CDF results from the MSPI, there is no single comprehensive source of updated LERF results across the industry. Thus, *significant* early release frequency distributions from the IPE results [4] are provided for reference. Given these qualifiers, it is generally observed that the frequencies of large radiological release from advanced LWR designs are approximately one to four orders of magnitude lower than the mean of operating plants. In Figure 5, the contribution to LRF by hazard type or operating state for two active and two passive advanced designs are provided. The contribution of LPSD to LRF for advanced designs is not insignificant because of assumptions regarding the inability to isolate containment during some phases of refueling operations.

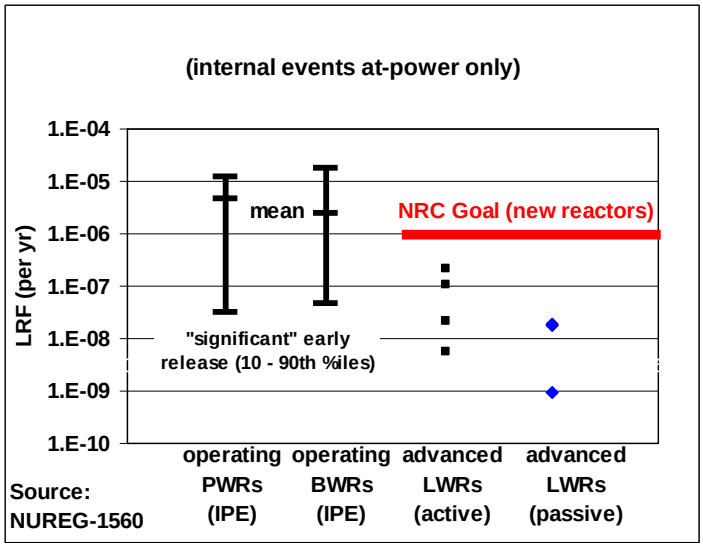

Figure 4. LRF by plant type

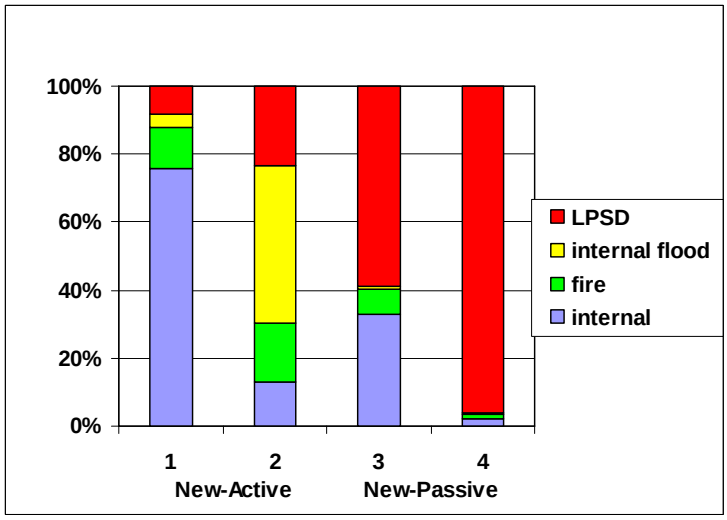

Figure 5. LRF contribution by hazard or operating state

In the SRM to SECY-89-102 [10], the Commission set an objective for CCFP of 0.1 for evolutionary designs as a means of providing a balance between prevention of core damage and mitigation. Figure 6 provides the CCFP for internal events for the seven advanced LWR designs. All meet the intent of the Commission's safety goal objectives. These values can be contrasted with higher IPE results of about 0.4 mean for PWRs with large dry/subatmospheric containments, 0.7 for BWR Mark I, and about 0.5 for both Mark II and III [4]. An important observation for advanced LWR designs is that there appears to be inherent limitations as to how low a value of CCFP can be achieved. Core damage *prevention* and *mitigation* functions are not independent. Systems and water sources available for mitigation may also be used for prevention, thus strongly coupling together core cooling and containment functions. For example, AP1000 in-vessel core debris retention (IVR) capability precludes challenges to containment. However, the success of IVR depends on the ability to depressurize the reactor coolant systems, draw water into the lower reactor cavity, and allow for passive containment cooling [16].

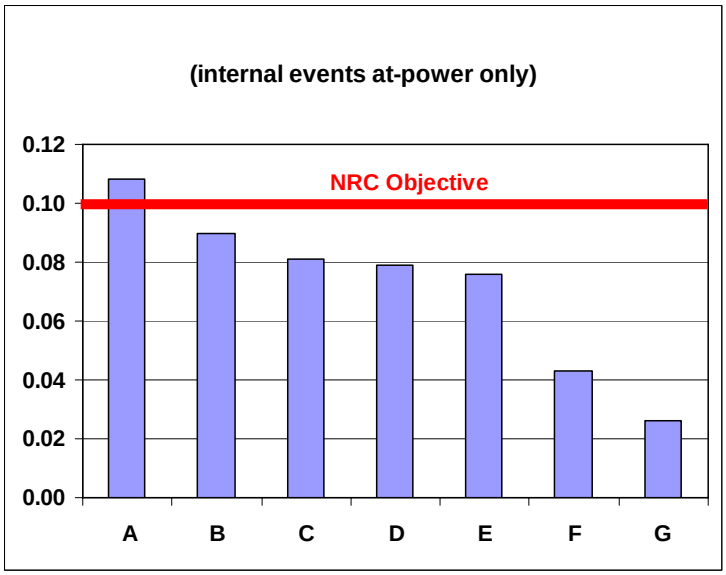

Figure 6. CCFP for new light-water reactor designs

2.4 Comparison of Other Risk Attributes and Insights

Table II lists a number of risk attributes and identifies whether the attribute is applicable to all, most, or some new LWR designs. For example, all the designs have three or more levels of redundancy for most mitigating and support systems. Additionally, all designs have physical separation of the safety systems, thus minimizing the contribution of spatial hazards to risk such as fire, internal flooding, and to a certain degree external events. The new PWR designs all have in-containment refueling water storage pools/pits, thus eliminating emergency core cooling switchover to recirculation as a major contributor to CDF as seen in earlier designs.

The contribution to CDF and LRF from ISLOCA is very low ($<10^{-9}$ /yr for all designs). This is in large part due to the staff recommendation in SECY-90-016 [9] to design (to the extent

practicable) all systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength at least equal to the full RCS pressure. Given the typical piping material and schedule proposed for use in new designs, the survival probability under ISLOCA conditions is found to be 90% or better [11]. Likewise, because of diverse reactor shutdown methods, ATWS events are found to contribute less than 10^{-7} /yr to CDF for all new designs.

Table II. Risk attributes of new LWRs

Risk Attribute	Applicable to:		
	All designs	Most designs	Some designs
High level of redundancy	x		
Physical separation of safety systems	x		
In-containment refueling water storage (PWRs)	x		
Very low contribution from ISLOCA (< 1E-9/yr)	x		
Low contribution to CDF from ATWS (< 1E-7/yr)	x		
Rapid RCS depressurization capability	x		
Core melt mitigation capability	x		
Containment combustible gas control capability	x		
AC-independent decay heat removal		x	
Low contribution to CDF from SBO (< 1E-7/yr)		x	
Significant credit in PRA for primary “Bleed & feed” (PWRs)		x	
Low % contribution to CDF from loss of HVAC		x	
Low % contribution to CDF from loss of cooling water support systems		x	
AC-independent containment cooling		x	
Low % contribution to CDF from RCP seal LOCAs (PWRs)			x
AC-independent reactor vessel make-up capability			x

All new designs, both BWRs and PWRs, have redundant and often diverse means of rapid RCS depressurization for design basis as well as severe accidents. This results in the ability to use low pressure injection systems in the event that high pressure systems are not available for core cooling, as well as minimizing the potential for severe accidents resulting in high pressure melt ejection scenarios. Moreover, all new designs have some engineered means of mitigating core melt accidents, either via lower reactor cavity flooding, a dedicated severe accident heat removal system, and/or other ex-vessel debris cooling capability. Finally, either through the use of glow plug igniters, autocatalytic recombiners, or containment inerting, all new LWR designs have means of containment combustible gas control for up to 100% fuel cladding-water reaction.

Table III shows the onsite AC power configuration for the five designs associated with actual or expected COLAs. The two passive designs have no need for emergency generators to provide AC power for large safety-related loads like emergency core cooling pumps. The designs with active mitigation capability have various combinations of diesel generators and combustion gas turbines. Most of the new designs also have some means of AC-independent core decay heat removal capability (i.e., without reliance on safety and non-safety on-site electrical generators). For example, two PWR designs have two steam turbine-driven emergency feedwater pumps apiece, while the ABWR has reactor core isolation cooling (RCIC), the ESBWR has multiple standby isolation condensers, and the AP1000 has the passive residual heat removal subsystem relying on one-time alignment of valves. As a result, the contribution of station blackout (SBO) events to CDF is less than 10^{-7} /yr for most designs. Figure 7 shows the *relative* contribution of loss of offsite power (LOOP) events in general and SBO in particular to internal events CDF. (The values for the ABWR are from a revised design originally intended for South Texas Project Units 3&4 [12], and are lower than those for the certified design [11].) LOOP/SBO events are clearly low contributors to CDF for the passive designs, AP1000 and ESBWR. However, for the new designs with active safety features the relative contribution of LOOP/SBO events to CDF are comparable to the tens of percents seen in the IPE results summary [4].

Table III. Onsite AC power generators

Design		Emergency (Class 1E)		SBO/AAC or non-safety (non-Class 1E)	
		Diesel Generator	Combustion Gas Turbine	Diesel Generator	Combustion Gas Turbine
Passive	AP1000	2 x 35 kW (1)	0	2 x 4 MW	0
	ESBWR	0	0	2 x 12 MW 2 x 1 MW (1)	0
Active	ABWR	3 x 7 MW	0	0	1 x 20 MW
	US-APWR	0	4 x 4.5 MW	0	2 x 4 MW
	U.S. EPR	4 x 9.5 MW	0	2 x 3.9 MW	0

Notes: (1) Ancillary for post-accident monitoring, lighting, HVAC, and other loads (post 72 hr).

Most of the new PWR designs take significant credit in the PRA for “bleed & feed” cooling of the RCS in the event of loss of all steam generator cooling. For example, the risk achievement worth (RAW) for operator failing to initiate this procedure is 10 or greater for the U.S. EPR and US-APWR [18,19].

Most of the new designs, especially the passive plants, have low contribution to CDF from loss of heating, ventilation and air conditioning (HVAC). However, “operator failing to recover room cooling locally” contributes about 40% to the U.S. EPR internal events CDF and has a RAW of about 30 [18]. Similarly, most designs have relatively low contribution to internal

events CDF from loss of cooling water support systems (e.g., component cooling water (CCW) and service water). However, component cooling water related failures contribute about 25% to internal events CDF at the US-APWR, and the RAW associated with operator failing to connect fire service water for alternate CCW is about 13 [19].

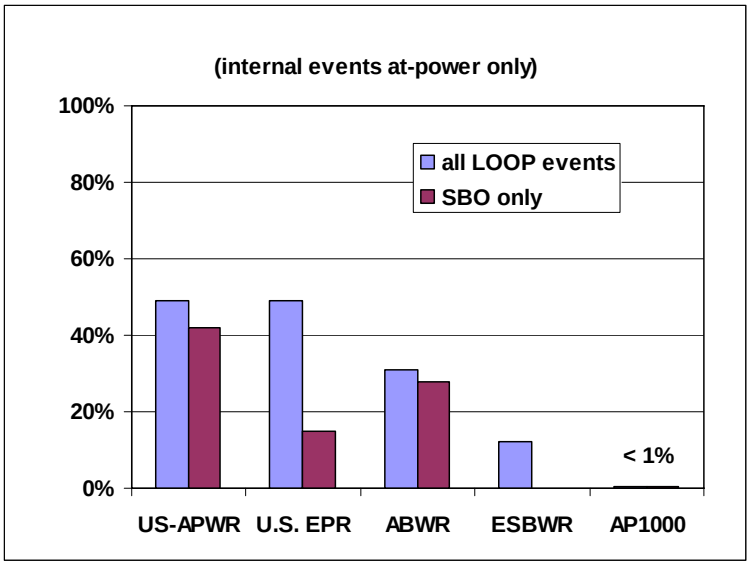

Figure 7. Loss of offsite power contribution to CDF

Most of the new PWR have some design feature to minimize the contribution of reactor coolant pump (RCP) seal LOCA to CDF. For example, the AP1000 has canned RCPs. The System 80+ relies on an RCP seal cooling pump powered by the combustion gas turbine. And the U.S. EPR has a standstill seal system with metal-to-metal contact. Nevertheless, RCP seal LOCAs are still major contributors to CDF for some designs. For the US-APWR, about 74% of the internal events CDF are associated with RCP seal LOCAs [19]. For the U.S. EPR, the RCP seal LOCA contributes about 24% to internal events CDF, 30% to internal flooding, and 43% to fire [18].

Finally, additional insights into the risk profiles for new 'active' and 'passive' plant designs can be gathered from the cited references. One can observe differences in contribution to CDF by various transient and LOCA initiators, and by the differing Fussell-Vesely (F-V) importances of human actions and hardware failures. For example, the highest F-V importances for one passive design are observed to be:

- Line strainer plugged
- Common-cause failure of squib valves
- Common-cause failure of sump screens, and
- Common-cause failure of gravity line check valves.

The high importances of these failure modes are uncharacteristic for reactor designs with active mitigation systems.

3 CONCLUSIONS

This paper demonstrates that the risk of new LWR designs as measured by CDF and LRF is substantially lower than the fleet of currently operating plants by one or more orders of magnitude. For all new LWR designs, the contribution of ATWS, ISLOCA, and SBO on an *absolute* scale are low because of features specifically designed to address these events.

Moreover, one can observe a clear distinction in the risk profiles between the passive designs (e.g., AP1000 and ESBWR), and those employing more conventional active mitigation systems (e.g., ABWR, U.S. EPR, and US-APWR). The passive designs tend to have a risk profile with balanced contributions from LOCAs and transients. There is minimal dependence on support systems, and offsite power is of low importance. Passive component failures tend to have the highest F-V importance measures.

On the other hand, the risk profiles for some of the new active designs are shown to mirror IPE results to some extent. For example, the *relative* contribution of support system initiators such as loss of CCW to CDF, the importance of HVAC as a support system, and the large impact of RCP seal LOCAs tend to resemble the risk attributes of operating plants in many regards.

4 ACKNOWLEDGMENTS

The author thanks his colleagues on the NRC staff who have tabulated information and data cited in this paper, notably, P. Appignani, T. V. Clark, G. Gulla, T. Hilsmeier, G. Kelly, J. Lai, M. Melnicoff, M. Patterson, H. Phan, M. Pohida, and N. Saltos.

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