Risk Metrics for Operating New Reactors

Introduction

As new reactors progress through the 10 CFR 52 licensing process (i.e., Part 52), the focus is shifting from evaluation of the basic standardized design toward operational risk considerations. This transition has become somewhat confounded due to the differences in the metrics used in the Part 52 licensing process versus those used in the regulatory programs that apply to operating reactors.

The purpose of this paper is to consider the value and applicability of various risk metrics for the two distinct phases of plant licensing: design and operation. Furthermore, the paper discusses the quantitative thresholds that may apply to those risk metrics.

Evolution of Risk Metrics

In order to frame the situation, a brief summary is provided on the evolution of risk metrics that has occurred over the past 20 years.

The Part 52 licensing process and associated industry guidance for consideration of severe accident risks was promulgated in the early 1990s. At that time, the Safety Goal Policy Statement [Ref. 1] had only recently been issued and the industry had relatively little experience with using risk information in regulatory decision-making.

The Part 52 design certification process [Ref. 2] includes a requirement that the licensing of advanced reactor design would include a probabilistic risk assessment (PRA) for internal hazards (i.e., internal events, internal flooding, and internal fires) for at-power and selected shutdown operating states. External hazards that are site-specific (e.g., seismic events, high winds, external floods, etc.) are primarily addressed using design features and capabilities intended to minimize risks. For seismic hazards, a PRA-based seismic margin analysis (SMA) is performed to demonstrate that there is a high confidence of a low probability of core damage scenarios occurring for seismic events even more severe than the Safe Shutdown Earthquake (SSE).

Through a series of SECYs and associated Staff Requirements Memorandums (SRMs) issued during 1990 [Ref. 3, 4], the Commission elected to include three risk metrics and associated quantitative goals in the design certification process:

- **Core Damage Frequency (CDF) < 1 x 10^{-4}/year** – a measure of overall safety performance in the prevention of severe accidents
- **Large Release Frequency (LRF) < 1 x 10^{-6}/year** – a measure of prevention of significant offsite consequences
- **Conditional Containment Failure Probability (CCFP) < 0.1** – a measure of the capability of the design to mitigate a severe accident

By the early-1990s, the Individual Plant Examinations (IPEs) for the operating fleet of reactors were drawing to a close and the Nuclear Regulatory Commission (NRC) began to grapple with how to best implement the Safety Goal Policy Statement and severe accident considerations in regulatory decision-making. In 1993, the Commission unanimously voted to abandon efforts to define “large release” after the staff found it impossible to provide a simple definition that was not "several orders of magnitude more conservative than the QHOs" (Quantitative Health Objectives) [Ref. 5].

In 1995, the Nuclear Regulatory Commission (NRC) issued the PRA Policy Statement [Ref. 6] that encouraged the use of PRA in all regulatory matters. In 1998, the NRC issued Regulatory Guide (RG) 1.174 [Ref. 7], providing a risk-informed integrated decision-making framework. The development of RG 1.174 relied upon a body of work that demonstrated that CDF and large, <u>early</u> release frequency (LERF) were the appropriate surrogates for the Quantitative Health Objectives (QHOs) of the Safety Goal Policy Statement for operating reactors (and not CCFP and LRF). Of equal significance, the framework defined by RG 1.174 established the concept of "risk-informed" decision-making in which risk results are one input to a decision, along with other factors such as maintaining defense-in-depth, sufficient safety margins, and performance monitoring.

Today, the risk-informed process, metrics, and guidelines defined in RG 1.174 have been incorporated into numerous licensee and regulatory programs for operating reactors including the Maintenance Rule, Reactor Oversight Process, Technical Specifications, etc. Since the issuance of RG 1.174 and its companion application-specific RGs various derivative metrics have been included in licensee and regulatory programs, e.g., ∆CDF, ∆LERF, incremental conditional core damage probability (ICCDP), and incremental conditional large early release probability (ICLERP). While RG 1.174 allows small risk increases, in practice the majority of regulatory applications by RG 1.174 have been developed and implemented to minimize any risk increase and this has resulted in improved overall safety. The improved safety focus gained through these applications has contributed to an overall reduction in CDFs industry-wide [Ref. 8].

In summary, while the operating reactors have adopted the RG 1.174 approach that uses CDF and LERF, the regulatory documents related to PRAs for Combined Operating Licenses (COLs) [Ref. 9] have continued to rely upon the original risk metrics defined in the original Part 52 design certification process, i.e., CDF, LRF, and CCFP.

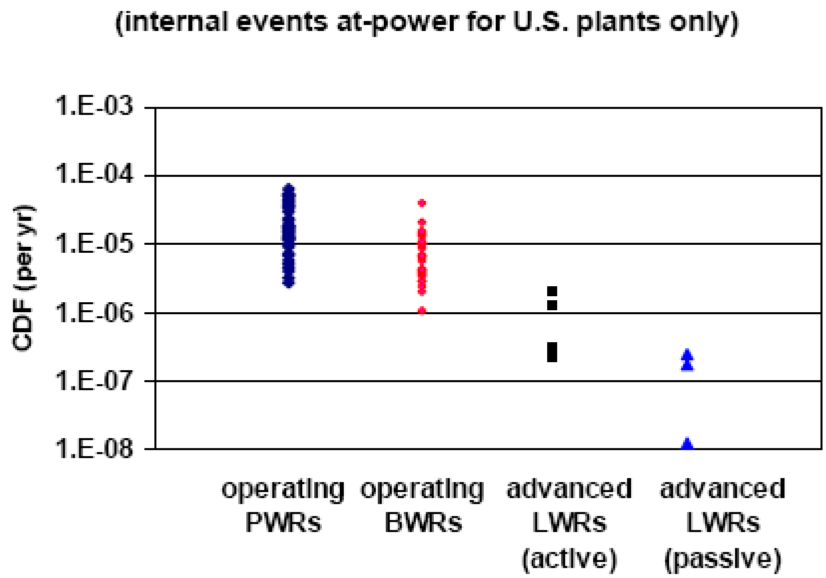
However, as the advanced reactor designs are progressing through the licensing process and appear to be heading toward construction and operation, NRC has identified potential issues involving the risk metrics being applied to an operating reactor and those used in the design and licensing process under Part 52.

Role of PRA and Risk Metrics in Design and Licensing

PRA fundamentally addresses the risks associated with design and operation. The Part 52 design certification and COL process both use extensive PRA insights for all new reactors. PRA has been shown to be an effective tool to inform the design and minimize the potential for beyond design basis vulnerabilities. All of the certified designs have used PRA in this manner.

As the design certification PRAs have demonstrated, the computed risks for the new reactors are lower than comparable operating designs (i.e. PWR versus PWR) when only internal events are considered (Figure 1). Generally, the design certification PRAs are within about an order of magnitude of the lower end of the internal events CDF range for current reactors.

Figure 1
Internal Events Core Damage Frequency Perspectives [Ref. 10]



The design certification PRA scope has generally been limited to internal events, internal flooding, and internal fires for power operations and selected shutdown operational states. Seismic risks have not been quantified, but have been addressed using PRA-based SMA.

Another important part of the severe accident capabilities of plants licensed under Part 52 involves the deterministic requirements for each design to include severe accident mitigation features such as:

- Hydrogen generation and control systems,
- Reactor primary system depressurization systems,
- Ex-vessel core debris cooling capability, and
- Robust containment design to prevent releases within 24 hours.

From a design perspective, the requirements and risk goals of the Part 52 process have put appropriate focus on the following:

- A low and balanced computed CDF,
- A low computed CCFP (<0.1) for the corresponding computed CDF, and
- A low LRF for the corresponding computed CDF.

The use of SMA ensures a pragmatically minimal seismic risk, but excludes the seismic hazard risk from the computed CDF.

Part 52 requirements for severe accident mitigation and the associated risk metrics have been effective in establishing robust designs with low computed risk levels, a computed CDF not dominated by any particular contributors (i.e., balanced risk profile without vulnerabilities), and strong severe accident containment performance. As noted, the computed CDF does not include seismic risk.

Large Release Frequency (LRF)

Each of the design certification submittals and COLs (through reference to the DCD) has been expected to demonstrate that the design supports a LRF goal of < 1×10^{-6}/year for the quantified risk contributors. However, given difficulties in defining "large release" that led to the NRC's decision to abandon the development of a specific definition, each reactor vendor has been left to provide their own definitions for LRF. In all cases, the criteria used would be considered much less than "large" (see Attachment 1 for further discussion of large release definitions). In some cases, the vendors simply stated that they did not define large release, but elected to use a criterion that is "much less than large" [Ref. 11]. In all cases, robust new plant designs, including severe accident mitigation features, have met the LRF goal, despite the use of conservative definitions.

Conditional Containment Failure Probability (CCFP)

The Commission also approved the use of CCFP as a risk metric in the design certification and COL process. Once again, a standard definition has not been established, so each vendor adopted their own approach to computing CCFP. In general, CCFP has been defined as the fraction of the computed CDF that results in LRF (i.e., CCFP = LRF/CDF). All designs have been able to show that CCFP < ~0.1 for the scope of quantified hazards, albeit to different CCFP definitions. Given the conservative definitions of LRF used and the balanced CDF profiles, this has served to confirm the robustness of the plant design and the effectiveness of the severe accident mitigation features required under Part 52.

General Conclusions on the Severe Accident Capability of New Reactors

Consistent with Commission policy, new reactor designs have an enhanced level of severe accident prevention and mitigation capability resulting from a number of common attributes [Ref. 12]:

- High level of redundancy
- Physical separation of safety systems
- Very low contribution from interfacing systems loss of coolant accident (ISLOCA)
- Low contribution to CDF from anticipated transient without scram (ATWS)
- Rapid reactor primary system depressurization capability
- Core melt mitigation capability
- Containment combustible gas control capability

As a result, the certified designs have achieved a higher standard of severe accident safety performance than prior designs and they provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety functions in preventing core damage, containment failure, and large release.

The Role of Risk Metrics in New Reactors Operational Phase

As plants licensed under Part 52 transition into operation, they will fall under the requirements and implementation practices of Part 50 for operating reactors. The two primary risk metrics used in evaluating operating reactors are CDF and LERF. During the 1990s, these metrics were demonstrated to be acceptable surrogates for the QHOs, and were elevated to subsidiary safety objectives as part of the implementation of RG 1.1.74. CDF is generally regarded as a surrogate for the individual latent cancer fatality risk QHO and LERF has been shown to be an adequate surrogate for the individual early fatality risk QHO [Ref. 13].

The origin of LERF as a risk metric was the PSA Applications Guide [Ref. 14] developed by EPRI in the mid-1990s. LERF was proposed by industry as a surrogate for the early fatality QHO, based on the insights from NUREG-1150 [Ref. 15] and other Level 3 PRAs. Such studies have shown that LERF is actually more than an order of magnitude greater than the frequency of a "large release," when a large release is defined as one that can result in one or more early fatalities (as was done in NUREG-1150). It is worth noting that the staff's original proposal to the Commission in SECY 89-102 was similar: "a large release is a release that has the potential for causing an offsite early fatality" [Ref 16]. The reason that a LERF of 10^{-5}/year is roughly comparable to a LRF of 10^{-6}/year is that not all large early releases occur at a time when conditions (e.g., wind direction and speed, stability class, evacuation progress, etc.) would cause an early fatality (See Attachment 1). So, in a sense, LERF is a surrogate for both LRF and the early fatality QHO. Furthermore, the factor of ten (10) difference in the quantitative guidelines for CDF and LERF aligns well with the concept of a CCFP < 0.1. Attachment 1 provides a more comprehensive discussion of the relation between LRF and LERF.

Quantitative Risk Thresholds for Operating New Reactors

The Commission has been consistent in maintaining that new reactors should not be measured against a lower quantitative CDF threshold than operating reactors. When the staff first proposed that new reactors be evaluated using a mean CDF target less than 1.0×10^{-5} event per reactor-year [Ref. 17], the Commission rejected the recommendation [Ref. 18] and reiterated their position from the previous year [Ref. 3], supporting the use of 10^{-4} per year of reactor operation as a core damage frequency goal.

Concerns that tighter risk metrics are needed to prevent the erosion of new plant safety and severe accident performance echo concerns that led to NRC staff proposals in the mid-1990s for a suite of new "applicable regulations" to codify requirements for severe accident features in each design certification rule. The basis for the Commission's rejection of the staff proposal [Ref. 19] is applicable to the current consideration of tighter risk metrics for new plants.

In 2008, in the release of the revised Advanced Reactor Policy Statement, the Commission again made it clear that:

> *… the policy statement does not state that advanced reactor designs must be safer than the current generation of reactors, but rather that they must provide the same degree of protection of the environment and public health and safety and the common defense and security that is required for current-generation light-water reactors. The goal of the policy statement update is to encourage advanced reactor designers to consider safety and security in the early stages of design in order to identify potential design features and/or mitigative measures*

> *that provide a more robust and effective security posture with less reliance on operational programs.* [Ref. 20]

In fact, as discussed above, the Part 52 design certification and COL process ensures that the designs do incorporate risk insights and design features that provide a robust safety capability.

Furthermore, in developing Regulatory Guide 1.174, NRC developed CDF and LERF acceptance guidelines that were derived from the QHOs of the safety goal. Regulatory Guide 1.174 states the following:

> *In theory, one could construct a more generous regulatory framework for consideration of those risk-informed changes that may have the effect of increasing risk to the public. Such a framework would include, of course, assurance of continued adequate protection (that level of protection of the public health and safety that must be reasonably assured regardless of economic cost). But it could also include provision for possible elimination of all measures not needed for adequate protection, which either do not effect a substantial reduction in overall risk or result in continuing costs that are not justified by the safety benefits. Instead, in this regulatory guide, the NRC has chosen a more restrictive policy that would permit only small increases in risk, and then only when it is reasonably assured, among other things, that sufficient defense in depth and sufficient margins are maintained. This policy is adopted because of uncertainties and to account for the fact that safety issues continue to emerge regarding design, construction, and operational matters notwithstanding the maturity of the nuclear power industry. These factors suggest that nuclear power reactors should operate routinely only at a prudent margin above adequate protection. The safety goal subsidiary objectives are used as an example of such a prudent margin.*

Given the Commission's consistent position on expectations for new reactors and the consistency between a LERF guideline of 10^{-5}/reactor year and a LRF of 10^{-6}/reactor year described above, it is clear that the quantitative acceptance guidelines of RG 1.174 could be considered not only appropriate, but consistent with previous Commission policy for new reactors.

Problems with Establishing Alternate Risk Metrics and/or Thresholds

The NRC has identified six risk metric options for consideration [Ref. 10]. Based upon the above discussion, the industry believes the current metrics (Option 1 of the NRC paper) are technically justified and appropriate for all plants, based on reasonable assurance of public health and safety, including operation at a prudent margin above adequate protection. The introduction of new risk metrics or thresholds could create a number of issues:

1. Inconsistency with Commission Policy: Most importantly, as discussed above, the existing risk metrics are consistent with and derived from the NRC Safety Goal Policy Statement which has been long accepted and reiterated in Commission statements such as the cited NRC SRM on SECY 90-16, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," June 26, 1990. In the intervening years there has been no Commission direction to provide alternate requirements or goals. Certain of the proposed NRC options would establish de facto new safety goals or subsidiary objectives without proper revision of the

underlying Policy Statement. The Commission has consistently stated that new reactors should not be measured by more restrictive quantitative risk metrics, including as recently as last year. Many of the issues raised in the staff paper have been previously considered and resolved. Thus, introduction of new or different risk metrics would be contrary to long-standing Commission positions.

2. New Risk Metrics Would Penalize Safer Plants: As described above, new plants have been designed using risk insights. Different risk metrics would create a double standard that would penalize plants for being safer (e.g., limit their operational flexibility and subject them to enforcement at low thresholds), and could lead to allocation of NRC inspection resources at thresholds where there is essentially no impact on public health and safety.

3. New Risk Metrics Would Create Public Perception Problems: Many of the new reactors in the COL process are planned to be on a site with an existing reactor. Having different metrics creates challenges for the NRC inspectors as well as the public in understanding how similar findings at two co-located sites could be considered to have different levels of significance.

4. New Risk Metrics Values Could be Associated with High Uncertainties: For some of the NRC proposed options, the risk metrics could be so low that they would challenge the resolution capability of PRA technology. Decisions would be made on extremely small risk values that are almost unnoticeable within uncertainty bands. Existing metrics already suffer from this problem to a degree, and this problem would be exacerbated with some of the proposed options, especially for the reactor oversight process (ROP) and the significance determination process (SDP) determinations. The imposition of de minimus risk thresholds could also have the unintended consequence of truncating risk-informed activities in new plants that would undermine the observed benefit of a risk-informed focus.

5. Current Risk Metrics are also Supported with Additional Requirements: Risk -informed regulation through RG 1.174 has not led to increased CDF/LERF values for operating plants – in fact, the opposite has been demonstrated. Risk-informed regulation has led to an increased focus on risk significant items and safety performance has improved. This reality contradicts a fundamental premise of the NRC paper: an apparent presumption that the entire risk margin available through RG 1.174 could be consumed. All RG 1.174 applications require advance NRC review and approval, and there is a significant body of practical experience from risk applications at operating plants. RG 1.174 is not solely risk-based and it requires four other regulatory considerations to be addressed, including safety margins and defense-in-depth. In reality, these other considerations are routinely employed by NRC staff to limit or reject the proposed changes even when risk thresholds are met. Additionally, even for current plants, the NRC has rarely granted changes outside of "very small" region of RG 1.174 Figures 3 and 4. In practice this has limited changes by an order of magnitude compared to the "allowable" acceptance guideline. This further decreases the margin between new reactor risk metrics and the quantitative thresholds.

6. New plants s are Subject to a Comprehensive Change Control Process with explicit consideration of severe accidents: New reactors licensed under Part 52 already have a comprehensive change control process with respect to severe accident capabilities. Changes to fundamental plant design or plant Technical Specifications are subject to

prior NRC review and approval. Changes in design or implementation details (so-called Tier 2 information) are subject to an enhanced 50.59-like process that explicitly addresses the potential to increase the likelihood or consequences of severe accidents, and if triggered, would require prior NRC review and approval.

7. Current Risk Metrics Contain Deterministic Backstops: Risk-informed applications generally contain deterministic backstops that protect against very small risk impacts leading to non-conservative operational decisions. For instance, Technical Specification Initiative 4B limits all risk-informed completion times (RICTs) to 30 days maximum, regardless of how small the magnitude of the computed increase in core damage probability is.

8. Risk Profile for New Reactors is Not Yet Complete: The staff's proposed options appear to be based on an incomplete risk picture (i.e., the DCD/COL PRAs). It can be reasonably expected that the DCD risk results will increase as new plants are required through 10 CFR 50.71(h)) to develop PRAs for NRC endorsed consensus standards, which currently would include internal events, fire, and external events including seismic (RG 1.200, Revision 2 was issued in March 2009). The NRC options paper is silent on this aspect and appears to presume the DCD values will carry forward as the required scope of the PRA increases. Given the above, the calculated risk metrics for new reactors are likely to increase and therefore be closer to current plants than being portrayed today. That is, the one to four orders of magnitude difference cited by the staff will decrease as other site-specific risk contributors, such as seismic, are more fully quantified. There already exists a variation in baseline risk values (CDF/LERF) for operating plants, and this was explicitly considered in the development of RG 1.174. Given this, the current approach should remain valid for new plants.

Conclusions

The Part 52 licensing process and Commission policy, quite appropriately, puts an increased emphasis on the severe accident capability of new reactors and the resulting computed risk metrics of CDF, CCFP, and LRF. Consistent with Commission policy, these risk metrics, in combination with other deterministic requirements, have resulted in new reactors having enhanced severe accident safety performance and enhanced margins of safety, as compared to prior designs.

As these new reactor designs transition to the operational phase, it is appropriate to transition the evaluation of severe accident safety to the risk-informed process that was developed by the NRC after the Part 52 process was implemented. Reliance on CDF and LERF provides risk metrics that are philosophically consistent with the Part 52 risk metrics and appropriately aligns the safety metrics of new reactors with the rest of the operating fleet.

The existing RG 1.174 risk metrics, quantitative acceptance guidelines, and integrated decision-making process fits well with the Commission objectives for new reactors. That is, the risk metrics of RG 1.174 and the derivative metrics used in other applications are consistent with the risk metrics used in Part 52 (CDF and LERF as a surrogate for LRF) and the defense-in-depth and safety margin principles will ensure that the robust severe accident design features provide enhanced safety and severe accident protection throughout the operational phase.

References

1) USNRC, “Safety Goals for the Operation of Nuclear Power Plants,” 51 FR 28044; August 4, 1986 as corrected and republished at 51 FR 30028; August 21, 1986.

2) USNRC, 10 CFR 52.47(a)(27).

3) USNRC, SRM on SECY 89-102, “Implementation of Safety Goals,” June 15, 1990.

4) USNRC, SRM on SECY 90-16, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements,” June 26, 1990.

5) USNRC, SECY 93-138, “Recommendation on Large Release Definition,” May 19, 1993 and associated SRM on June 10, 1993.

6) USNRC, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” Federal Register, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.

7) USNRC, “Regulatory Guide 1.174 - An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” July, 1998.

8) EPRI, “Safety and Operational Benefits of Risk-informed Initiatives,” February 2008.

9) USNRC, “Regulatory Guide 1.206 - Combined License Applications for Nuclear Power Plants,” June, 2007.

10) Memo from R.W. Borchart to NRC Commissioners, “Alternative Risk Metrics For New Light-Water Reactor Risk-Informed Applications,” USNRC, February 12, 2009.

11) “ABWR Design Control Document/Tier 2,” Rev. 0.

12) Donald Dube, “Comparison of New Light-Water Reactor Risk Profiles,” ANS PSA 2008 Topical Meeting, Knoxville, Tennessee, September, 2008.

13) USNRC, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860, Appendix D - Derivation of Risk Surrogates for LWRs,” December, 2007.

14) D.E. True, et al, “PSA Applications Guide,” EPRI TR-105396, August, 1995.

15) USNRC, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, December 1990.

16) USNRC, SECY 89-102, “Implementation of Safety Goal Policy,” March 30, 1989.

17) USNRC, SECY 90-16, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” January 12, 1990.

18) USNRC, SRM on SECY 90-16, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” June 26, 1990.

19) USNRC, SRM on SECY 96-077, “Certification of Two Evolutionary Designs,” December 6, 1996.

20) USNRC, “Policy Statement on the Regulation of Advanced Reactors,” 73 FR 60612, October 14, 2008.

Attachment 1

The Relationship of Large Early Release Frequency and Large Release Frequency

-- A Historical Perspective --

Recently, there has been renewed discussion regarding the definition of large release frequency (LRF), spawned by the need for operational decision-making metrics for new reactors licensed under 10 CFR 52. The purpose of this paper is to provide a historical perspective on the development of the risk metrics large early release frequency (LERF) and Large Release Frequency (LRF), and their relationship to each other.

<u>History of "Large Release"</u>

The notion of a risk metric related to "large release" dates back to the issuance in 1986 of the Nuclear Regulatory Commission's (NRC's) Safety Goal Policy Statement (SGPS) [Ref. A1-1]. The Safety Goal Policy Statement (SGPS) provides qualitative and quantitative definitions of "how safe is safe enough" for nuclear power plants. In the SGPS, the Commission directed the staff to develop guidance related to "large releases":

> *"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."*

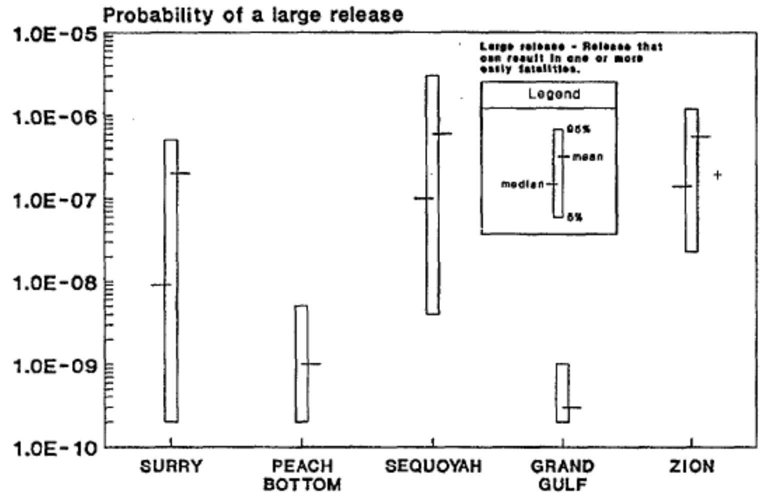
In response to the SGPS direction, the staff proposed a qualitative definition for "large release" in SECY 89-102 [Ref. A1-2]:

> *"A large release is a release that has a potential for causing an offsite early fatality."*

In the resulting staff requirements memorandum (SRM) on SECY 89-102 [Ref. A1-3], the Commission directed the staff to develop a more specific definition and supporting rationale, consistent with criteria provided by the Advisory Committee on Reactor Safeguards (ACRS).

In 1990, the NRC issued NUREG-1150 [Ref. A1-4], a risk reference document containing Level 3 PRAs for five U.S. nuclear power plants. In the figure below, the staff computed the large release frequencies (LRF) for each plant consistent with the staff's original definition, i.e., a release that can result in one or more early fatalities:

Figure 1
NUREG-1150 Large Release Frequencies for Internal Events
(Fig. 12-5 of Ref. A1-6)



In SECY 90-405 [Ref. A1-5], the staff provided two more specific definitions of large release for further investigation that met the criteria provided by the ACRS and Commission. In the SRM on SECY 90-405 [Ref. A1-6], the Commission expressed a preference for a quantitative definition of LRF in terms of an absolute value or equivalent curies released.

In 1993, the staff concluded research on a quantitative definition and sent the Commission SECY 93-138 [Ref. A1-7], which recommended termination of work on a quantitative definition of large release. The staff's conclusion was that "development of a large release definition and magnitude, beyond the simple qualitative statement released to the 10^{-6} per year large release frequency (such as is currently contained in the Commission's Safety Goal Policy Statement), is not practical or required for design or regulatory purposes." In the SRM on SECY 93-138 [Ref. A1-8], the Commission concurred with the Staff's recommendation.

History of "Large Early Release"

The origin of large early release frequency (LERF) ties back to the EPRI PSA Applications Guide [Ref. A1-9], published in 1995. The PSA Applications Guide was the industry's initial effort to define risk metrics and quantitative thresholds for decision making. The industry effort was aware of the NRC's decision to terminate the effort to define large release. It was recognized that CDF alone was not sufficient in demonstrating that the Quantitative Health Objectives (QHOs) of the SGPS were met. Work done on plant-specific PRAs, as well as NUREG-1150, had shown that CDF was a reasonable surrogate for the latent health effect QHO, but another metric was required for the early fatality QHO. The concept of LERF tied to a mechanistic definition of a Level 2 PRA endstate was created by the EPRI team. Rather than explicitly tying the metric to a quantitative definition of the release, the characteristics of the release were used. Specifically, the release had to be both "large" (i.e., a rapid unscrubbed release of fission products) *and* "early" (i.e., before emergency protective actions had been completed).

A few years later when the NRC developed the integrated risk-informed decision making process described in Regulatory Guide 1.174, the NRC also adopted LERF as a risk metric. Work done by the ACRS staff [Ref. A1-10] confirmed the alignment of LERF to the early fatality QHO. More recently, NUREG-1860 [Ref. A1-11] reiterated the validity of LERF as a surrogate for the QHO.

The Relationship of LERF and LRF

In the original development of LERF as a risk metric, EPRI evaluated the relationship between LERF and LRF. Although the quantitative thresholds for LERF are an order of magnitude greater than the original LRF frequency expressed in the Safety Goal Policy Statement, the two metrics are actually consistent. This is best explained via an example.

As shown above in Figure 1, when a Level 3 PRA is available, it is possible to quantify the original qualitative definition of LRF provided by the staff in SECY 89-102. In NUREG-1150, the staff computed LRF as the frequency of a release that can result in one or more early fatalities. The results are compiled below for the internal events based on Figure 1 above:

Table 1
NUREG-1150 Computed Large Release Frequency for Internal Events

Plant	Large Release Frequency (/year)
Peach Bottom	1.00E-09
Surry	2.00E-07
Grand Gulf	3.00E-10
Sequoyah	6.00E-07
Zion	6.00E-07

NUREG-1150 was published before the concept of LERF had been adopted for risk-informed decision-making. However, based on the frequency of the accident progression bins used to define containment status in the Level 2 portion of the NUREG-1150 analysis, it is possible to estimate LERF for each of the plants. Table 2 provides a summary of LERF based on the accident progression bins for each plant based on the applicable figure in NUREG-1150.

Table 2
NUREG-1150 Estimated LERF Results

Table 2.a
Peach Bottom Internal Events (Fig. 4.5 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB >200 psi, Early WWF	No	n/a
VB <200 psi, Early WWF	No	n/a
VB >200 psi, Early DWF	Yes	1.48E-06
VB <200 psi, Early DWF	Yes	7.94E-07
VB, Late WWF	No	n/a
VB, Late DWF	No	n/a
VB, CV	No	n/a
No CF	No	n/a
No VB	No	n/a
No Core Damage	No	n/a
	Total	2.27E-06

Table 2.b
Surry Internal Events (Fig. 3.5 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB, Alpha, Early CF	Yes	1.23E-07
VB > 200 psi, Early CF	Yes	1.64E-07
VB < 200 psi, Early CF	Yes	0.00E+00
VB, BMT, or Late CL	No	n/a
Bypass	Yes	5.00E-06
VB, No CF	No	n/a
No VB	No	n/a
	Total	5.29E-06

Table 2.c
Grand Gulf Internal Events (Fig. 6.4 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB, Early CF, Early SPB, No CS	Yes	6.46E-07
VB, Early CF, Early SPB, CS	No	n/a
VB, Early CF, Late SPB	No	n/a
VB, Early CF, No SPB	No	n/a
VB, Late CF	No	n/a
VB, Venting	No	n/a
VB, No CF	No	n/a
No VB	No	n/a
	Total	6.46E-07

Table 2.d
Sequoyah Internal Events (Fig. 5.4 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB, Early CF (During CD)	Yes	2.79E-07
VB, Alpha, Early CF (at VB)	Yes	1.12E-07
VB > 200 psi, Early CF (at VB)	Yes	1.95E-06
VB < 200 psi, Early CF (at VB)	Yes	1.28E-06
VB, Late CF	No	n/a
VB, BMT, Very Late CF	No	n/a
Bypass	Yes	3.12E-06
VB, No CF	No	n/a
No VB, Early CF (During CD)	No	n/a
No VB	No	n/a
	Total	6.75E-06

Table 2.e
Zion Internal Events (Fig. 7.3 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
Early CF	Yes	4.73E-06
Late CF	No	n/a
Bypass	Yes	2.37E-06
No CF	No	n/a

These estimated LERF results can then be compared relative to the LRF results (Table 3)

Table 3
Comparison of NUREG-1150 LERF and LRF Results

Plant	Estimated LERF (/yr)	Large Release Frequency (/yr)	Conditional Probability of LRF Given LERF
Peach Bottom	2.27E-06	1.0E-09	0.04%
Surry	5.29E-06	2.0E-07	3.8%
Grand Gulf	6.46E-07	3.0E-10	0.05%
Sequoyah	6.75E-06	6.0E-07	8.9%
Zion	7.10E-06	6.0E-07	8.5%

In all cases, the LRF is more than an order of magnitude below the estimated LERF. The reason for this is that not every large early release results in an early fatality due to weather and/or partially implemented emergency protective actions. Thus, the LERF quantitative acceptance guideline of RG 1.174 of 10^{-5}/year is consistent with, or potentially conservative with respect to, a LRF acceptance guideline of 10^{-6}/year.

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