Why Are Japanese Nuclear Power Plants "So Safe"? — Insights from Safety Improvement Assessments

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Abstract: Following the accident at Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Station, the Japanese regulatory authority required licensees to perform and disclose probabilistic risk assessment (PRA) every five years under the Periodic Safety Assessment of Continuous Improvement System. This study compares the at-power internal-event core damage frequencies (CDFs) of Japanese generation (Gen) II pressurized water reactors (PWRs) with those of U.S. Gen II PWRs and U.S. Gen III/III+ reactors, and analyzes the characteristics of Japanese PRA practice based on publicly available information. The comparison shows that the CDFs of Japanese Gen II PWRs are, on average, approximately one order of magnitude lower than those of U.S. Gen II PWRs, and that some Japanese plants report values comparable to, or lower than, those of U.S. Gen III/III+ reactors. These markedly low CDFs cannot be explained solely by the treatment of internal flooding or by differences in assessment timing. Rather, the difference appears to reflect multiple aspects of PRA modeling practice, including reliability data, CDF profiles, accident sequence structures, plant equipment features, and equipment credits. These results indicate that low CDFs should be interpreted not as direct measures of plant safety, but as risk estimates that reflect the data, assumptions, modeling choices, and equipment credits used in the PRA. This study points out that, to link risk-informed applications to continuous safety improvement, it is important to continuously enhance the transparency and defensibility of PRA.

1. INTRODUCTION

Following the accident at Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Station (the 1F accident), the Japanese regulatory authority required licensees to perform probabilistic risk assessment (PRA) every five years under the Periodic Safety Assessment of Continuous Improvement System introduced in December 2013. For generation (Gen) II pressurized water reactors (PWRs) that have resumed operation, Level 1 and Level 2 PRAs have been conducted for at-power internal, seismic, and tsunami events, together with a Level 1 PRA for shutdown internal events [1–12]. The reported core damage frequencies (CDFs) generally fall in the range of 10^{-6} to 10^{-5} per reactor-year (ry). These values are substantially lower than those reported for comparable plants in other countries. Such low CDFs are difficult to attribute solely to differences in plant design, especially among plants of similar type and generation. They suggest the need to examine the assumptions, modeling choices, scope definitions, and input data adopted in the PRA.

This study compares the publicly reported at-power internal-event CDFs of Japanese Gen II PWRs with those of U.S. Gen II PWRs and U.S. Gen III/III+ reactors, and identifies key characteristics of Japanese PRA practice. Possible contributors to the observed differences are then examined based on publicly available information, with particular attention to reliability data, CDF profiles, accident sequence structures, plant design and equipment features, and equipment credits. Based on these analyses, this study discusses issues that need to be addressed to further advance risk-informed applications in Japan.

The remainder of this paper is organized as follows. Section 2 compares the at-power internal-event CDFs of Japanese Gen II PWRs, U.S. Gen II PWRs, and U.S. Gen III/III+ reactors. Section 3 examines possible contributors to the differences in CDFs, focusing on reliability data, CDF profiles, and plant design and equipment features. Section 4 discusses the implications of these findings for risk-informed applications in Japan. Section 5 presents the conclusions.

2. COMPARISON OF INTERNAL-EVENT CDFS

2.1. Data and Scope of Comparison

This study uses the CDF from at-power internal-event Level 1 PRA as the primary metric for comparing PRA results between Japanese and U.S. PWRs, because it is less dependent on site-specific hazards than external-event risk metrics and relatively comparable information is available in public sources.

For the Japanese data, the point-estimate CDFs reported in the Safety Improvement Assessment reports submitted and disclosed under the Periodic Safety Assessment of Continuous Improvement System were used [1–12]. The scope of the analysis covers eleven Japanese PWR units for which PRA results had been published by 2025: Mihama Unit 3, Takahama Units 1, 3, and 4, Ohi Units 3 and 4, Ikata Unit 3, Sendai Units 1 and 2, and Genkai Units 3 and 4. These units are all classified as Gen II PWRs. In Japanese PRA practice, technical standards developed by the Atomic Energy Society of Japan are generally referred to as a technical basis.

For the U.S. data, CDFs for Gen II PWRs and Gen III/III+ reactors were used as comparison groups. For U.S. Gen II PWRs, the CDF values were taken primarily from Severe Accident Mitigation Alternatives (SAMA) analyses conducted as part of environmental reviews for license renewal, the Generic Environmental Impact Statement (GEIS), and related supplements to NUREG-1437 [13,14]. SAMA analysis is performed to compare the risk reduction benefits of severe accident mitigation alternatives with their implementation costs, based on plant-specific PRA. PRA used in U.S. SAMA analyses is developed in accordance with guidance such as NEI 05-01 Rev. A [15]. In addition, the technical adequacy of PRA used for risk-informed applications in the U.S. is supported through a framework that includes peer review against the American Society of Mechanical Engineers / American Nuclear Society PRA Standard and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.200 [16]. It should be noted, however, that these CDF values are based on analyses performed for license renewal reviews and do not necessarily represent the most recent PRA results for each plant.

For U.S. Gen III/III+ reactors, CDF values for new reactor designs reported in sources such as NUREG-2249 [17] were used as supplementary reference values. These values are based on PRAs performed at the design, design certification, or combined license review stage, rather than on PRAs supported by operating experience from existing reactors.

Differences in PRA scope must also be considered in the comparison. In the Japanese Safety Improvement Assessment reports, internal flooding is generally treated separately from internal events. In contrast, the CDF values reported in U.S. SAMA analyses or in the GEIS may include contributions from internal flooding. To improve consistency in the scope of the comparison, when the contribution from internal flooding could be identified from publicly available information, it was subtracted from the reported U.S. CDF. The resulting values were then treated as at-power internal-event CDFs excluding internal flooding.

Table 1 summarizes the plants, reactor types, internal-event CDFs, report years, and sources used in the comparison. In this table, “report year” denotes the year associated with the public reporting or regulatory submission of the PRA result used in this study. For U.S. Gen II PWRs, the two years shown correspond to the license renewal application year and the license renewal issuance year. For U.S. Gen III/III+ reactors, they correspond to the combined license application year and, where applicable, the combined license issuance year; for projects that were suspended, withdrawn, or terminated, the second year indicates the year of the corresponding NRC review disposition.

Table 1: Data Sources of At-Power Internal-Event CDF Comparison

Country	Plant	Gen	Model	CDF (/ry)	Report year	Ref.
Japan	Mihama 3	II	MHI 3-loop PWR	3.20E-06	2023	[2]
Japan	Takahama 1	II	MHI 2-loop PWR	3.00E-06	2025	[3]
Japan	Takahama 3	II	MHI 3-loop PWR	7.50E-07	2023	[4]
Japan	Takahama 4	II	MHI 3-loop PWR	7.50E-07	2023	[5]
Japan	Ohi 3	II	MHI 4-loop PWR	1.20E-06	2023	[6]
Japan	Ohi 4	II	MHI 4-loop PWR	1.20E-06	2023	[7]
Japan	Ikata 3	II	MHI 3-loop PWR	3.10E-06	2025	[8]
Japan	Sendai 1	II	MHI 3-loop PWR	3.00E-06	2022	[9]
Japan	Sendai 2	II	MHI 3-loop PWR	3.00E-06	2023	[10]
Japan	Genkai 3	II	MHI 4-loop PWR	4.30E-06	2023	[11]
Japan	Genkai 4	II	MHI 4-loop PWR	4.30E-06	2023	[12]
U.S.	Beaver Valley 2	II	Westinghouse 3-loop PWR	8.33E-06	2007 / 2009	[13]
U.S.	Braidwood 1	II	Westinghouse 4-loop PWR	3.03E-05	2013 / 2016	[13]
U.S.	Braidwood 2	II	Westinghouse 4-loop PWR	2.94E-05	2013 / 2016	[13]
U.S.	Byron 1	II	Westinghouse 4-loop PWR	3.41E-05	2013 / 2015	[13]
U.S.	Byron 2	II	Westinghouse 4-loop PWR	3.24E-05	2013 / 2015	[13]
U.S.	Callaway 1	II	Westinghouse 4-loop PWR	1.70E-05	2011 / 2015	[13]
U.S.	Catawba 1/2	II	Westinghouse 4-loop PWR	3.30E-05	2001 / 2003	[13]
U.S.	Comanche Peak 1/2	II	Westinghouse 4-loop PWR	1.22E-06	2022 / 2024	[14]
U.S.	Harris 1	II	Westinghouse 3-loop PWR	7.64E-06	2006 / 2008	[13]
U.S.	Indian Point 2	II	Westinghouse 4-loop PWR	1.32E-05	2007 / 2018	[13]
U.S.	Indian Point 3	II	Westinghouse 4-loop PWR	9.30E-06	2007 / 2018	[13]
U.S.	Millstone 3	II	Westinghouse 4-loop PWR	2.60E-05	2004 / 2005	[13]
U.S.	Palo Verde 1/2/3	II	Combustion Engineering 2-loop/CE80 PWR	5.10E-06	2008 / 2011	[13]
U.S.	Seabrook 1	II	Westinghouse 4-loop PWR	5.66E-06	2010 / 2019	[13]
U.S.	South Texas Project 1/2	II	Westinghouse 4-loop PWR	3.90E-06	2010 / 2017	[13]
U.S.	St. Lucie 2	II	Combustion Engineering PWR	2.38E-05	2001 / 2003	[13]
U.S.	Summer 1	II	Westinghouse 3-loop PWR	5.60E-05	2002 / 2004	[13]
U.S.	Vogtle 1/2	II	Westinghouse 4-loop PWR	1.55E-05	2007 / 2009	[13]
U.S.	Waterford 3	II	Combustion Engineering 2-loop PWR	1.10E-05	2016 / 2018	[13]
U.S.	Wolf Creek 1	II	Westinghouse 4-loop PWR	3.00E-05	2006 / 2008	[13]
U.S.	Fermi 3	III+	GE Hitachi ESBWR	1.70E-08	2008 / 2015	[17]
U.S.	Comanche Peak 3/4	III+	U.S. APWR	1.20E-06	2008 / 2013	[17]
U.S.	Calvert Cliffs 3	III+	U.S. EPR	5.30E-07	2007 / 2015	[17]
U.S.	South Texas Project 3/4	III	ABWR-GEH	1.60E-07	2007 / 2016	[17]
U.S.	Turkey Point 6/7	III+	Westinghouse AP1000	2.40E-07	2009 / 2018	[17]
U.S.	Vogtle 3/4	III+	Westinghouse AP1000	2.40E-07	2008 / 2012	[17]
U.S.	V.C. Summer 2/3	III+	Westinghouse AP1000	2.40E-07	2008 / 2012	[17]
U.S.	Levy 1/2	III+	Westinghouse AP1000	2.40E-07	2008 / 2016	[17]
U.S.	William States Lee 1/2	III+	Westinghouse AP1000	2.40E-07	2007 / 2016	[17]
U.S.	Bellefonte 3/4	III+	Westinghouse AP1000	2.40E-07	2007 / 2016	[17]

2.2. Comparison Results

Figure 1 compares the at-power internal-event CDFs for Japanese Gen II PWRs, U.S. Gen II PWRs, and U.S. Gen III/III+ reactors. As described in the previous section, the CDFs used in this comparison are at-power internal-event Level 1 PRA values excluding internal flooding. Each point in the figure represents the CDF of an individual plant. The figure combines box plots and violin plots.

This figure shows that the CDFs of Japanese Gen II PWRs are distributed in a lower range than those of U.S. Gen II PWRs. The Japanese CDFs range from 7.5×10^{-7} to 4.3×10^{-6} /ry, whereas the CDFs of U.S. Gen II PWRs are distributed over a wider range from 3.9×10^{-6} to 5.6×10^{-5} /ry. The CDFs of U.S. Gen III/III+ reactors are generally distributed in an even lower range. Some Japanese plants show CDF values comparable to, or lower than, those reported for U.S. Gen III/III+ reactors.

Table 2 summarizes the descriptive statistics of the CDFs for each group. The mean and median CDFs of Japanese Gen II PWRs are approximately one order of magnitude lower than those of U.S. Gen II PWRs. The interquartile range of the Japanese plants is also located in a lower CDF range than that of the U.S. plants, indicating that the difference is not limited to the mean value but is observed across the overall distribution. In addition, the Japanese CDFs are concentrated in a relatively narrow range, whereas the U.S. Gen II PWRs show larger plant-to-plant variability.

Table 2: Summary Statistics of At-Power Internal-Event CDFs (/ry)

	n	Min	Q1	Med	Mean	Q3	Max	SD
Japan Gen II (PWR)	11	7.50E-07	1.20E-06	3.00E-06	2.53E-06	3.15E-06	4.30E-06	1.33E-06
U.S. Gen II (PWR)	26	3.90E-06	9.31E-06	1.75E-05	2.36E-05	3.76E-05	5.60E-05	1.69E-05
U.S. Gen III/III+	18	1.70E-08	2.40E-07	2.40E-07	3.41E-07	2.40E-07	1.20E-06	3.26E-07

Note: n = number of plants or reactor units; Min = minimum; Q1 = first quartile; Med = median; Mean = arithmetic mean; Q3 = third quartile; Max = maximum; SD = standard deviation.

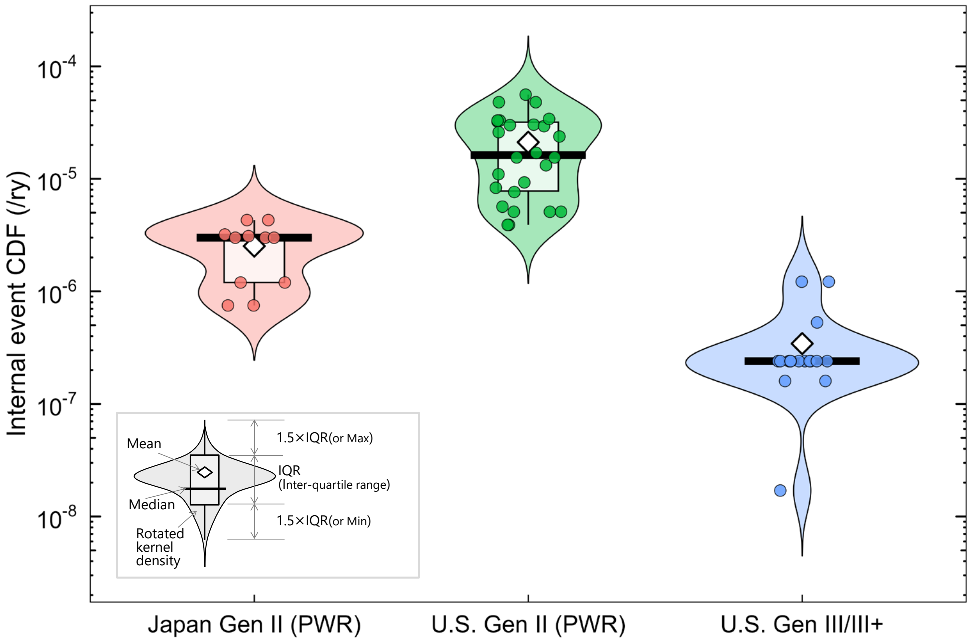

Figure 1: At-Power Internal-Event CDFs for Japanese and U.S. Reactors

When interpreting the difference between the Japanese and U.S. results, the difference in the timing of the underlying assessments should also be considered. The Japanese CDFs are mainly based on Safety Improvement Assessment submissions in the early 2020s, whereas the CDFs for U.S. Gen II PWRs are mainly based on SAMA analyses performed for license renewal reviews from the 2000s to the 2010s. Differences in assessment timing may affect PRA maturity and, consequently, reported CDFs.

To examine this point, Figure 2 shows the relationship between the report year and CDF for Japanese and U.S. Gen II PWRs. The report year is defined as in Table 1. Within the set of publicly available data compiled in this study, no clear monotonic trend is observed between report year and CDF. In particular, the data do not show a clear tendency for older assessments to have higher CDFs and more recent assessments to have lower CDFs. Therefore, the observed difference between Japanese and U.S. CDFs cannot be explained by the difference in assessment timing alone.

The comparison in this section shows that the at-power internal-event CDFs of Japanese Gen II PWRs are markedly lower than those of U.S. Gen II PWRs in terms of the mean, median, and overall distribution. This difference remains clearly visible in the publicly available data even after accounting for the treatment of internal flooding and differences in report year.

At the same time, the Japanese and U.S. plants compared here are all Gen II PWRs of broadly similar safety design philosophy. The approximately one-order-of-magnitude difference in CDF should therefore not be interpreted simply as a difference in plant-specific safety. Rather, the difference is likely to reflect multiple aspects of PRA practice, including PRA scope, initiating-event frequencies, reliability data, accident sequence modeling, human reliability analysis, and the treatment of conservative or realistic assumptions, in addition to differences in plant design and equipment features.

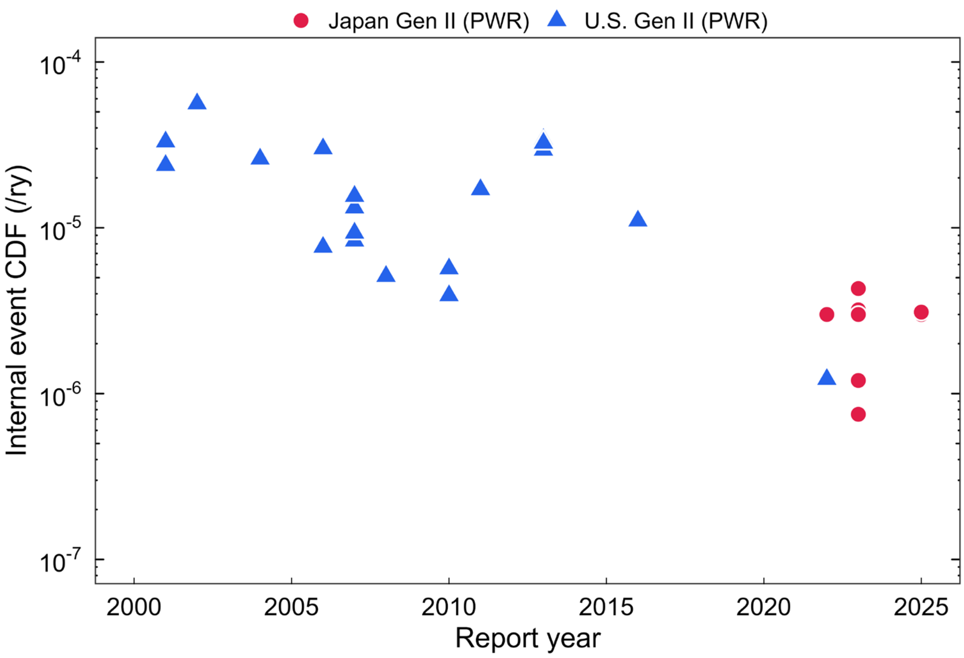

Figure 2: Relationship Between Report Year and At-Power Internal-Event CDF

3. POSSIBLE CAUSES OF DIFFERENCES IN CDFS

This section examines possible contributors to the observed difference in the CDFs, to the extent possible based on publicly available information. Section 3.1 discusses reliability data. Section 3.2 examines CDF profiles, including the contribution structure of initiating events and accident sequences. Section 3.3 considers differences in plant design and equipment features. Section 3.4 summarizes these factors. The objective of this section is not to re-evaluate the PRA models or to quantify the contribution of each factor rigorously, but to identify the main explanatory factors that can be inferred from public information and to clarify the limitations of such an analysis.

3.1. Reliability Data

The CDF in PRA depends on basic event probabilities, including component failure rates, failure probabilities on demand, common cause failure probabilities, and human error probabilities. Therefore, even if the accident sequence structure is the same, the evaluated CDF can vary substantially depending on the reliability data used as input.

A previous study comparing Japanese component reliability data for PRA with U.S. data showed that, for several components and failure modes, including emergency diesel generators, pumps, turbine-driven pumps, and engine-driven pumps, the mean failure rates or failure probabilities on demand in the Japanese data tend to be lower than those in the U.S. data [18]. For some items, the difference reaches approximately one order of magnitude. However, the Japanese values are not lower for all components and failure modes; both the magnitude and direction of the difference depend on the specific item.

Such differences in reliability data may be one factor contributing to the lower CDFs reported for Japanese PWRs. In particular, if the failure rates of components important to safety functions are evaluated as lower, the CDFs of accident sequences such as loss of electrical power, loss of secondary heat removal, and failure of Emergency Core Cooling System (ECCS) injection would also decrease. However, this study does not recalculate CDFs using alternative reliability data and therefore cannot quantitatively decompose the contribution of reliability data to the observed difference. Accordingly, differences in reliability data are treated here as a plausible contributing factor to the difference between Japanese and U.S. CDFs.

3.2. CDF Profiles

This section compares CDF profiles, namely the contribution structure of initiating events or accident sequences. Because the classifications used in publicly available PRA results differ between Japan and the United States, CDFs are first summarized by common initiating-event categories where possible. Accident-sequence-group CDFs are then used as supplementary information for Japanese plants, for which initiating-event-level CDFs are publicly available only for a limited number of units.

Figure 3 shows CDFs by initiating-event category for Japanese and U.S. Gen II PWRs. For the Japanese plants, the analysis is limited to Sendai Units 1/2 and Genkai Units 3/4, for which initiating-event-level CDF information is publicly available. For comparison, the initiating events or initiating-event groups reported in the original PRA documents were aggregated into common categories: loss-of-coolant accident (LOCA), bypass/containment bypass, transients/secondary-side events, electrical power loss, support-system loss, anticipated transient without scram (ATWS)/shutdown failure, and other/residual events. This aggregation simplifies plant-specific PRA classifications, but it was used to provide an overview of the main contributors to at-power internal-event CDFs in Japan and the United States.

Figure 3 shows that, although there is substantial plant-to-plant variability among U.S. Gen II PWRs, electrical power loss is one of the major contributing categories for many plants. This suggests that differences in the treatment of loss of offsite power, station blackout, emergency power systems, recovery actions, and the associated initiating-event frequencies and modeling assumptions may be important in understanding the difference between Japanese and U.S. CDFs. In particular, systematic differences in the frequency of loss of offsite power or in the probability of recovery following loss of electrical power could have a significant effect on the total CDF.

At the same time, the contribution from electrical power loss alone cannot explain the overall difference between Japanese and U.S. CDFs. Even when the contribution from electrical power loss is excluded, several U.S. Gen II PWRs still show CDF contributions of the order of 10^{-6} to 10^{-5} /ry from other categories, such as LOCA, transients/secondary-side events, and support-system loss. By contrast, for the Japanese plants for which initiating-event-level CDFs are available, namely Sendai Units 1/2 and Genkai Units 3/4, the absolute contributions of individual categories are generally lower. Thus, although electrical power loss is an important feature of the U.S. CDF profiles and warrants further examination, it is unlikely to be the single dominant factor explaining the approximately one-order-of-magnitude difference in at-power internal-event CDFs between Japan and the United States.

Figure 4 presents at-power internal-event CDFs for the eleven Japanese Gen II PWRs by accident sequence group. This classification is based on major functional failures or accident progressions leading to core damage, rather than on initiating-event categories, and is therefore used as supplementary information to characterize the internal structure of Japanese CDFs.

Figure 4 shows that loss of ECCS recirculation and loss of secondary heat removal are major contributors for many Japanese PWRs. In contrast, Takahama Units 3/4 and Ohi Units 3/4 show relatively small contributions from loss of ECCS recirculation. In addition, Takahama Units 3/4 show small contributions from loss of secondary heat removal. These differences may be related to plant-specific equipment features, such as automation of ECCS recirculation switchover and the number of pressurizer power-operated relief valves. These features are examined in the next section.

Overall, the CDF profiles indicate that the difference between Japanese and U.S. CDFs is not limited to the magnitude of the total CDF. Contributions from electrical power loss are large for many U.S. plants, but this category alone cannot explain the overall difference. The Japanese accident-sequence-group profiles also show plant-to-plant differences. Therefore, CDF differences should be examined in terms of initiating-event frequencies, accident sequence composition, equipment features, and their treatment in PRA.

3.3. Plant Design and Equipment Differences

Differences in plant design and equipment features may also affect reported CDFs. For Japanese PWRs, one major equipment feature introduced following the 1F accident is the installation of additional severe accident management equipment, including the Specialized Safety Facility. These systems may contribute to CDF reduction by providing additional means for core damage prevention and accident mitigation. However, a previous assessment estimated that the CDF reduction effect of the Specialized Safety Facility is approximately a factor of two, comparable to the effect of severe accident management equipment added in response to the new regulatory requirements [19]. Although this represents an important risk reduction effect, it is not sufficient by itself to explain the approximately one-order-of-magnitude difference observed between Japanese and U.S. Gen II PWRs.

Within the Japanese PWR fleet, several design and equipment differences are consistent with the CDF profiles discussed in the previous section. In particular, Takahama Units 3/4 and Ohi Units 3/4 show relatively small contributions from loss of ECCS recirculation. This is attributable to the automation of the switchover of the ECCS water source from the refueling water storage tank to the recirculation sump, which reduces the contribution from operator failure in the recirculation switchover [20]. In addition, Takahama Units 3/4 show small contributions from loss of secondary heat removal. This is attributable to the installation of three pressurizer power-operated relief valves (PORVs), which increases the credited reliability of feed-and-bleed cooling following loss of secondary heat removal [20].

Table 3 summarizes selected equipment features related to these accident sequences, namely whether ECCS recirculation switchover is automated and the number of PORVs. The table shows that automated ECCS recirculation switchover and multiple PORVs are not unique to Japanese PWRs; some U.S. Gen II PWRs also have similar equipment features. Therefore, these features are relevant for explaining differences in CDF among Japanese PWRs, but they do not by themselves explain the overall difference in CDF between Japanese and U.S. Gen II PWRs. The difference between Japanese and U.S. CDFs should instead be understood as the combined result of equipment features, reliability data, initiating-event frequencies, accident sequence modeling, and other PRA assumptions.

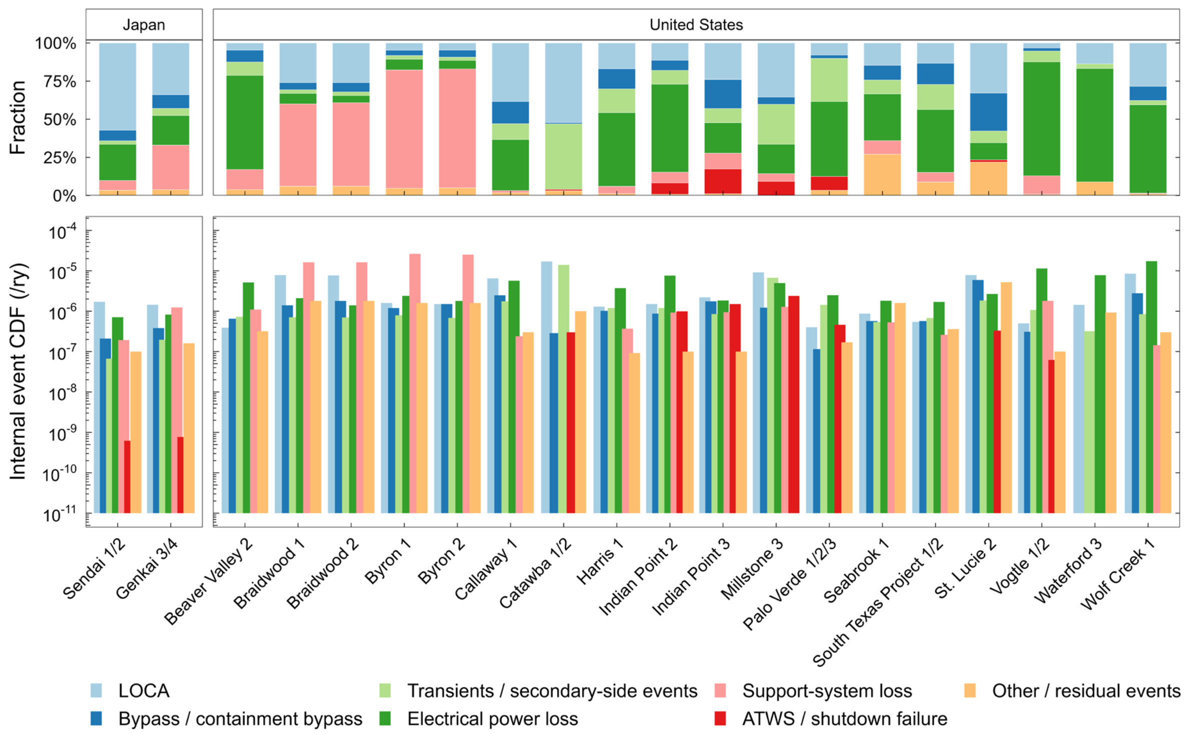

Figure 3: Internal-Event CDF Profiles by Initiating-Event Category for Japan and U.S. Gen II PWRs

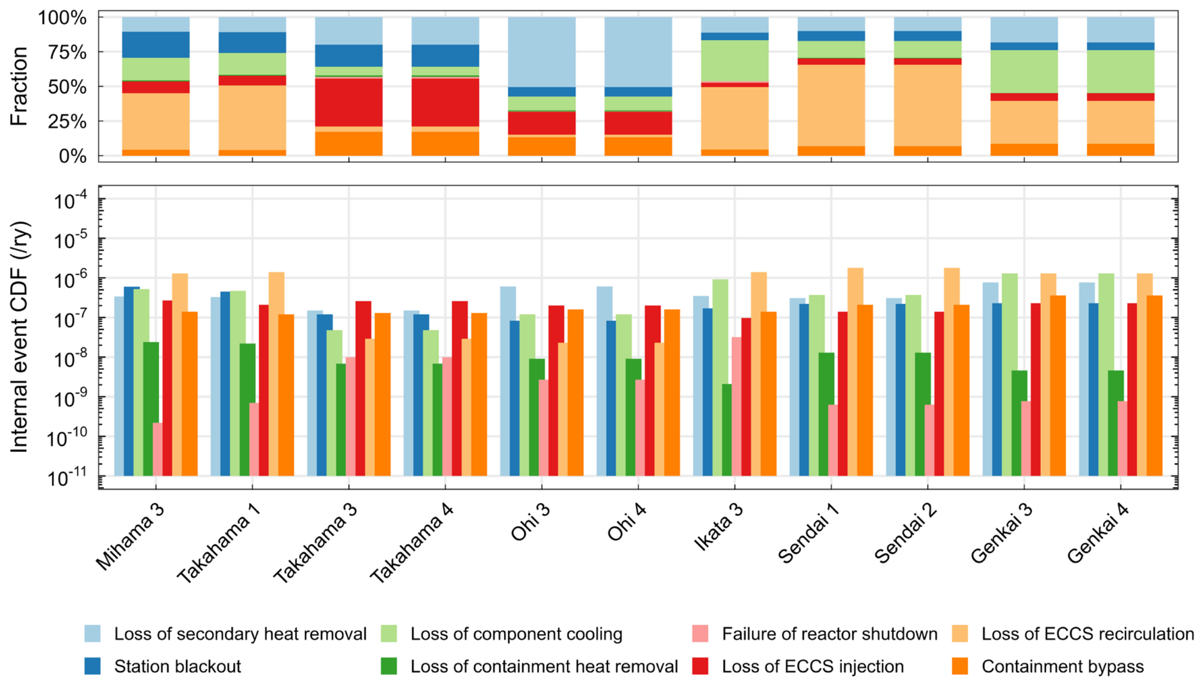

Figure 4: Internal-Event CDF Profiles by Accident Sequence Group for Japanese Gen II PWRs

3.4. Summary of Possible Causes

The preceding analysis suggests that the low internal-event CDFs reported for Japanese Gen II PWRs cannot be explained by a single factor. With respect to reliability data, previous studies have reported that Japanese failure rates or failure probabilities on demand tend to be lower than U.S. values for several important components and failure modes. Such differences may contribute to lower CDFs for

Japanese plants. However, this study does not recalculate CDFs using alternative reliability data and therefore cannot quantitatively decompose their contribution.

The comparison of CDF profiles indicates that contributions from electrical power loss tend to be large for U.S. Gen II PWRs. However, electrical power loss alone cannot explain the overall difference between Japanese and U.S. CDFs, because U.S. plants also show substantial contributions from other categories, including LOCA, transients/secondary-side events, and support-system loss. Similarly, plant design and equipment features, such as the Specialized Safety Facility, automation of ECCS recirculation switchover, and the number of PORVs, may contribute to CDF reduction, but none of these factors alone is sufficient to explain the approximately one-order-of-magnitude difference between Japan and the United States.

Therefore, the low internal-event CDFs reported for Japanese PWRs should be understood not as the result of differences in plant-specific safety, but as the combined effect of reliability data, initiating-event frequencies, accident sequence structures, plant equipment features, PRA scope, and modeling assumptions. Other factors not examined in detail within the scope of the publicly available sources reviewed in this study may also affect CDFs. These include the effects of operational management and maintenance on initiating-event frequencies and equipment reliability and unavailability, differences in severe accident management (SAM) equipment, procedures, and training, human reliability analysis (HRA), and the treatment of common cause failures (CCFs). Thus, the observed difference should be interpreted in light of both the factors identified here and the limitations of the analysis.

Table 3: ECCS Recirculation Switchover Automation and PORV Configuration

Country	Plant	Gen	CDF (/ry)	ECCS recirculation switchover	PORVs
Japan	Mihama 3	II	3.20E-06	Manual	2
Japan	Takahama 1	II	3.00E-06	Manual	2
Japan	Takahama 3	II	7.50E-07	Auto	3
Japan	Takahama 4	II	7.50E-07	Auto	3
Japan	Ohi 3	II	1.20E-06	Auto	2
Japan	Ohi 4	II	1.20E-06	Auto	2
Japan	Ikata 3	II	3.10E-06	Manual	2
Japan	Sendai 1	II	3.00E-06	Manual	2
Japan	Sendai 2	II	3.00E-06	Manual	2
Japan	Genkai 3	II	4.30E-06	Manual	2
Japan	Genkai 4	II	4.30E-06	Manual	2
U.S.	Beaver Valley 2	II	8.33E-06	Semi-auto / manual	3
U.S.	Braidwood 1	II	3.03E-05	Semi-auto / manual	2
U.S.	Braidwood 2	II	2.94E-05	Semi-auto / manual	2
U.S.	Byron 1	II	3.41E-05	Semi-auto / manual	2
U.S.	Byron 2	II	3.24E-05	Semi-auto / manual	2
U.S.	Callaway 1	II	1.70E-05	Semi-auto / manual	2
U.S.	Catawba 1/2	II	3.30E-05	Semi-auto / manual	3
U.S.	Comanche Peak 1/2	II	1.22E-06	Semi-auto / manual	2
U.S.	Harris 1	II	7.64E-06	Auto	3
U.S.	Indian Point 2	II	1.32E-05	Manual	2
U.S.	Indian Point 3	II	9.30E-06	Manual	2
U.S.	Millstone 3	II	2.60E-05	Semi-auto / manual	2
U.S.	Palo Verde 1/2/3	II	5.10E-06	Auto	0
U.S.	Seabrook 1	II	5.66E-06	Semi-auto / manual	2
U.S.	South Texas Project 1/2	II	3.90E-06	Auto	2
U.S.	St. Lucie 2	II	2.38E-05	Auto	2
U.S.	Summer 1	II	5.60E-05	Semi-auto / manual	3
U.S.	Vogtle 1/2	II	1.55E-05	Semi-auto / manual	3
U.S.	Waterford 3	II	1.10E-05	Auto	0
U.S.	Wolf Creek 1	II	3.00E-05	Semi-auto / manual	2

4. IMPLICATIONS FOR RISK-INFORMED APPLICATIONS IN JAPAN

The analyses in the preceding sections show that the internal-event CDFs reported for Japanese Gen II PWRs are markedly lower than those for U.S. Gen II PWRs, and that this difference appears to involve multiple aspects of the PRA modeling process, including reliability data, CDF profiles, accident sequence structures, plant equipment features, and equipment credits. These low CDFs should therefore be regarded not as direct representations of plant safety, but as results that call for examination of the data, assumptions, modeling choices, and equipment credits underlying Japanese PRAs.

Figures 5 and 6 show how PRA results have changed in Japan and the United States. For many Japanese PWRs, the CDFs reported under the Periodic Safety Assessment of Continuous Improvement System are higher than the PRA results after the implementation of voluntary accident management measures in 2004 [21]. In contrast, for U.S. PWRs, the CDFs in the SAMA analyses tend to be lower than the 1996 License Renewal (LR) GEIS estimates or Individual Plant Examination (IPE) results, which mainly represent plant-specific PRA results from the early to mid-1990s. This contrasting trend suggests that, in earlier Japanese PRAs, sufficiently defensible data, assumptions, and modeling choices may not have been adopted to support risk-informed applications. This observation is consistent with Apostolakis's characterization of the "10^{-6} culture" in Japan [23].

This point is important for advancing risk-informed applications under safety goals. The issue is not that Japanese CDFs are low, but whether such low values are supported by transparent assumptions, data, modeling choices, and equipment credits, and whether they are defensible as risk information for decision-making. If PRA underestimates risk, using its results as a basis for demonstrating compliance with safety goals or the adequacy of safety measures may work toward justifying the status quo rather than promoting improvement through risk-informed applications. Such use would not lead to continuous safety improvement, which is one of the key lessons from the 1F accident.

Therefore, to advance risk-informed applications in Japan in a substantive manner, PRA should be developed using the best available knowledge and technology at the time, and the bases for major risk contributors, modeling assumptions, equipment credits, sensitivities, uncertainties, and data should be presented transparently. This does not mean that risk-informed applications should be deferred until a complete PRA is available. However, when PRA is used for decision-making, it is essential to continually examine and improve whether its results provide transparent and defensible risk information.

5. CONCLUSIONS

To clarify the characteristics of Japanese PRA practice and to identify issues for advancing risk-informed applications in Japan, this study examined at-power internal-event CDFs reported for Japanese Gen II PWRs and compared them with those reported for U.S. Gen II PWRs. Although internal-event CDFs are less dependent on site-specific hazards than external-event risk metrics, the comparison showed that the CDFs of Japanese Gen II PWRs are, on average, approximately one order of magnitude lower than those of U.S. Gen II PWRs. Some Japanese plants also show CDFs comparable to, or lower than, those reported for U.S. Gen III/III+ reactors.

These markedly low CDFs cannot be explained solely by the treatment of internal flooding or by differences in assessment year. The analysis suggests that differences in reliability data, CDF profiles, accident sequence structures, and plant equipment features may act in combination. However, none of these factors alone appears sufficient to explain the observed difference between Japan and the United States. Therefore, the low CDFs reported for Japanese PWRs should be understood not as direct representations of plant safety, but as results that strongly reflect the data, assumptions, modeling choices, and equipment credits used in the PRA. Within the scope of the publicly available sources examined in this study, important factors, including operational management and maintenance practices, SAM equipment, procedures, and training, HRA, and CCF treatment, could not be examined in detail and should be addressed in future work.

This finding has important implications for risk-informed applications under safety goals. If PRA underestimates risk, a low CDF may serve as a formal basis for justifying the status quo rather than as information that supports continuous safety improvement. To advance risk-informed applications in Japan, PRA should be developed and continuously improved with transparency and defensibility, based on the best available knowledge and technology at the time.

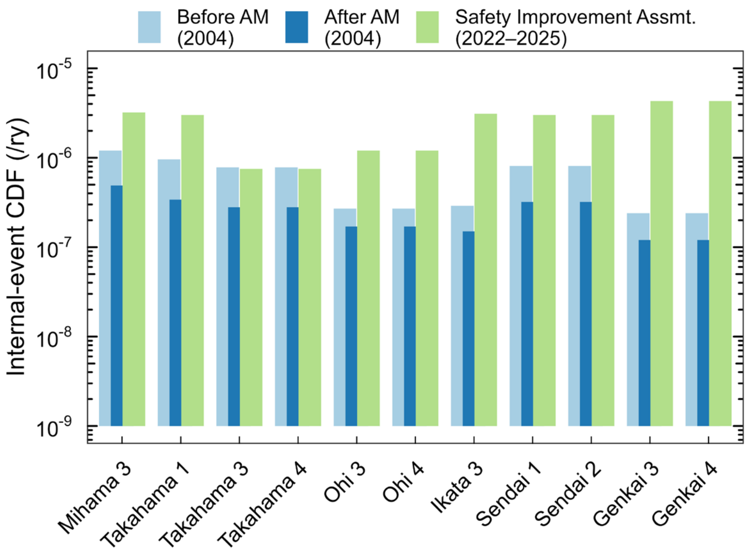

Figure 5: Evolution of Internal-Event CDF Estimates for Japanese Gen II PWRs (reproduced from [22])

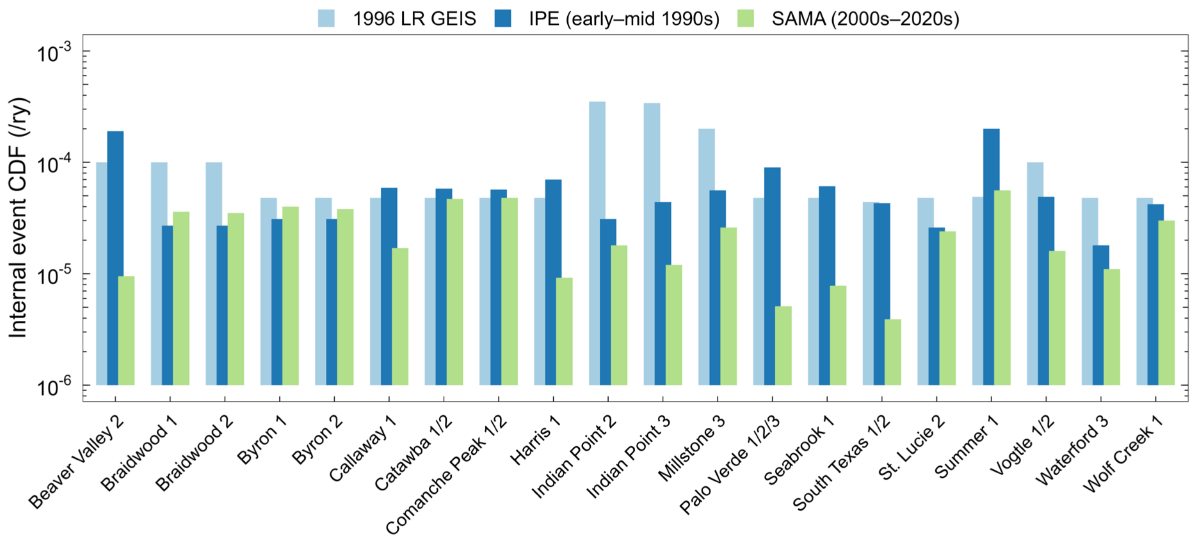

Figure 6: Evolution of Internal-Event CDF Estimates for U.S. Gen II PWRs

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