Table 1-2—U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	ADAMS Accession No. ML003707849, January 12, 1990, and the related staff requirements memorandum (SRM), ADAMS Accession No. ML003707885, June 26, 1990.		Entire Ch 19
19.0-SAC-07	**SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," ADAMS Accession No. ML003708021, April 2, 1993, and the related SRM, ADAMS Accession No. ML003708056, July 21, 1993.	Y	6.2.5 Entire Ch 19
19.0-SAC-08	**SECY-96-128**, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708224, June 12, 1996, and the related SRM, ADAMS Accession No. ML003708192, January 15, 1997.	~~N/A-VEN~~Y (Item VII)	~~N/A~~6.2.5
19.0-SAC-09	**SECY-97-044**, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708316, February 18, 1997, and the related SRM, ADAMS Accession No. ML003708232, June 30, 1997.	~~N/A-VEN~~Y (Non-Safety related severe accident heat removal system)	~~N/A~~19.2
	The first five NRC policy statements provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The Commission SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur.		
	For the first aspect of the review, the staff's acceptance criteria consists of a determination that the applicant has adequately demonstrated that the design properly balances preventive and mitigative features and represents a reduction in risk when compared to existing operating plants.		

Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR
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	PRA Insight	Disposition
1	**High level of redundancy and independence for safety systems** The U.S. EPR design incorporates four trains of most safety systems, and provides for significant separation: Four trains of the safety injection systems (LHSI, MHSI, and accumulators). Four trains of emergency feedwater (EFW), supplying four steam generators. Each train has an EFW water storage tank for its suction source. Four safety trains of support systems (cooling trains, building HVAC, and electric power).	6.3 10.4.9.2.1 CT 9.2.2, 9.2.1.2 HVAC 9.4.5 EP 8.1.2
2	**Physical separation of safety systems** In addition to being highly redundant, the four trains of safety systems are physically separated by being located in different S~~s~~afeguard ~~b~~Buildings. This significantly reduces the potential for core-damage accidents due to internal flooding, internal fires, or external events for which spatial considerations are important.	3.8.4, 6.3.2.6
3	**In-containment refueling water storage tank (IRWST)** The design of the IRWST eliminates some failure modes that have been important for current-generation plants: Use of the IRWST eliminates the need to change system alignment by switching suction sources for safety injection following a LOCA. The failure to accomplish this switchover has been an important contributor to failure of long-term safety injection for many current-generation PWRs. Eliminating the need for switchover also obviates the need to isolate the suction path used during the injection phase. For some current-generation PWRs, failure to isolate this path has been assessed to result in inadequate NPSH for the safety injection paths, and may create a release path after the recirculation path is opened. The reactor containment building affords the IRWST better protection against some types of external events than is the case for equivalent tanks at current-generation plants.	6.3.2.2.2

Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR
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PRA Insight		Disposition
4	**High level of redundancy and independence for onsite power supply system** The U.S. EPR design includes both emergency diesel-generators (EDG) and station blackout diesel generators (SBODG) that serve as an alternate AC source. These onsite power sources have the following features: ● There are four EDGs, one supporting each safety division. This provides substantial redundancy to maintain the function of safety systems following a loss of offsite power. ● There are two backup SBODGs ~~diesel generators~~ for AAC. The SBODGs ~~diesel generators~~ are diverse from the EDGs in design, manufacturer, cooling, actuation and control, fuel oil and operating environment. This affords significant defense against potential common-cause failures that might affect all of the diesel generators. ● The SBODGs ~~diesel generators~~ can be aligned to back up two divisions of the safety loads if the EDGs are unavailable, and can be used to support systems provided to mitigate severe-accident conditions.	8.3.1.1.5 8.4.1 8.4.1
5	**Reliability of normal AC power supplies** Among the provisions incorporated into the design of the U.S. EPR to provide for improved reliability of the normal supply of AC power, reducing the demand for emergency power from the diesel-generators, are the following: ● The design includes the capability to withstand a full load rejection without tripping the reactor. In the event of a load rejection, the reactor and turbine would automatically run back to a power level sufficient to allow the main generator to continue to supply the plant auxiliary loads. This design would reduce the potential for reactor trip and challenge to onsite emergency power systems for grid-centered loss of power events. ● During normal operation, two auxiliary transformers supply power directly from the switchyard to all four safety-related switchgear divisions. An additional three transformers supply the non-safety-related switchgear. Since the main generator does not normally supply auxiliary loads in this configuration, a reactor trip does not create a demand for fast transfer to an offsite power source. Moreover, there are redundant feeds for each switchgear (safety-related and non-safety-related), so that loss of an individual auxiliary transformer will not affect the continued supply of offsite power to plant loads.	8.3.1.1 8.2.1.1

Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR
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	PRA Insight	Disposition
6	**Significance of AC power to the core-damage results** Despite the provisions made for the reliable supply of offsite and onsite AC power, the risk results indicate that losses of offsite power are among the dominant contributors to the frequency of core damage. Since the U.S. EPR employs active safety systems that derive their motive power from AC sources, this is to be expected. The CDF remains low because of the level of redundancy and diversity incorporated into the AC systems.	19.1.4.1.2.2
7	**Modest contribution of SLOCA** Small LOCAs are less significant than are losses of offsite power. This is large part due to the four-train redundancy of the safety injection systems. The contribution from SLOCAs is, however, still important on a relative basis, because of the potential for common-cause failures of the systems needed to prevent core damage (e.g., common injection check valves, MHSI and actuation systems).	19.1.4.1.2.2
8	**Provisions to limit the impact of sequences involving failure to scram** The extra borating system (EBS) provides manual injection capability of highly borated water into the reactor pressure vessel (RPV) in the event that the reactor shutdown system does not function properly. EBS is a two-train system which further reduces the potential contribution of accidents involving a failure to scram	6.8
9	**Reduced potential for a small LOCA due to failure of reactor coolant pump (RCP) seals** The potential for RCS leakage or small LOCA (SLOCA) due to failure of reactor coolant pump (RCP) shaft seals has been an important risk contributor for many PWRs. The U.S. EPR design includes a stand still seal for each RCP. The stand still seal is a pneumatic, "metal-to-metal" seal that serves as a back-up seal, and is independent of the normal shaft seal. The stand still seal system reduces the risk of a LOCA event as a result of postulated RCP seal degradation.	5.4.1.2.1
10	**Reduced potential for release pathway following a steam generator tube rupture (SGTR)** ● Among the features of the MHSI system is the provision for a shutoff head below the setpoints for the main steam safety valves (MSSV). In the event of an SGTR, the lower MHSI shutoff head limits the pressure differential that forces reactor coolant through the broken tube. The lower MHSI pressure will not challenge the associated MSSV to open (with possible failure to re-close). This reduces the potential for a release pathway from the RCS through the MSSV.	Table 6.3-3 Table 10.3-2 15.6.3.1.1

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	PRA Insight	Disposition
11	**A state-of-the-art digital instrumentation and control (I&C) system** The U.S. EPR uses state-of-the-art digital systems for I&C functions. The reliability of these systems enhances the automatic initiation of functions important to maintaining core cooling, including the following: ● Reactor shutdown, ● Emergency feedwater, and ● Safety injection	7.1.1.4.1
	The man-machine interface implemented through a fully computerized control room also optimizes the information available to the operators.	7.1.1.1
	Because of the level of redundancy of such systems, concerns regarding the potential for common-cause failures must be addressed. A number of important measures have been taken to limit the potential for CCFs for the digital I&C systems of the U.S. EPR, including the following: ● The Protection System employs subsystems called diversity groups to accomplish essential actuations. These subsystems are functionally diverse and independent. The diversity results from the use of different application programs and different parameter/sensor inputs. No information is shared between diversity groups via network connections. ● The outputs of the protective system (PS) are connected to diverse reactor trip devices. ● The ESF functions are also divided between the diverse subsystems to obtain maximum functional diversity. In addition to the functional diversity provided by the subsystems within the PS and the diversity of the reactor trip devices, there is additional defense-in-depth provided in the I&C architecture. This includes the following:	7.1.1.4.1
	● Trip reduction features of the RCSL and PAS systems, which provide control, surveillance, and limitation functions to reduce reactor trips and PS challenges. Among these features is the automatic power reduction that is not credited in the PRA.	7.1.1.4.5 7.1.1.4.6
	● Backup trip and actuation functions are performed by the non-safety-related I&C system (i.e., the PAS). The potential for software CCFs is minimized by such measures as the following:	7.4.1.1
	● High quality software design tools, ● A deterministic operating system, ● Built in monitoring and testing, and ● Built in functional diversity.	7.1.1.1 7.1.1.2

Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR
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	PRA Insight	Disposition
12	**Diversity of some elements of HVAC** Diversity is incorporated into the design of the safety chilled water system through the use of air cooling for the refrigeration units in Divisions 1 and 4, and CCW cooling for the refrigeration units of Divisions 2 and 3.	9.2.8.2.2
13	**Potential cross-train impact of loss of HVAC** Because of the normal configuration with two trains of CCW in operation, a loss of HVAC for the building in which one CCW operating train is located can have consequences that affect HVAC for the building in which the standby CCW train is located. For example, as the systems are modeled in the PRA, a failure of HVAC with failure to recover cooling for SB 1 has a potential to result in the following effects: • A complete loss of the AC and DC buses in Division 1. • Loss of operating CCW pump Division 1 and failure of CCW common header switchover • Loss of CCW flow for thermal-barrier and motor cooling of RCPs 1 and 2. • Loss of charging pump 1. • Loss of cooling to the safety chillers Division 2 and loss of HVAC in SB 2	9.2.2.2.1 9.4.5 9.4.6 19.1.4.1.1.3
14	**A large, robust containment** The U.S. EPR has a containment that can withstand a variety of challenges, including the following: • The containment has a free volume of about 2.8 x 10^6 ft^3, and a design pressure of 62 psig. This volume and relatively high design pressure provide significant capacity to accommodate the loadings due to a LOCA, a main steam-line break inside containment, or severe-accident phenomena. • The containment is also designed to maintain its integrity when challenged by external forces, including the impact from aircraft and the loadings from seismic events.	6.2.1.1.2 6.2.1.5.3 6.2.1.1.1

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PRA Insight		Disposition
15	**Primary depressurization system (PDS)** The U.S. EPR is equipped with a PDS that goes well beyond the capabilities for depressurization in current-generation PWRs to address the potential for accidents that might progress with the RCS at high pressure. This system is comprised of two trains with four depressurization valves, independent of three pressurizer safety valves, that can provide the following benefits: • The SADVs can be used to provide a bleed path independent of the PSVs to support feed-and-bleed cooling in the event of a total loss of feedwater to the steam generators. This feature of the system further reduces the potential for occurrence of a core-damage accident. • In the event of a severe accident, the primary purpose of the SADVs is to prevent the progression from taking place with the RCS at high pressure. Depressurization of the RCS limits the potential for induced failures of the RCS due to the generation of high-temperature gases. This is of particular interest because it further reduces the potential for induced failure of tubes in the steam generators; such failure could create the possibility of a path for radionuclide release that would bypass the containment boundary. • Depressurization of the RCS also limits the dispersion of core debris to the containment atmosphere, essentially eliminating the possibility of direct containment heating.	19.2.3.3.4 19.2.2.6 19.1.4.2.1.2 19.2.3.3.4
16	**Provisions to control combustible gases** The containment is equipped with passive autocatalytic recombiners. These recombiners prevent the buildup of hydrogen concentration so as to limit the size of any hydrogen deflagration and prevent hydrogen detonation	6.2.5.2.1
17	**Core-melt retention system** A passive device allows water from the IRWST to flood the corium spreading area to remove heat from below the core debris via the cooling water channels. This design limits the potential for core-concrete interactions that could cause pressurization of the containment via the generation of non-condensable gases.	19.2.3.3.3.1

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PRA Insight		Disposition
18	**Severe-accident heat removal system** The severe accident heat removal system (SAHRS) provides a means for removing heat from containment following a severe accident. Features of the SAHRS that play an important role in the Level 2 PRA include the following: ● The system supports passive cooling of the molten core debris. ● The system includes a containment spray mode that enhances scrubbing of fission products from the containment atmosphere. ● The system provides for active recirculation of cooling water for the molten core debris. ● Active elements of the SAHRS rely on the SBO diesel generators, providing a degree of diversity and independence from the safety systems involved in core cooling. In addition to containment heat removal credited in Level 2, the SAHRS is also credited in some Level 1 sequences for cooling IRWST if the heat removal function of LHSI fails. The demands/ challenges to the SAHRS are relatively low in frequency due to the four train reliability of LHSI heat removal and overall low CDF. The SAHRS is a single train, which has a dedicated CCW and ESW cooling capability. The system is manually initiated.	19.2.3.3.3.2
19	**Main steam relief trains for reliable heat removal** Each main steam line is equipped with a MSRT. To provide for both reliable operation and limited potential for spurious operation, each MSRT is equipped with four solenoid valves. The configuration of the solenoid valves is, however, such that two 480 VAC MCCs must be available to support operation of each MSRT. Therefore, if selected pairs of MCCS are lost (e.g., 32BRA and 33BRA), all four MSRTs will fail closed.	10.3.2.2
20	**Sensitivity to human reliability** The Level 1 internal events CDF is sensitive to probabilities for human failure events. The U.S. EPR employs active safety systems, and in unlikely sequences of multiple trains failures, operators are credited to initiate recovery actions (e.g., loss of HVAC recovery, feed and bleed, or fast cooldown function). The HRA is performed under assumptions that the operating procedures and guidelines will be well written and complete. This applies to operator training as well.	18.6

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PRA Insight		Disposition
~~21~~	~~**Nature of the probability distribution for CDF**~~ ~~Uncertainty in the CDF has been quantified explicitly by propagating distributions for the parameters of the basic events that comprise the models for core damage. It is typical for the mean value of such a distribution to be somewhat higher than the point estimate obtained by propagating only the mean values for each parameter. For the U.S. EPR, however,the mean value from the Monte Carlo simulation is significantly larger than the point estimate. This is due "state of knowledge correlation" as defined in the ASME PRA Standards, which is most important for cutsets that contain multiple basic events whose probabilities are based on the same data, in particular when the uncertainty on the parameter value is large. Given redundancy of the U.S. EPR safety trains, such cutsets are common in the EPR PRA model. For example, cutsets with multiple DG failure to run may include up to six basic events with the same data. In this case, in the Monte Carlo sampling approach, the same value is used for each basic event probability, since the "state of knowledge" about the parameter value is the same for each event. This results in a mean value for the joint probability that is larger than the product of the mean values of the event probabilities.~~	
21	**EDGs and SBODGs are assigned to different common-cause groups** This PRA modeling assumption will be confirmed by assuring diversity between EDGs and SBODGs.	8.4.1
22	**Floods caused by a break in a system with very large flooding potential (ESWS or DWS) are assumed to be contained below ground level of the affected buildings (SB or FB)** Bases for this assumption are the following: (1) those systems are automatically isolated if the building sump detects a large flooding event, and (2) expansive time is needed to flood a building up to ground level, so operator isolation is likely to succeed if automatic isolation fails.	3.4.3.1 3.4.3.3 3.4.3.4 3.4.3.5
23	**Isolation of EFW tank leaks or pipe breaks is assumed possible for any break location** Pipe breaks in the EFWS are treated as flooding events with the potential to drain all four EFW tanks. The PRA assumed that the operators would have the ability to manually isolate an EFW pipe break occuring in any of the four SBs with isolation valves in another unaffected SB, and to initiate DWS makeup to the tanks of the intact EFW trains.	3.4.3.4 10.4.9.2.1

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PRA Insight		Disposition
24	**Flooding event would not affect the electrical and I&C rooms of a Safeguard Building** Flood paths are provided in the Safeguard Buildings, so that water from a break anywhere in the building would be stored in the lower elevation of the building. In particular, a flooding event would not affect the electrical and I&C rooms of a Safeguard Building. All electrical/I&C equipment is located above the maximum postulated flood level.	3.4.3.4
25	**Cable separation in the MCR Cable Spreading Area** Due to divisional separation measures in the MCR Cable Spreading Area, a fire in the cable spreading area is assumed to disable only one electrical safety division. Non-safety division cables are also assumed to be separated from the safety divisions.	9.5.1.2.1
26	**The remote shutdown workstation is in a fire and flood area separate from the main control room** Although a main control room fire may defeat manual actuation of equipment from the main control room, it will not affect the automatic functioning of safe shutdown equipment via the PS or manual operation from the remote shutdown station. Sufficient instrumentation and control is provided at the remote shutdown station to bring the plant to safe shutdown conditions in case the control room must be evacuated. There are no differences between the main control room and remote shutdown workstation controls and monitoring that would be expected to affect safety system redundancy and reliability.	3.4.3.4 9.5.1.2.1
27	**MCR & RSS ventilation systems** The main control room has its own ventilation system, and is pressurized. This prevents smoke, hot gases, or fire suppressants originating in areas outside the control room from entering the control room via the ventilation system. The ventilation system for the remote shutdown workstation is independent of the ventilation system for the main control room.	6.4.2.4 9.4.1.3
28	**Seismic margins analysis** The plant level HCLPF is ≥ 1.67 SSE and there are no spatial seismic interaction issues.	COL Item 19.1-6 COL Item 19.1-9
29	**Shutdown management guidelines** The shutdown guidelines as described in the Shutdown Management Guidelines, NUMARC 91-06, are considered when developing the plant specific operations procedures.	13.5.2

Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR
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PRA Insight		Disposition
30	**The low probability that the IRWST suction strainers are plugged during shutdown** The IRWST design (e.g., large, separation between suction lines, debris retaining capability) and plant procedures (e.g., foreign material control) provide reasonable assurance that this probability is low.	6.3.2.2.2
31	**Closing containment hatches and penetrations** The ability to close containment hatches and penetrations in accordance with procedures and training during Modes 5 and 6 prior to steaming is important.	13.5.2 COL Item 19.1-9
32	**Low pressure reducing station auto isolation** In shutdown isolation, low pressure reducing station auto isolation on low loop level is important to prevent possible RCS flow diversion through CVCS.	9.3.4.2.2
33	**Automatic level control at mid-loop** Automatic level control at mid-loop is important to reduce the likelihood of RHR pump cavitation.	5.4.7.2.1
34	**In-containment refueling water storage tank/SD** As stated in the Insight #3, the design of the IRWST eliminates some failure modes that have been important for current-generation plants: in shutdown operation IRWST inside containment reduces impacts of RHR flow diversions which lead to LOCAs inside containment, not outside.	6.3.2.2.2
35	**RHR auto isolation on Safeguards Building sump level** In shutdown operation, RHR auto isolation and pump shutoff on a high Safeguards Building sump level, divisionally based, is an important protection from RHR LOCAs outside containment.	5.4.7.2.1
36	**Instrumentation through RPV top head** The U.S. EPR location of the RPV instrumentation, which is through the top head rather than the lower head, reduces the likelihood of a LOCA during maintenance.	5.3.3.1.1
37	**Automatic MHSI actuation** In shutdown operation, automatic MHSI actuation on a low RCS (hot leg) loop level or on a low dPsat (for cold shutdown) is important to mitigate losses of RHR, LOCAs and flow diversions.	5.4.7.2

Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR
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PRA Insight		Disposition
38	**Sensitivity to human reliability in shutdown** Similarly to the Insight #20, the shutdown CDF is sensitive to probabilities for human failure events. Important human actions in shutdown are operator isolation of various flow diversions, operator actions to control draindown in mid loop and operator manual actuations of RHR/LHSI pumps. The PRA assumed that instrumentation to support these actions is available (e.g., loop level and sump level indications and alarms) and that the written procedures covering these actions are implemented and maintained.	18.6
39	**An alternate decay heat removal path** An alternate decay heat removal path in shutdown can be established by operator action to manually open PSV valves or primary depressurization valves and to initiate MHSI/LHSI injection.	5.2.2
40	**Physical separation of safety systems/SD** As stated in the Insight #2, complete physical separation of the U.S. EPR safety systems significantly reduces the potential for core-damage accidents due to internal or external hazards in shutdown. It is assumed that this separation also makes it possible to implement controls during maintenance in shutdown to protect operating trains. The PRA assumed that written procedures to cover Fire Protection Program are implemented and maintained.	5.4.7.2 9.5.1.6

EPRI HRA Calculator incorporates the SPAR–H worksheet, which is a major component of the SPAR-H method, and the SPAR-H dependency rating system. Validation of proper installation and execution of the code is performed.

The EPRI HRA Calculator development is directed by the EPRI HRA/PRA tools Users Group. Membership currently includes 19 utilities comprising more than 60 nuclear power plants in the U.S. and one international member (the CANDU Owners Group).

19.1.4.1.2 Results from the Level 1 PRA for Operations at Power

19.1.4.1.2.1 Risk Metrics

Total CDF from internal events is 2.8E-07/yr, less than 1E-06/yr. This is well below the NRC goal of 1E-04/yr (SECY-90-016, Reference 30) and the U.S. EPR probabilistic design goal of 1E-05/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.4.1.2.7.

19.1.4.1.2.2 Significant Initiating Events

The significant initiating events and their contribution to the internal CDF are given in Table 19.1-6—U.S. EPR Significant Initiating Event Contributions – Level 1 Internal Events. Only those initiating events that contribute more than one percent to the total internal events CDF are listed in the table. All initiating events and their contributions are illustrated in Figure 19.1-4—U.S. EPR Initiating Events Contribution - Level 1 Internal Events. As can be seen from Table 19.1-6 and Figure 19.1-4, the LOOP initiating event strongly dominates the internal events CDF (close to 50 percent). This is not a surprise because the U.S. EPR is an active plant with no passive systems. In order to illustrate in more detail the total LOOP contribution to CDF, the LOOP sequences were divided into four categories.

- LOOP events (no seal LOCA, no SBO) contribute 30 percent to the total CDF.
- LOOP events leading to seal LOCA contribute 5 percent to the total CDF.
- LOOP events leading to SBO conditions contribute close to 10 percent to the total CDF
- LOOP events leading to seal LOCA and SBO conditions contribute close to 5 percent to the total CDF.

The next biggest contributors to plant risk are SLOCA and general transient.

- SLOCA contribution can be attributed to a larger range in the break sizes and the corresponding higher frequency of SLOCA, and to common injection system failures (signals or common injection check valves).

A very conservative sensitivity case was evaluated to estimate combined effects of different assumptions; many assumptions with the worst effect were combined as presented in the table. The overall result is an increase by approximately 15 times in the CDF to 5E-06/yr, still well below the NRC goal of 1E-04/yr.

The CDF results were not sensitive to the assumption on mission time for long term cooling, or on the assumptions about isolation of the EFW tanks leaks.

A simple sensitivity analysis (not reported in Table 19.1-15) was performed for the ISLOCA events, using mean values for the ISLOCA IE frequencies, versus point estimates. Since ISLOCA event contribution to the CDF is negligible, the effect of this change on the CDF was also negligible (less than one percent).

Table 19.1-15 shows only moderate improvements in CDF if some design changes are considered, or less conservative assumptions are made. The one design change which may be considered in the future (~~16~~ 7 percent improvement) is to realign MSRIVs so that they would not require two electrical divisions for their operation.

19.1.4.1.2.7 Uncertainty Analysis

Uncertainty on the Level 1 Internal Events PRA results is quantified using the built-in uncertainty analysis capabilities of Risk Spectrum. ~~This PRA uncertainty quantification evaluates parametric uncertainty. Modeling uncertainty is addressed with limited scope in a separate uncertainty evaluation.~~The results are shown in Figure 19.1-5—U.S. EPR Level 1 Events Uncertainty Analysis Results – Cumulative Distributions for Internal Events CDF. Two distributions are presented, one that only incorporates parametric uncertainty and one that incorporates three cases of modeling uncertainty. The results of parametric uncertainty are summarized below:

- CDF Internal Events Mean Value: 4.2E-07/yr.
- CDF Internal Events 5% Value: 3.1E-08/yr.
- CDF Internal Events 95% Value: 1.2E-06/yr.

This ninety-fifth percentile CDF value is more than an order of magnitude below the NRC goal of 1E-04/yr.

As can be seen from the results for parametric uncertainty, the mean value from Monte Carlo simulation is larger than the point estimate. This is due to the “state of knowledge correlation” as defined in the ASME PRA Standards, which is most important for cutsets that contain multiple basic events whose probabilities are based on the same data, particularly when the uncertainty of the parameter value is large. Given the redundancy of the U.S. EPR safety trains, such cutsets are expected in the U.S. EPR PRA model. In this case, in the Monte Carlo sampling approach, the same value is used for each basic event probability, since the “state of knowledge” about the parameter value is the same for each event. This results in a mean value for the joint probability that is larger than the product of the mean values of the event probabilities.

Importance of the redundant equipment and the state-of-knowledge dependencies is limited for the equipment where common cause failures dominate the results. The impact of the redundant equipment is more important in the case where equipment single failures are also significant contributors to the results, like in the cases of the diesel generators. In this evaluation a state-of-knowledge correlation between EDGs and SBODGs was not considered because they belong to the different common cause (different vendors, locations, cooling and starting systems, fuel supplies).

More detailed discussion on parametric and modeling uncertainty is as follows:

Parametric uncertainty was quantified by selecting an uncertainty distribution for each input parameter. Distributions mostly applied are Lognormal, Beta and Gamma, as described below for each type of parameter:

- Initiating Events: Uncertainty distributions were obtained from the same source as the mean values. For initiating events evaluated by fault trees, lognormal distribution was fit to the uncertainty distribution obtained from the RS run. Exceptions are IE frequencies for flooding and fire events, which are based on limited information, and, for their modeling, a constrained non-informative distribution (CNI) was used. This will be discussed in the corresponding sections for internal fire and floods.
- Failure Rates: Uncertainty distributions were obtained from the used data source.
- Digital I&C Failure Rates: Lognormal distribution was used, an error factor of five was estimated from upper & lower confidence bounds in TXS documentation.
- Common Cause Parameters: Uncertainty parameters were obtained from the same source as CC factors. They were fit to lognormal distribution and only applied to the “beta” factor.
- LOOP Related Basic Events: Gamma distribution for LOOP frequency, with upper and lower bounds, was fit to various LOOP events (consequential LOOPs and LOOP in 24 hours).
- Human Error Probabilities: For pre-accident HEPs, a lognormal distribution with an error factor of 10 was used, as recommended in the ASEP method. For post-accident HEPs, a constrained non-informative prior (Beta) distribution was used, as recommended in the SPAR-H method.
- Various Parameters & Undeveloped Events: Constrained non-informative prior (Beta) distribution was used, to account for the limited state of knowledge.

- Time Related Parameters: For time-related parameters, like preventive maintenance duration (and corresponding unavailability), lognormal distribution was used, an error factor was estimated from upper and lower bounds, corresponding to upper and lower time estimates.

Modeling uncertainty was also specifically treated, but limited to three cases selected to illustrate a specific lack of modeling designs details. These cases are described below:

- CASE 1: This case is based on the uncertainty of success criteria for the number of EFW trains required to cool the plant through MSSVs. The considered spectrum of success criteria included (1) one, (2) two or (3) three out of four EFW pumps required. Each of the inputs was combined with the estimated probability of that particular success criterion. This uncertainty is modeled because in a design phase, the pump flow curve is not final.

- CASE 2: This case is based on the uncertainty of success criteria for the number of pressurizer safety valves required for a success of feed and bleed. The considered spectrum of success criteria included (1) one, (2) two or (3) three out of three required. Each of the inputs was combined with the estimated probability of that particular success criterion. This uncertainty is modeled because in a design phase, conservative assumptions are made on PSVs “bleeding” capabilities.

- CASE 3: This case is based on the uncertainty of success criteria for recovery of HVAC to SBs: electrical equipment & EFW pump rooms. The considered spectrum of success criteria included: (1) Loss of HVAC will not disable equipment, (2) Operator recovery is required in 4 hours, (3) Operator recovery is required in 2 hours, or (4) Operator recovery is not possible. This uncertainty is modeled because in a design phase, not enough information is available to predict room heat-up rates and equipment survivability.

~~The results of the uncertainty analysis for internal events are shown in Figure 19.1-5—U.S. EPR Level 1 Events Uncertainty Analysis Results—Cumulative Distributions for Internal Events CDF. Three distributions are presented, one that only incorporates parametric uncertainty; one that incorporates three cases of modeling uncertainty, and the one that incorporates parametric uncertainty, but w/o distribution on DG failures to run (a point estimate is used only). The reason that this curve is presented is discussed next:~~

~~As can be seen from the results for parametric uncertainty, the mean value from the Monte Carlo simulation is significantly larger than the point estimate. This is due to the “state of knowledge correlation” as defined in the ASME PRA Standards, which is most important for cutsets that contain multiple basic events whose probabilities are based on the same data, particularly when the uncertainty of the parameter value is large. Given the redundancy of the U.S. EPR safety trains, such cutsets are common in the U.S. EPR PRA model. For example, cutsets with multiple DG failures to run may include up to six basic events with the same data. In this case, in the Monte Carlo~~

~~sampling approach, the same value is used for each basic event probability, since the "state of knowledge" about the parameter value is the same for each event. This results in a mean value for the joint probability that is larger than the product of the mean values of the event probabilities. To illustrate this, a Monte Carlo simulation is performed after eliminating uncertainty connected with DG failure rates to run (a point estimate is used). As illustrated in Figure 19.1-5, this assumption had significantly reduced the mean value which is now very close to the point estimate value. It is an interesting conclusion: redundancy in the important safety trains reduces overall risk, but increases degrees of uncertainty.~~

19.1.4.1.2.8 PRA Insights

The U.S. EPR is an active plant, thus CDF is dominated by LOOP-related events (approximately 50 percent). Still, total LOOP CDF is small at <1.5E-07/yr. This small contribution is a result of the U.S. EPR high redundancy in trains and diversity in emergency power supplies.

Loss of cooling trains (CCW/ESW) and seal-LOCA contributions to CDF are less than 10 percent. This relatively small contribution is a result of the U.S. EPR redundancy in the cooling trains and the SSSS design, which contributes to RCP seal reliability.

The top cutsets show that the plant risk is strongly influenced by the performance of support systems–HVAC and electrical. This is because the support systems reflect important dependencies between highly redundant safety systems. These dependencies are discussed in this report, and the most important are summarized below:

- A total loss of an electrical division which supplies running CCW pump, could, without operator intervention, disable the second division through a loss of HVAC.
- Loss of two electrical divisions, combinations1 & 3, 1 & 4, 2 & 3, or 2 & 4, would disable MSRTS.
- Loss of Division 1 or Division 4 would disable the primary bleed function, a switchover of the CVCS to the IRWST suction, and the SAHRS.

Sensitivity studies did not identify any events where a design change would lead to a significant reduction in the CDF.

Even though Level 1 PRA analysis (at-power, internal events) identifies some hidden dependencies, it shows no outliers and confirms the robustness of the U.S. EPR design.

- Melt stabilization.
- Containment heat removal.
- Monitoring activity distribution within the containment and potential releases to the environment.

The review of equipment survivability is documented in Table 19.1-23—Evaluation of Equipment Survivability for Level 2.

The following headers in the CET were also reviewed, but are not relevant for equipment survivability:

- No induced hot leg rupture.
- RCS pressure remains high in small LOCA sequences.
- No reactor pit damage due to lower head failure due to in-vessel steam explosion.
- Reactor pit not damaged by ex-vessel steam explosion.

The review of the CET and assessment of equipment credited in light of plans for equipment qualification for severe accidents has concluded that, with the exception of the hydrogen recombiners, none of the equipment credited in the CET models should be considered affected by the severe accident conditions expected to occur during the progression through the Level 2 CET. Consequential damage to the recombiners due to accelerated flame phenomena is considered in the CET model.

19.1.4.2.2 Results from the Level 2 PRA for Operations at Power

19.1.4.2.2.1 Risk Metrics (LRF, CCFP)

Total LRF from internal events is 2.2E-08/yr. This is well below the NRC goal and U.S. EPR probabilistic design goal of 1E-06/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.4.2.2.7.

The CCFP from all internal events (at power) large release sequences is 0.076. This meets the NRC goal of less than approximately 0.1 CCFP.

19.1.4.2.2.2 Internal Events Core Damage Release Category Results

The Release Categories and their contribution to the internal events LRF and the associated CCFP are shown in Table 19.1-24—Internal Events Release Category Results – Large Release Frequency.

Approximately 66 percent of the LRF for internal events is from Release Category RC304. This Release Category represents containment failure before vessel failure with no MCCI occurring and with unavailability of the SAHRS spray for fission

19.1.4.2.2.7 Uncertainty Analysis

~~An integrated uncertainty analysis was performed for the U.S. EPR Level 1 and Level 2 PRAs.~~The results of the uncertainty evaluation for the Level 2 Internal Events LRF are presented in Figure 19.1-9—Level 2 Internal Events Uncertainty – Cumulative Distribution for LRF.

The uncertainty results are summarized below:

- LRF Internal Events Mean Value: 3.1E-08/yr.
- LRF Internal Events 5% Value: 5.8E-10/yr.
- LRF Internal Events 95% Value: 9.0E-08/yr.

This ninety-fifth percentile LRF value is more than an order of magnitude below the NRC goal of 1E-06/yr.

The basis for the input uncertainty distributions for systems related basic events and operator actions is discussed in Section 19.1.4.1.2.7.

For quantitative evaluation of the overall uncertainty on the LRF, discrete distributions were added for the Level 2 phenomenological basic events. These events are identified in the PRA database by use of the prefix “L2PH”. The distribution form chosen for these basic events is double delta. Thus, a probability is assigned for each of two deterministic outcomes for this type of basic event: there is a probability that the event is sure to occur (relative frequency of one) and another that it is sure not to occur (relative frequency of zero). As discussed in Section 19.1.4.2.2.6, this is an appropriate paradigm for such events, since, generally, it is the case that they do not represent random occurrences. Rather they represent events that are expected to have deterministic, but unknown, outcomes. For each event, the probability of the “sure occurrence” outcome is, therefore, equal to the mean value of the basic events.

~~The results of the uncertainty evaluation for the LRF are presented in Figure 19.1-9—Level 2 Internal Events Uncertainty – Cumulative Distribution Function for LRF. The fifth percentile for the LRF is 1.3E-9/yr. The ninety-fifth percentile is 9.7E-8/yr. This ninety-fifth percentile LRF value is more than an order of magnitude below the goal of 1E-6/yr.~~

19.1.4.2.2.8 PRA Insights

The key insights from the Level 2 PRA for internal events are discussed below.

First, it is noted that the LRF is dominated by sequences entering from the Level 1 which represent a severe challenge to the containment or in which the containment function is already defeated (bypassed). These sequences are those discussed in Section

5. Unisolated flooding is contained inside the RB annulus and reaches the level of the electrical penetrations to the containment.

Rough estimates are used to assign probabilities of doors failing under a water pressure. If propagation occurs, the safety systems in the adjacent building are considered failed. If the flood is not isolated and it is contained in the annulus, the water level is assumed to reach containment penetrations. Control and power cables pass through the annulus in air-tight conduits. They enter the containment through the connection boxes, whose ability to withstand the effects of flooding is not known. In this evaluation, given that no specific information is available, it was conservatively estimated that, if flooded, the connection boxes to the containment would fail with a probability of 0.5. If the connection boxes fail, it was also assumed that connection with the containment, including all instrumentation, is lost and core damage is assumed.

Flooding scenarios are quantified using the same fault tree and event tree logic used in the Level 1 internal events evaluation. Mitigating systems that are assumed to be unavailable in a flooding scenario are disabled in the fault tree for this specific scenario.

19.1.5.2.2 Results of Internal Flooding Evaluation

19.1.5.2.2.1 Risk Metrics

The total CDF from internal flooding events is 6.1E-08/yr, less than 1E-07/yr. This is well below the NRC goal of 1E-04/yr (SECY-90-016, Reference 30) and the U.S. EPR probabilistic design goal of 1E-05/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.5.2.2.7.

19.1.5.2.2.2 Significant Initiating Events

All flooding initiating events modeled (flooding scenarios) and their contribution to the internal flooding CDF are given in Table 19.1-40—U.S. EPR Initiating Events Contributions – Level 1 Internal Flooding. Flooding initiating events and their contributions are illustrated in Figure 19.1-12—U.S. EPR Initiating Event Contributions – Level 1 Flooding. As can be seen from Table 19.1-40 and Figure 19.1-12, the flood contained in the annulus dominates the internal flooding CDF. Although this scenario has a low frequency, it is conservatively modeled as directly resulting in core damage if the connection boxes to the containment fail as a result of the flood.

The next biggest contributor to the flooding risk is a flood in SB 1 or SB 4 that extends to the FB. This flood is divided into two categories: floods caused by a break in the emergency feedwater system (EFWS) (the third largest contributor) and floods caused by a break in any other system (the second largest contributor). The reason for this which a flood affects simultaneously SB 1 or SB 4 and the FB, disabling both CCW CH 1 or 2 and the CVCS directly leading to a loss of seal cooling. The internal flooding CDF is not sensitive to the probability of the CVCS requiring a switchover to IRWST. This can be explained by the fact that the CVCS is directly failed by a flood extending to the FB.

The importance of seal LOCA sequences could also be attributed to a conservative assumption of not crediting a recent design change that allows a crosstie of the RCP thermal barrier cooling to different CCW common headers.

The impact on the CDF of the assumptions specific for the flooding events modeling is also studied. The assumption on the isolation of an EFW shows only a mild impact on the flooding CDF, because the failure of isolation and make-up to the EFWS is dominated by the probability of a consequential LOOP, which would disable the make-up option.

19.1.5.2.2.7 Uncertainty Analysis

The results of the uncertainty evaluation for the Level 1 Flooding Events CDF are presented in Figure 19.1-13—U.S. EPR Internal Flooding Events Uncertainty Analysis Results – Cumulative Distribution for Flood Events CDF.

The uncertainty results are summarized below:

- CDF Internal Flooding Events Mean Value: 6.1E-08/yr.
- CDF Internal Flooding Events 5% Value: 3.1E-09/yr.
- CDF Internal Flooding Events 95% Value: 2.2E-07/yr.

This ninety-fifth percentile CDF value is more than two orders of magnitude below the NRC goal of 1E-04/yr.

Uncertainty on the Level 1 Flooding PRA results is quantified using a process similar to that described for the internal events in Section 19.1.4.1.2.~~6~~7. Parametric uncertainty was represented by selecting an uncertainty distribution for each parameter type~~,~~ including flooding initiating events, as described in Section 19.1.4.1.2.~~6~~7. ~~Modeling uncertainty was represented with limited scope by adding uncertainty to the success criteria of EFW pumps and primary relief valves, and by adding uncertainty to the times to overheat for electrical equipment on a loss of HVAC, and to the effectiveness of creative alternate cooling means. These modeling uncertainties are described in detail in Section 19.1.4.1.2.6.~~

~~The results of the uncertainty analysis for flooding events are shown in Figure 19.1-13—U.S. EPR Internal Flooding Events Uncertainty Analysis Results—Cumulative Distribution Function for CDF. Two distributions are presented, one that~~

~~only incorporates parametric uncertainty; and one that incorporates the three cases of modeling uncertainty in addition to the parametric uncertainty. The inclusion of modeling uncertainties increases the point estimate by about a factor of two and reduces the ratio of the mean to the point estimate, indicating a reduction in the impact from "state-of-knowledge" correlation.~~

19.1.5.2.2.8 PRA Insights

The largest contributor to the flooding CDF is the flood in the annulus. It accounts for 50 percent of the overall flooding CDF. This high contribution to the plant risk highlights a vulnerability of annulus pipe break events. It is also the result of conservative assumptions made due to the lack of a detailed design of the annulus electrical penetrations.

Flooding in the SB 1 or SB 4 is dominated by the seal LOCA scenarios, because this flood causes a complete loss of seal cooling to two of the RCPs, and a single failure in the isolation of the RCP seals results in a seal LOCA with a probability estimated to be 0.2. Seal LOCA sequences contribute to more than 30 percent of the flooding events CDF. This corresponds to 60 percent of the risk from sequences other than the flood in the annulus.

Dependencies between support systems also play a significant part in the internal flooding CDF. The sequences where systems fail on total or partial loss of the HVAC represent about 12 percent of the flooding events CDF. This corresponds to 24 percent of the risk from sequences other than the flood in the annulus.

The flood due to the EFWS pipe break has a relatively low contribution to CDF because of a low pipe-break frequency, but has a relatively high conditional core damage probability due to the potential drainage of all EFWS tanks.

Even though several conservative assumptions were made in the analysis, the total risk from flooding events is low with a CDF of less than 1E-07/yr. This illustrates the robustness of the U.S. EPR design and the good spatial separation of the safety trains.

19.1.5.2.3 Level 2 Risk Metrics for Flooding Events (LRF and CCFP)

Total LRF from internal flooding events is 1.1E-09/yr. This is well below the NRC goal and U.S. EPR probabilistic design goal of 1E-06/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.5.2.3.6.

The CCFP from all flooding (at power) large release sequences is approximately 0.018. This meets the NRC goal of less than approximately 0.1 CCFP.

The observations made in Section 19.1.4.2.2.6 (internal events) regarding flame acceleration and in-vessel steam explosion are also relevant in the case of floods. The deflagration events were evaluated as being close to a physically unreasonable probability level, even with the use of some conservatism in the modeling. The U.S. EPR Level 2 analysis assessed in-vessel steam explosion causing containment failure as a very low probability event, but not of sufficiently low probability for it to be removed from the model. Sensitivity to steam explosions arises because, if not excluded from the model, these events are applicable to a large proportion of core damage sequences.

Thermally-induced steam generator sequences play a significant role in LRF for flood events. Flood sequences involving seal LOCAs are significant LRF contributors. 38 percent of LRF involves consequential seal LOCAs from flooding events and 29 percent of LRF also involves a depressurized secondary side of the steam generators. These proportions slightly exceed the corresponding contributions of these sequences to CDF (seal LOCAs contribute 26 percent of CDF, and 13 percent of CDF involves seal LOCAs with a depressurized secondary). In view of this information, sensitivity studies were undertaken to study the factors influencing the induced SGTR contribution to LRF for floods. The sensitivity to manual depressurization and availability of feedwater was therefore studied. It was found that, for the case of flood events, neither the unavailability of primary depressurization nor the unavailability of feedwater individually had a large impact on the frequency of RC702. However, the combined impact of both being unavailable had a significant impact on both the RC702 frequency and on LRF. The sensitivity study with combined unavailability of depressurization and feedwater suggested a five times increase in LRF.

19.1.5.2.3.6 Uncertainty Analysis

The results of the uncertainty evaluation for the Level 2 Flooding Events LRF are presented in Figure 19.1-14 —U.S. EPR Level 2 Flooding Events Uncertainty Analysis Results – Cumulative Distribution for CDF.

The uncertainty results are summarized below:

- LRF Internal Flooding Events Mean Value: 1.2E-09/yr.
- LRF Internal Flooding Events 5% Value: 1.0E-12/yr.
- LRF Internal Flooding Events 95% Value: 1.2E-09/yr.

This ninety-fifth percentile LRF value is more than two orders of magnitude below the NRC goal of 1E-06/yr.

~~An integrated uncertainty analysis was performed for the U.S. EPR Level 1 and Level 2 PRAs.~~

The basis for the input uncertainty distributions for systems related basic events and operator actions is discussed in the sub-sections related to the Level 1 PRA. As discussed in Section 19.1.4.2.2.7, for quantitative evaluation of the overall uncertainty on the LRF, discrete distributions were added for the Level 2 phenomenological basic events. These events are identified in the PRA database by use of the prefix “L2PH”. The distribution form chosen for these basic events is discussed in Section 19.1.4.2.2.7.

~~The results of the uncertainty evaluation for the LRF for floods are presented in Figure 19.1-14—U.S. EPR Level 2 Flooding Events Uncertainty Analysis Results—Cumulative Distribution Function for CDF. The fifth percentile for the LRF is 1.5E-10/yr. The ninety-fifth percentile is 3.4E-9/yr. This ninety-fifth percentile LRF value is more than two orders of magnitude below the goal of 1E-6/yr~~

19.1.5.2.3.7 PRA Insights

As also discussed in Section 19.1.4.2.2.4 for internal events, sequences involving containment failure due to loads from an accelerated flame originating in the lower, middle or upper equipment rooms prior to vessel failure are visible contributors to LRF, the specific contribution being 75 percent in the case of internal floods. The key features of the analysis of accelerated flames and their impact on containment are discussed in Section 19.1.4.2.2.4 and not repeated here.

In the absence of the specific challenges and bypasses of containment seen in the internal events analysis, the results for LRF for flooding events are dominated by severe accident phenomenological issues. The specific issue for floods is the possibility of an accelerated flame arising from hydrogen combustion in the lower or middle equipment rooms during the in-vessel phase of a high pressure core melt. Further background discussion on the analysis of this issue is provided in Section 19.1.4.2.2.4.

Incoming sequences from the Level 1 feature flood-induced seal LOCAs in conjunction with a depressurized secondary side. The phenomena of thermally-induced steam generator tube rupture, which was assessed as having a large probability for equivalent two-inch LOCAs (seal or otherwise) with a depressurized secondary side and an absence of feedwater to the steam generators (therefore also features in the results approximately 15 percent contribution to LRF) but is not dominant. The contribution of this phenomenon is discussed in Section 19.1.5.2.2.3. Sensitivity studies showed a significant increase in LRF due to this phenomena only in the bounding case of assumed concurrent unavailability of feedwater and depressurization functions; individual unavailabilities were not significant.

The importance results for floods show only one operator action from the Level 2 model as contributing. This action is the operator manual backup for containment isolation. LRF shows sensitivity to this action based on its RAW.

be credited even if a consequential LOOP occurs. Manual suppression is credited only in the MCR because it is constantly manned.

Fire scenarios are quantified using the same fault tree and event tree logic used in the Level 1 internal events evaluation. Mitigating systems that are assumed to be unavailable in a fire scenario are not credited. A different value was used for consequential LOOP for fire events leading to a controlled shutdown. The value is estimated based on the value for the consequential LOOP leading to auto scram, reduced by a factor of five. The reduction is based on an estimate that 20 percent of fire initiators leading to a controlled shutdown may result in an automatic plant trip. The fifteen fire scenarios selected in the internal fires PRA are defined in Table 19.1-64. This table gives the fire scenario identifier and description, summarizes the effects the scenario has on mitigating systems, defines the suppression credited, and gives the scenario frequency and basis for that frequency.

19.1.5.3.2 Results from the Internal Fire Risk Evaluation

19.1.5.3.2.1 Risk Metrics

The total CDF from internal fire events is 1.8E-07/yr, less than 1E-06/yr. This is well below the NRC goal of 1E-04/yr (SECY-90-016, Reference 30) and the U.S. EPR probabilistic design goal of 1E-05/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.5.3.2.7.

19.1.5.3.2.2 Significant Initiating Events

All fire scenarios/initiating events modeled and their contribution to the internal fire CDF are given in Table 19.1-65—U.S. EPR Initiating Event Contributions – Level 1 U.S. EPR Important Cutset Groups – Level 1 Fire. Fire initiating events and their contributions are illustrated in Figure 19.1-15. As can be seen from Table 19.1-65 and Figure 19.1-15, 10 out of 15 fire initiating events contribute less than one percent of the internal fire CDF. The fire in the AC switchgear room of SB 1 or SB 4 is the single largest contributor. This could be explained by the importance of electrical Divisions 1 and 4 for the supply of front-line and support systems, as explained in the discussion of system dependencies in Section 19.1.4.1.1.3.

The next two biggest contributors to fire risk are the fire in the MSS/MFWS valve room and the fire in the MCR. The valve room contribution results largely from a specific fire-induced sequence that combines spurious operation of an MSRT and the inability to close two MSIVs (see Section 19.1.5.3.2.3). The MCR contribution includes the failure of the operator action to transfer to the RSS following a fire in the MCR. Although this failure probability is low, it is assumed to directly result in core damage.

a high importance value in the internal fire risk, because of the high occurrence of seal LOCA sequences among the dominant fire scenarios. For the same reason, an assumption on the probability that CVCS switchover to the IRWST may be required also has a high importance value in the internal fire risk.

The importance of seal LOCA sequences could also be attributed to a conservative assumption of not crediting a recent design change that allows a crosstie of the RCP thermal barrier cooling to different CCW common headers.

The impact on the CDF of the assumptions specific for the fire events modeling is also analyzed. The fire CDF is found to be sensitive to an assumption of a fire affecting both an MSRT and an MSIV. The modeling assumption on a complete separation of the safety and non-safety divisions in the CSR is also found to have a high impact on the fire CDF.

19.1.5.3.2.7 Uncertainty Analysis

The results of the uncertainty evaluation for the Level 1 Fire Events CDF are presented in Figure 19.1-18—U.S. EPR Level 1 Internal Fire Events Uncertainty Analysis Results – Cumulative Distribution for Fire Events CDF.

The uncertainty results are summarized below:

- CDF Internal Fire Events Mean Value: 2.1E-07/yr.
- CDF Internal Fire Events 5% Value: 9.5E-09/yr.
- CDF Internal Fire Events 95% Value: 7.0E-07/yr.

This ninety-fifth percentile CDF value is more than two orders of magnitude below the NRC goal of 1E-04/yr.

Uncertainty on the Level 1 Fire PRA results is quantified using a process similar to that described for internal events in Section 19.1.4.1.2.~~6~~7. Parametric uncertainty was represented by selecting an uncertainty distribution for each parameter type including fire initiating events, as described in Section 19.1.4.1.2.~~6~~7. ~~The modeling uncertainty was represented with limited scope by adding uncertainty to the success criteria of EFW pumps and primary relief valves, and by adding uncertainty to the times to overheat for electrical equipment on a loss of HVAC, and to the effectiveness of creative alternate cooling means. These modeling uncertainties are described in detail in Section 19.1.4.1.2.6.~~

~~The results of the uncertainty analysis for fire events are shown in Figure 19.1-18—U.S. EPR Level 1 Internal Fire Events Uncertainty Analysis Results—Cumulative Distribution for Fire Events CDF. Two distributions are presented: one that only incorporates parametric uncertainty and one that incorporates the three cases of~~

~~modeling uncertainty in addition to the parametric uncertainty. The inclusion of modeling uncertainties increases the point estimate by less than a factor of two and has little impact on the ratio of the mean to the point estimate.~~

19.1.5.3.2.8 PRA Insights

The two cutsets that are the largest contributors to the fire CDF are the result of conservative modeling assumptions made due to the lack of detailed design or detailed procedures.

The scenario that contributes the most to fire risk is the fire in the switchgear room of SB 1 or SB 4. It accounts for over 40 percent of the overall fire CDF. This dominance highlights the reliance of some important safety functions (e.g., steam relief via MSRTs, or primary bleed) on a multiple number of electrical divisions. It is also the result of the modeling assumptions on the running train of CCW.

Seal LOCA sequences are important to the fire risk. They also contribute to over 40 percent of the overall fire CDF. If the CVCS switchover to the IRWST is required, the dominant fire scenario would result directly in a total loss of seal cooling to two of the RCPs, and a failure to isolate RCP 4 seals.

The importance measures of systems and components for the internal fires risk show that a broad spectrum of SSCs are risk-significant based on their FV, but none of them dominates. In other word the safety significance of components to the internal fires risk is equally distributed among systems and plant functions. This shows that there is no obvious vulnerability in the U.S. EPR design with respect to the mitigation of the credible fire scenarios. Even though several conservative assumptions were made in the analysis, the total risk from fire events is low with a CDF of less than 2E-07/yr. This illustrates the robustness of the U.S. EPR design and the good spatial separation of the safety trains in the U.S. EPR.

19.1.5.3.3 Level 2 Risk Metrics for Fire Events (LRF and CCFP)

Total LRF from internal fire events is 3.6E-09/yr. This is well below the NRC goal and U.S. EPR probabilistic design goal of 1E-06/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.5.3.3.6.

The CCFP from all fire events (at power) large release sequences is 0.02. This meets the NRC goal of less than approximately 0.1 CCFP

19.1.5.3.3.1 Fire Events Core Damage Release Category Results

The Release Categories and their contribution to the fire events LRF and the associated CCFP are shown in Table 19.1-75—Level 2 Fire Events Release Category Results - LRF.

involves seal LOCAs with a depressurized secondary. In view of this information, sensitivity studies were undertaken to study the factors influencing the induced SGTR contribution to LRF for fires. The sensitivity to manual depressurization and availability of feedwater was therefore studied. It was found that, for the case of fire events the unavailability of primary depressurization had a negligible impact on the RC702 frequency. Both the unavailability of feedwater individually and the combined impact of both being unavailable had a larger impact on both the RC702. However, this sensitivity was not significant when viewed in terms of its impact on LRF, which was increased by less than two times.

19.1.5.3.3.6 Fire Events Level 2 Uncertainty Analysis

The results of the uncertainty evaluation for the Level 2 Fire Events LRF are presented in Figure 19.1-19—U.S. EPR Level 2 Fire Events Uncertainty Analysis Results – Cumulative Distribution for LRF.

The uncertainty results are summarized below:

- LRF Internal Fire Events Mean Value: 3.8E-09/yr.
- LRF Internal Fire Events 5% Value: 3.6E-13/yr.
- LRF Internal Fire Events 95% Value: 3.3E-09/yr.

This ninety-fifth percentile LRF value is more than two orders of magnitude below the NRC goal of 1E-06/yr.

~~An integrated uncertainty analysis was performed for the U.S. EPR Level 1 and Level 2 PRAs.~~

The basis for the input uncertainty distributions for systems related basic events and operator actions is discussed in the sub-sections related to the Level 1 PRA. As discussed in Section 19.1.4.2.2.7, for quantitative evaluation of the overall uncertainty on the LRF, discrete distributions were added for the Level 2 phenomenological basic events. These events are identified in the PRA database by use of the prefix “L2PH.” The distribution form chosen for these basic events is discussed in Section 19.1.4.2.2.7.

~~The results of the uncertainty evaluation for the LRF for fires are presented in Figure 19.1-19—U.S. EPR Level 2 Fire Events Uncertainty Analysis Results—Cumulative Distribution Function for LRF. The fifth percentile for the LRF is 1.4E-10/yr. The ninety-fifth percentile is 1.3E-8/yr. This ninety-fifth percentile LRF value is nearly two orders of magnitude below the goal of 1E-6/yr.~~

place to control foreign material, but are not available in this phase. As a result, the present modeling of the IRWST suction strainers was not changed.

Preventive maintenance modeling was revised for LPSD because of obvious differences in risk management strategies from power operation. Assumptions on maintenance strategies are as follows:

- Maintenance on the SG systems is assumed to be performed on two SGs that are not available in states CAD and CBD.
- Maintenance on the other trains is assumed to occur in state E. One division is assumed to be out for maintenance during that state.

Available mitigating systems in different POSs are defined in Table 19.1-89—System Availability During Shutdown.

19.1.6.1.8 Fire & Flooding Events in Shutdown

Limited evaluation of fire and flooding initiators is performed in the LPSD PRA. Fire and flooding events are evaluated with bounding analyses similar to the analysis performed at-power. Since there is physical separation between RHR trains, and at least two are operating during shutdown, fires and floods can only impact one operating train. Because of the physical separation between operating and standby trains, the impact of the possible degradation in the fire and flood barriers during shutdown is assumed to be not significant. Transient combustibles and maintenance activities may result in a higher fire/flood frequency during shutdown in certain parts of the plant, but are judged to be not significant for the protected RHR trains providing decay heat removal. Based on these judgments, the risk from fire and flood events during at-power operation is assumed to envelop the risk during shutdown.

19.1.6.2 Results from the Low-Power and Shutdown Operations PRA.

19.1.6.2.1 Risk Metrics

The total CDF from shutdown events is 5.8E-08/yr, well below the NRC safety goal of 1E-04/yr (SECY-90-016) and the U.S. EPR probabilistic design goal of 1E-05/yr. Mean value and associated uncertainty distribution can be found in Section 19.1.6.2.7.

19.1.6.2.2 Significant Initiating Events

The significant shutdown initiating events and their contribution to shutdown core damage frequency are given in Table 19.1-90—U.S. EPR Initiating Events Contributions – Level 1 Shutdown. Only those initiating events that contribute more than one percent to the total internal events CDF are listed in the table. All initiating events and their contributions are illustrated in Figure 19.1-21—U.S. EPR Initiating Event Contributions – Level 1 Shutdown. As can be seen from Table 19.1-90 and separation between suction lines, debris retaining capability) and plant procedures (e.g., foreign material control) are expected to ensure that this probability is low.

- Risk from the pressurizer solid state was not considered. Inadvertent start of a reactor coolant pump or a MHSI pump could cause an overpressure event when the pressurizer is solid. The PSVs and RHR relief valves would protect the system from overpressure and the exposure time is small. Thus, overfill events that could lead to a low temperature overpressure event have been considered not likely and have not been identified as initiating events that could significantly contribute to risk.

19.1.6.2.6 Sensitivity Analysis

A sensitivity analysis was performed to evaluate the impact of general modeling assumptions, most of them are also analyzed in Level 1.

The sensitivity results are shown in Table 19.1-100—U.S. EPR Level 1 Internal Events Sensitivity Studies –Level 1 Shutdown. Several insights can be drawn from the sensitivity cases analyzed.

The LPSD CDF is found to be more sensitive to CCFs than the at-power CDF. Diversity of EDGs and SBOs is also found to have a strong impact. The sensitivity on HEPs is also strong. The LPSD CDF is also sensitive to the assumption on the unavailability of the UHS in SBO conditions, which did not have a significant impact on the at-power CDF. These high impacts could be explained by a high LOOP contribution to the LPSD CDF. Also, human actions are essential in shutdown. A sensitivity run was performed to evaluate a benefit from assuming that in the shutdown the UHS fans may not be required. The sensitivity run shows that the UHS fans were not important contributors to the LPSD risk.

A separate sensitivity case was run to check the preventive maintenance assumptions in the LPSD PRA. Preventive maintenance was extended from POS E to POS DU and POS CBU on one train of safety systems. This resulted in a 48 percent increase in the LPSD CDF.

19.1.6.2.7 Uncertainty Analysis

The results of the uncertainty evaluation for the LPSD operation CDF are presented in Figure 19.1-23—U.S. EPR Level 1 Shutdown Events Uncertainty Analysis Results – Cumulative Distribution for Low Power and Shutdown CDF.

The uncertainty results are summarized below:

- CDF LPSD Operation Mean Value: 9.9E-08/yr.

- CDF LPSD Operation 5% Value: 5.2E-09/yr.

- CDF LPSD Operation 95% Value: 2.2E-07/yr.

This ninety-fifth percentile CDF value is more than two orders of magnitude below the NRC goal of 1E-04/yr.

Uncertainty on the Level 1 Shutdown PRA results is quantified using a process similar to that described for internal events in Section 19.1.4.1.2.~~6~~7. Parametric uncertainty was represented by selecting an uncertainty distribution for each parameter type, as described in Section 19.1.4.1.2.~~6~~7. Modeling uncertainty was not represented in the shutdown model.

~~The result of the uncertainty analysis for LPSD operation is shown in Figure 19.1-23—U.S. EPR Shutdown Events Uncertainty Analysis Results – Cumulative Distribution for Low Power and Shutdown CDF. The distribution shown only incorporates parametric uncertainty. There is a relatively large ratio between the mean and the point estimate, indicating a larger impact from the “state-of-knowledge” correlation.~~

19.1.6.2.8 PRA Insights

The LPSD PRA results have shown that events leading to losses of RHR in shutdown are unlikely, but together contribute close to 40 percent of the shutdown risk. The dominant contributor to these initiating events is a LOOP during shutdown states. LOCAs in shutdown and the ultimate level drops in shutdown, contribute approximately 30 percent each to the LPSD CDF.

If the assumptions on the POS durations are to be neglected, the highest risk states are CBD and DU. These are the states where active draining to mid-loop occurs. The possibility to over drain and to have an uncontrolled level drop makes these states relatively risk-significant even though overall risk is low.

19.1.6.3 Low-Power and Shutdown Operations – Level 2 Assessment

19.1.6.3.1 Low Power and Shutdown Level 2 Approach

The analysis of shutdown conditions takes the results of the at-power Level 2 PRA and applies them, with appropriate assumptions, to the results of the shutdown PRA analysis. This approach is judged to be bounding for the low power/shutdown conditions, for both the release category frequencies and for the severity of the source terms expected from accidents initiated from the low power or shutdown states.

In the shutdown condition, the Plant Operating State is characterized by low pressure in the primary system. In state C, the RPV head is on the vessel, and the RCS is intact. This makes the primary system vulnerable to re-pressurization after core melt. In State D, the RPV head is removed, and the primary system remains at low pressure throughout the core melt and containment failure scenario.

19.1.7.3 PRA Input to the Reactor Oversight Process

At the design certification stage, the PRA is not used to support the Reactor Oversight Process.

As stated in FSAR Section 19.1.1.4, the COL applicant will describe the uses of PRA in support of licensee programs such as the Reactor Oversight Process during the operational phase.

19.1.7.4 PRA Input to the Reliability Assurance Program

The design certification PRA is used to provide input to the RAP. Specifically, the PRA is used to identify SSCs that are potentially risk-significant, and therefore should be considered by the RAP expert panel as candidate SSCs under the RAP program. The probabilistic approach to determining SSC risk significance is based on assessment of PRA importance measures. The PRA importance measures do not provide the only insight to SSC risk significance determination. In addition to the PRA importance measures, the expert panel also considers deterministic, safety analysis insights and appropriate operating experience when making the final determination of the RAP scope. Refer to FSAR Section 17.4 for a description of the Reliability Assurance Program.

As stated in FSAR Section 19.1.1.4, the COL applicant will describe the uses of PRA in support of licensee programs such as RAP implementation during the operational phase.

19.1.7.5 PRA Input to the Regulatory Treatment of Non-Safety-Related Systems Program

The U.S. EPR plant design is an evolutionary design primarily based on existing LWR technology and incorporates safety-grade active systems with no passive backup systems. As a result, the RTNSS process is not applicable to the U.S. EPR design. The U.S. EPR design is capable of meeting NRC requirements without the need for the RTNSS process.

19.1.8 Conclusions and Findings

General insights from the PRA analysis related to the different U.S. EPR design features are presented in Table 19.1-102—Summary of Insights from the PRA for the U.S. EPR. The numerical results are discussed below.

19.1.8.1 Risk Metrics:

The total CDF from internal events, internal flooding events, and internal fire events at power is 5.3E-07/yr. This is well below the NRC goal of 1E-04/yr (SECY-90-016), and the U.S. EPR probabilistic design goal of 1E-05/yr.

The total CDF from all events in shutdown is 5.8E-08/yr, also well below the NRC goal of 1E-04/yr (SECY-90-016), and the U.S. EPR probabilistic design goal of 1E-05/yr.

Total LRF from internal events, internal flooding events, and internal fire events at power is 2.6E-08/yr. This is well below the NRC goal and the U.S. EPR probabilistic design goal of 1E-06/yr.

The CCFP from internal events, internal flooding events, and internal fire events at power, for large release sequences is 0.05. This meets the NRC goal of less than approximately 0.1 CCFP.

Mean values and associated uncertainty distributions can be found in Section 19.1.8.4.

19.1.8.2 Risk Distribution:

The distribution of the at-power CDF from internal events, floods, and fires is illustrated in Figure 19.1-24—U.S. EPR Initiating Event Contributions to Total CDF at Power. Internal events contribute 55 percent to the total risk, fires 33 percent and floods 12 percent.

The distribution between the different plant operating states is illustrated in Figure 19.1-25—POS Contributions to Total CDF. At-power risk contributes 90 percent to the total risk. States CBD and DU dominate shutdown risk.

All at-power initiating events that contribute more than one percent to the total CDF at-power, are shown in Table 19.1-103—U.S. EPR Level 1 Top Initiating Event Contributions to the Total CDF at Power. The general LOOP initiating event (which is not SBO or RCP LOCA related) dominates the total risk. Fire in SB 1 or SB 4 switchgear room is the second largest contributor, followed by SLOCA, fire in the MCR and flood in the RB annulus.

The distribution of the at-power LRF from internal events, flood and fire initiating events is illustrated in Figure 19.1-26—U.S. EPR Level 2 Initiating Event Contribution to Total LRF. Internal events contribute 83 percent to the total risk, fires 13 percent and floods 4 percent. The largest contributors are SLBI (47 percent) and SGTR (11 percent).

The distribution of the release categories for the total at-power LRF is illustrated in Figure 19.1-27—U.S. EPR Level 2 Release Category Contribution to Total LRF. Early containment failures in the Release Category 300 family contribute approximately 75 percent to total LRF. Steam Generator Tube Ruptures contribute approximately 20 percent to the total LRF. Containment isolation failures contribute approximately 4 percent, and interfacing system LOCAs contribute approximately 1 percent to the total LRF.

19.1.8.3 Importance Ranking:

Significant SSCs, operator actions and common cause events are defined in the corresponding sections for internal, flood, fire and shutdown events.

Systems ranked based on the contribution to the total CDF at-power are illustrated in Figure 19.1-28—U.S. EPR System Ranked by Importance (FV) – Level 1 Total. The electrical system and ventilation system have the highest contribution to overall risk as could be concluded from the discussions in the earlier sections. The RCS, including RCP seals, also has a very high contribution.

19.1.8.4 Sensitivity and Uncertainty:

A sensitivity analysis was performed to evaluate the impact of a series of assumptions on the CDF from internal, fire and flooding events. The sensitivity results are shown in Table 19.1-104—U.S. EPR Level 1 total Events Sensitivity Studies. The insights that can be drawn from these results are similar to those that were presented for internal events, flooding events, and fire events in the corresponding sections. The impacts from all initiating events are reflected in the total CDF.

As it can be seen from the table, the total CDF is sensitive (delta CDF >100 percent) to the assumptions on HVAC room recovery, HEP values, EDGs and SBO DGs common cause group, and taking all safety train out for a year. It is also sensitive (delta CDF ≈100 percent) to the assumptions on the RCP seal LOCAs, consequential LOOP value, and offsite power recovery. A very conservative sensitivity case was evaluated to estimate combined effects of different assumptions. Overall result is an approximate 14 times increase in the CDF, to 7.5E-06/yr, still well below the NRC goal of 1E-04/yr. This again confirms robustness of the U.S. EPR design.

The results of the Level 1 uncertainty analysis for all internal, fire, and flood initiators are shown in Figure 19.1-29—U.S. EPR Level 1 Internal Events Total Uncertainty Analysis Results – Cumulative Distribution for All Internal, Fire and Flooding Events CDF. ~~Three distributions are presented: one that only incorporates parametric uncertainty, one that incorporates the three cases of modeling uncertainty in addition to the parametric uncertainty, and one that incorporates parametric uncertainty, but without a distribution for the diesel generators failures to run (a point estimate is used only). The inclusion of modeling uncertainties increases the point estimate by about a factor of four. The dominant contributors are assumptions on the EFW success criteria uncertainty, and the uncertainty in the heatup of electrical equipment on a loss of HVAC. The elimination of uncertainty in the diesel generator failure to run failure rate, significantly reduces the impact from "state-of-knowledge" correlation, reducing the mean CDF by a factor of four, due to the high level of redundancy and the high importance of the diesel generators.~~Treatment of parametric uncertainty is described in Section 19.1.4.1.2.7.

The uncertainty results are summarized below:

- CDF Internal, Fire & Flood Events Mean Value: 7.4E-07/yr.
- CDF Internal, Fire & Flood Events 5% Value: 8.7E-08/yr.
- CDF Internal, Fire & Flood Events 95% Value: 2.0E-06/yr.

This ninety-fifth percentile CDF value is more than one order of magnitude below the NRC goal of 1E-04/yr.

The results of the uncertainty analysis for LRF from all internal, fire, and flooding initiators are shown in Figure 19.1-30—U.S. EPR Level 2 Internal~~, Fire, and Flooding~~ Events Total Uncertainty Analysis Results – Cumulative Distribution for all Internal, Fire and Flood Events LRF.

- LRF Internal, Fire & Flood Events Mean Value: 3.6E-08/yr.
- LRF Internal, Fire & Flood Events 5% Value: 7.1E-10/yr.
- LRF Internal, Fire & Flood Events 95% Value: 1.1E-07/yr.

This ninety-fifth percentile LRF value is more than one order of magnitude below the NRC goal of 1E-04/yr.

~~The fifth percentile for LRF is approximately 2.8E-9/yr. The ninety fifth percentile is 1E-7/yr. This ninety fifth percentile is an order of magnitude below the goal of 1E-6/yr.~~

19.1.9 References

1. NUREG-0800, Section 19, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," SRP. U.S. Nuclear Regulatory Commission, June 2007.

2. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," U.S. Nuclear Regulatory Commission, April 2, 1993.

3. ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," The American Society of Mechanical Engineers, April 5, 2002.

4. ASME RA-Sa-2003, Addendum A to RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," The American Society of Mechanical Engineers, December 5, 2003.

Figure 19.1-5—U.S. EPR Level 1 Internal Events Uncertainty Analysis Results – Cumulative Distributions for Internal Events CDF

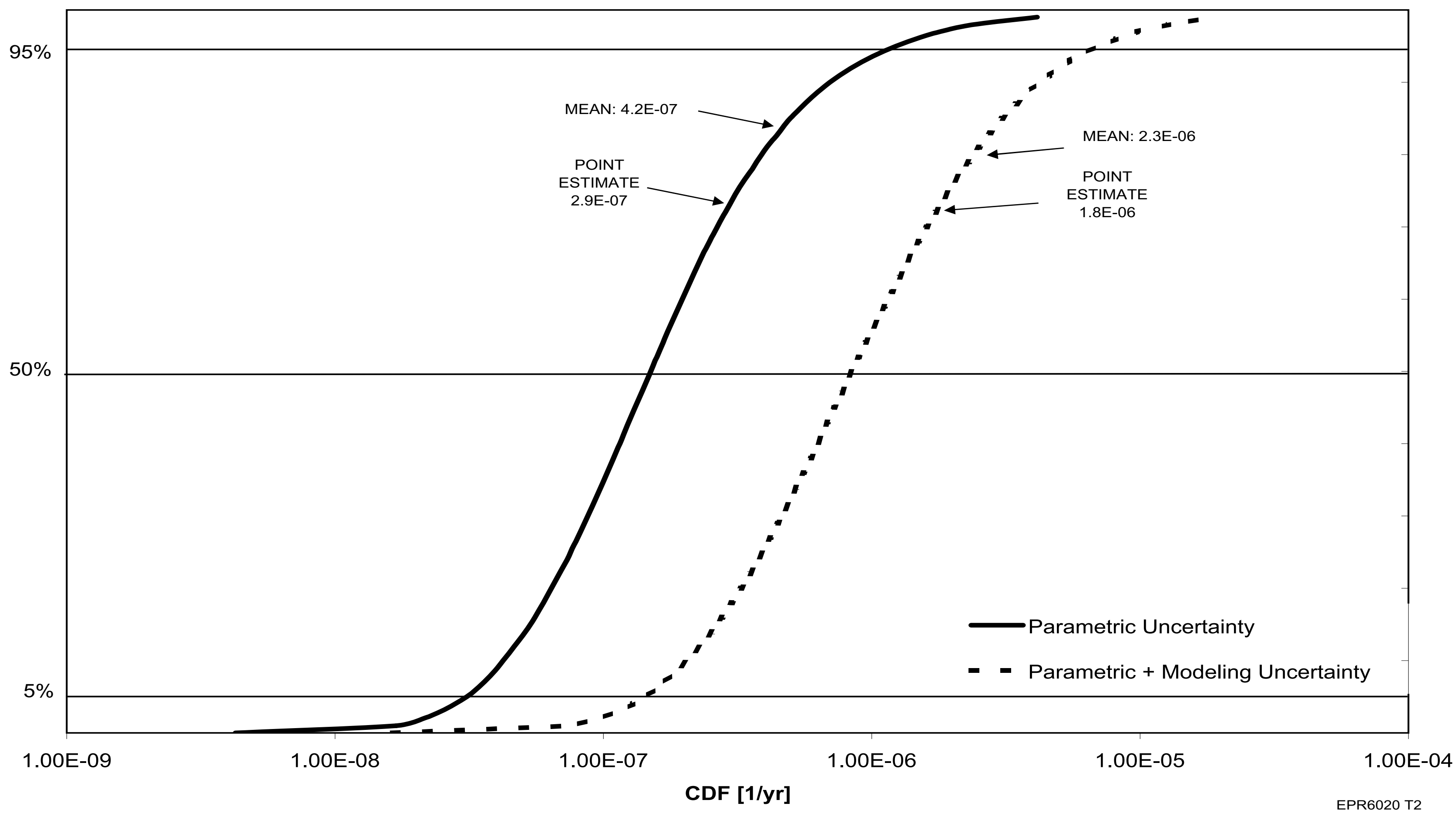


Figure 19.1-9—U.S. EPR Level 2 Internal Events Uncertainty Analysis Results – Cumulative Distribution for Internal Events~~Function for~~ LRF

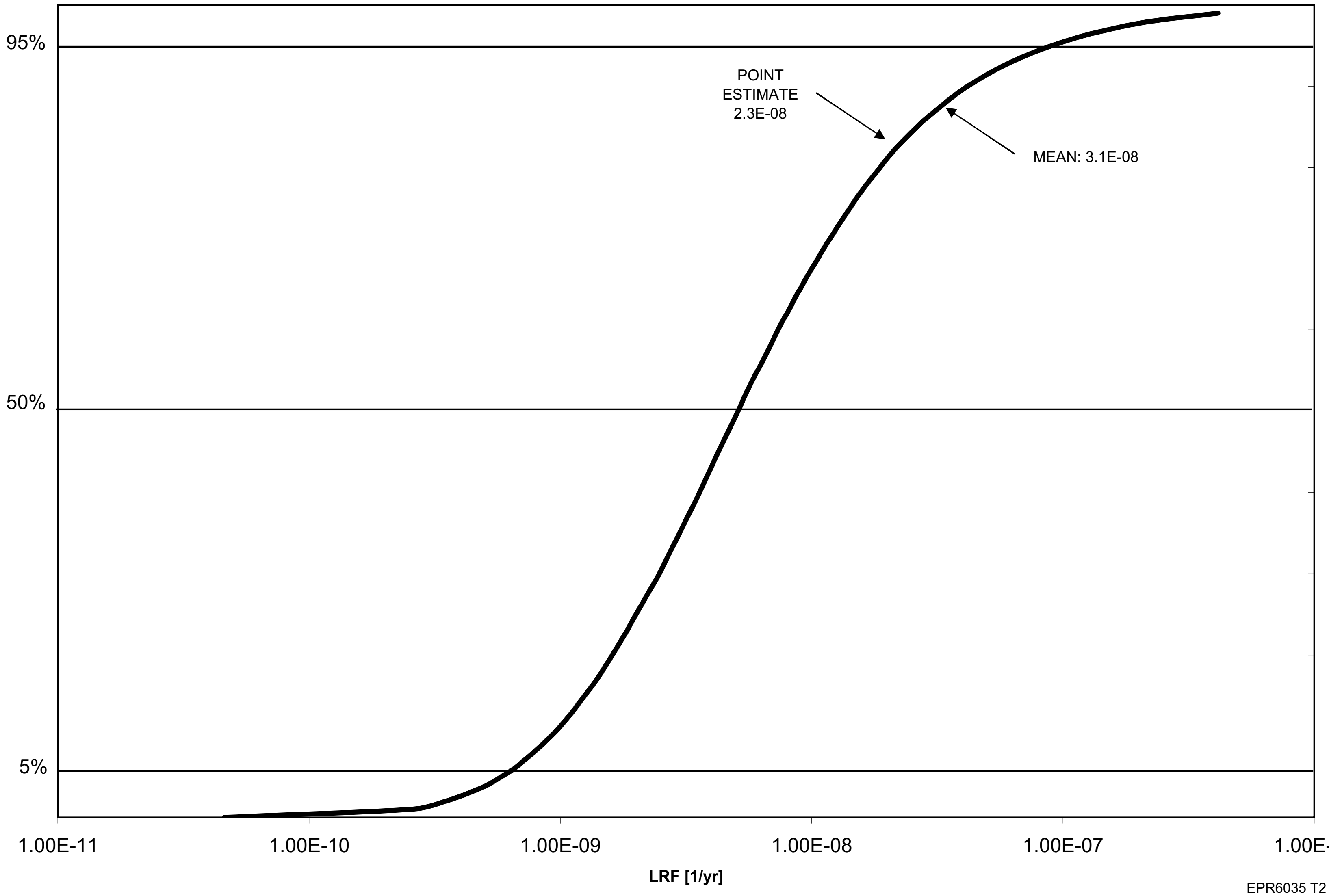


Figure 19.1-13—U.S. EPR Level 1 Internal Flood~~ing~~ Events Uncertainty Analysis Results~~—~~ – Cumulative Distribution for Flood Events CDF

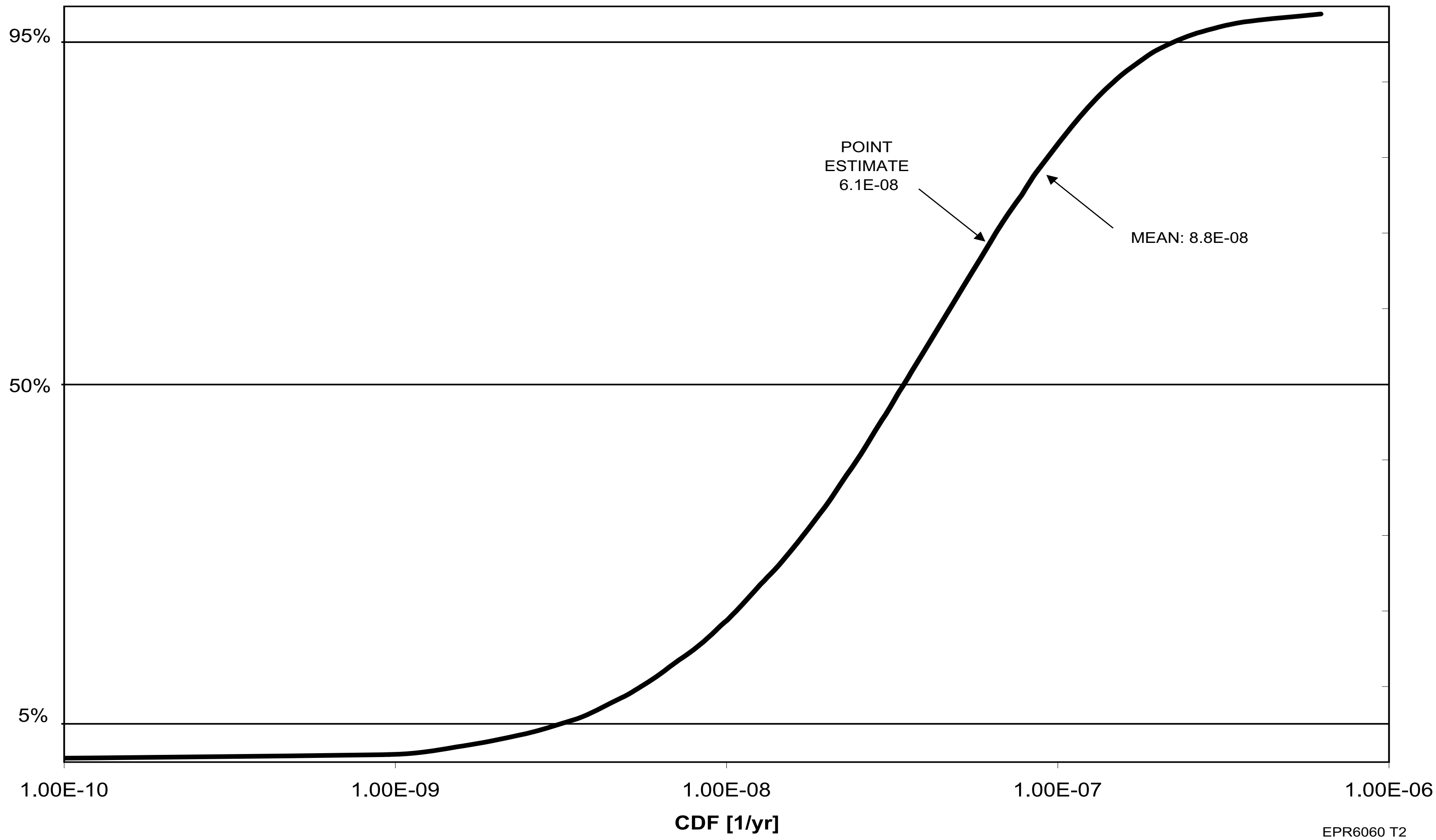


Figure 19.1-14—U.S. EPR Level 2 Flood~~ing~~ Events Uncertainty Analysis Results – Cumulative Distribution for Flood Events~~Function for~~ LRF

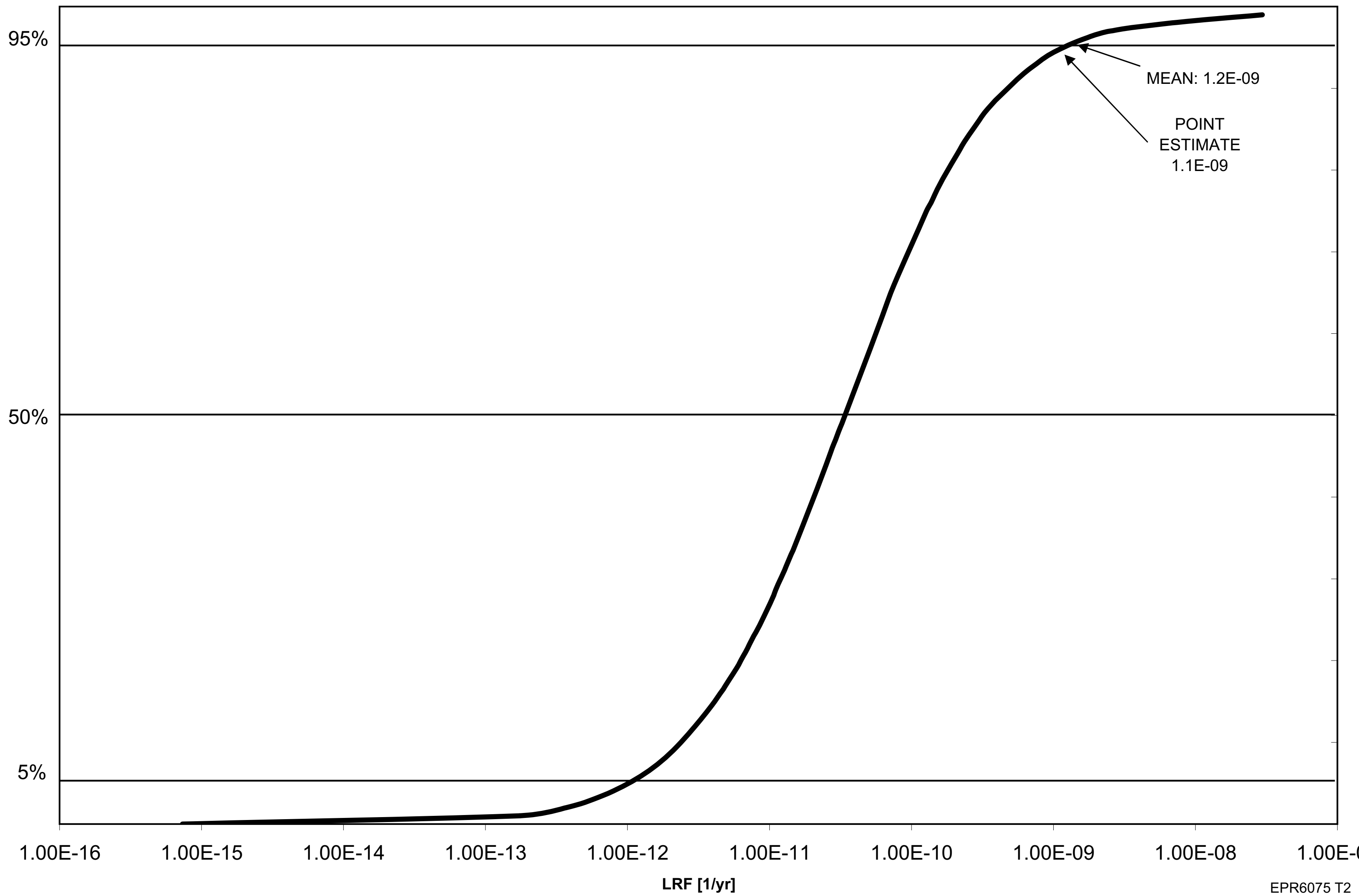


Figure 19.1-18—U.S. EPR Level 1 Internal Fire Events Uncertainty Analysis Results – Cumulative Distribution for Fire Events CDF

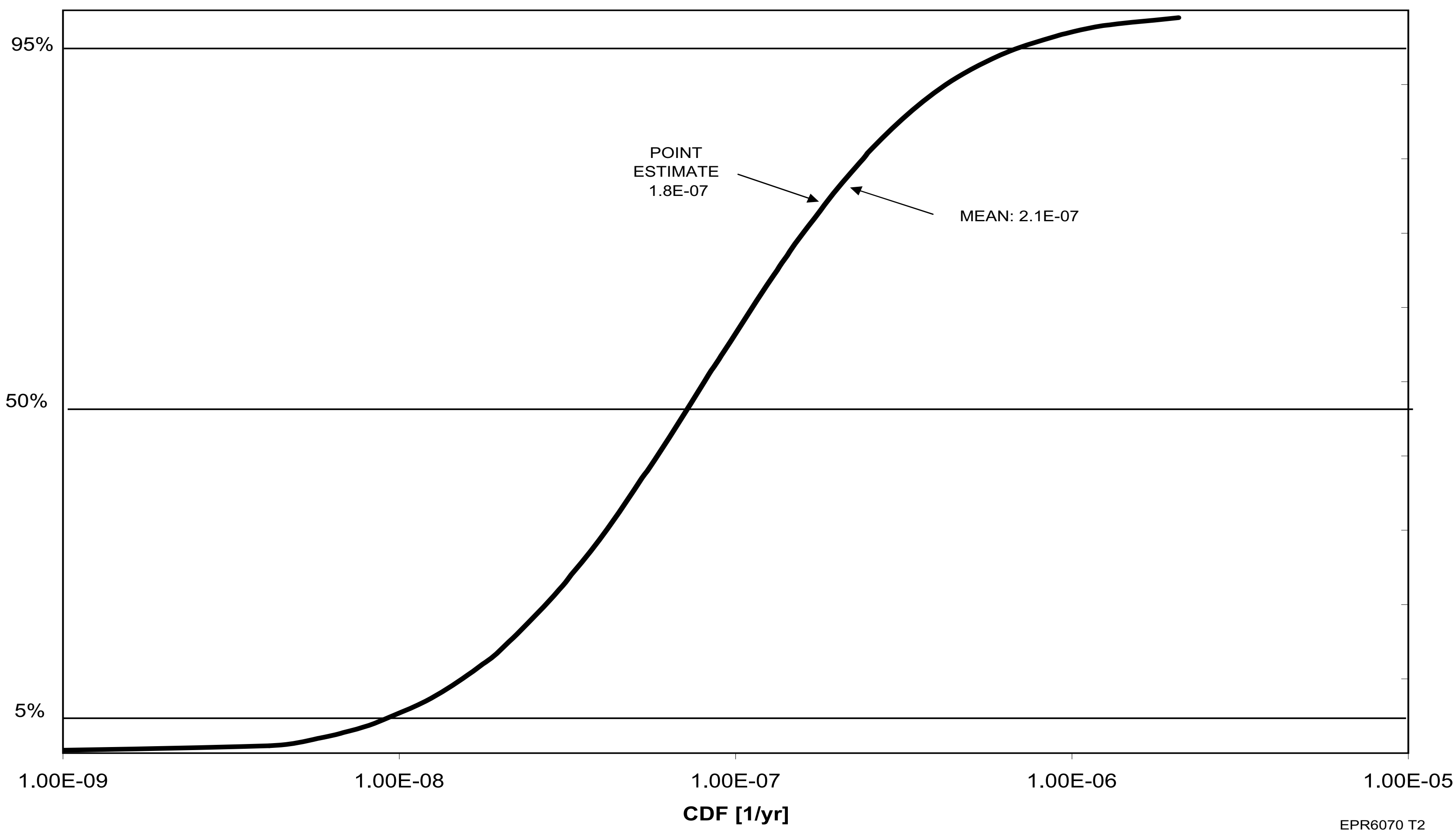


Figure 19.1-19—U.S. EPR Level 2 Fire Events Uncertainty Analysis Results – Cumulative Distribution Function for Fire Events LRF

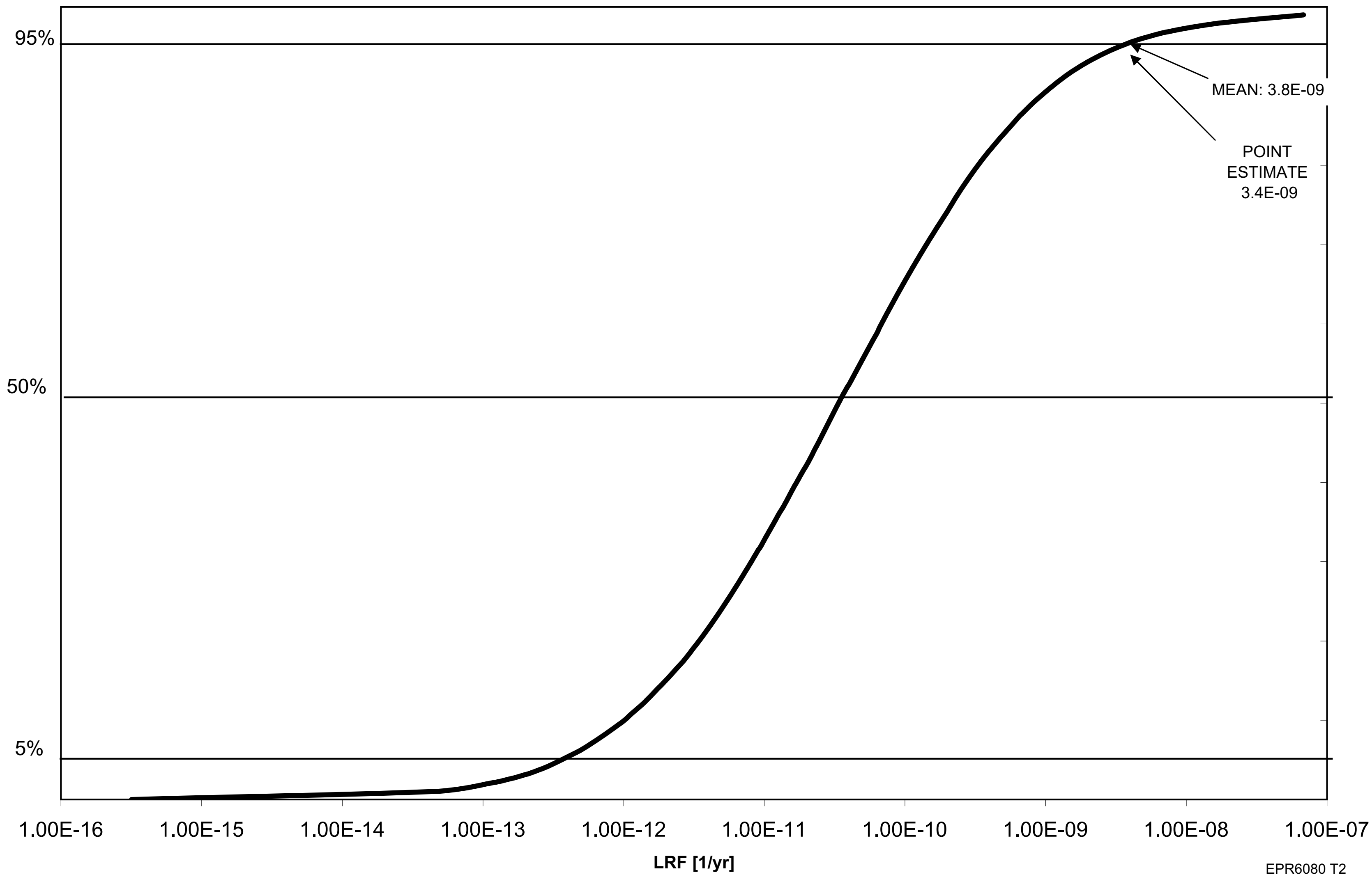


Figure 19.1-23—U.S. EPR Level 1 Shutdown Events Uncertainty Analysis Results – Cumulative Distribution for Low Power and Shutdown CDF

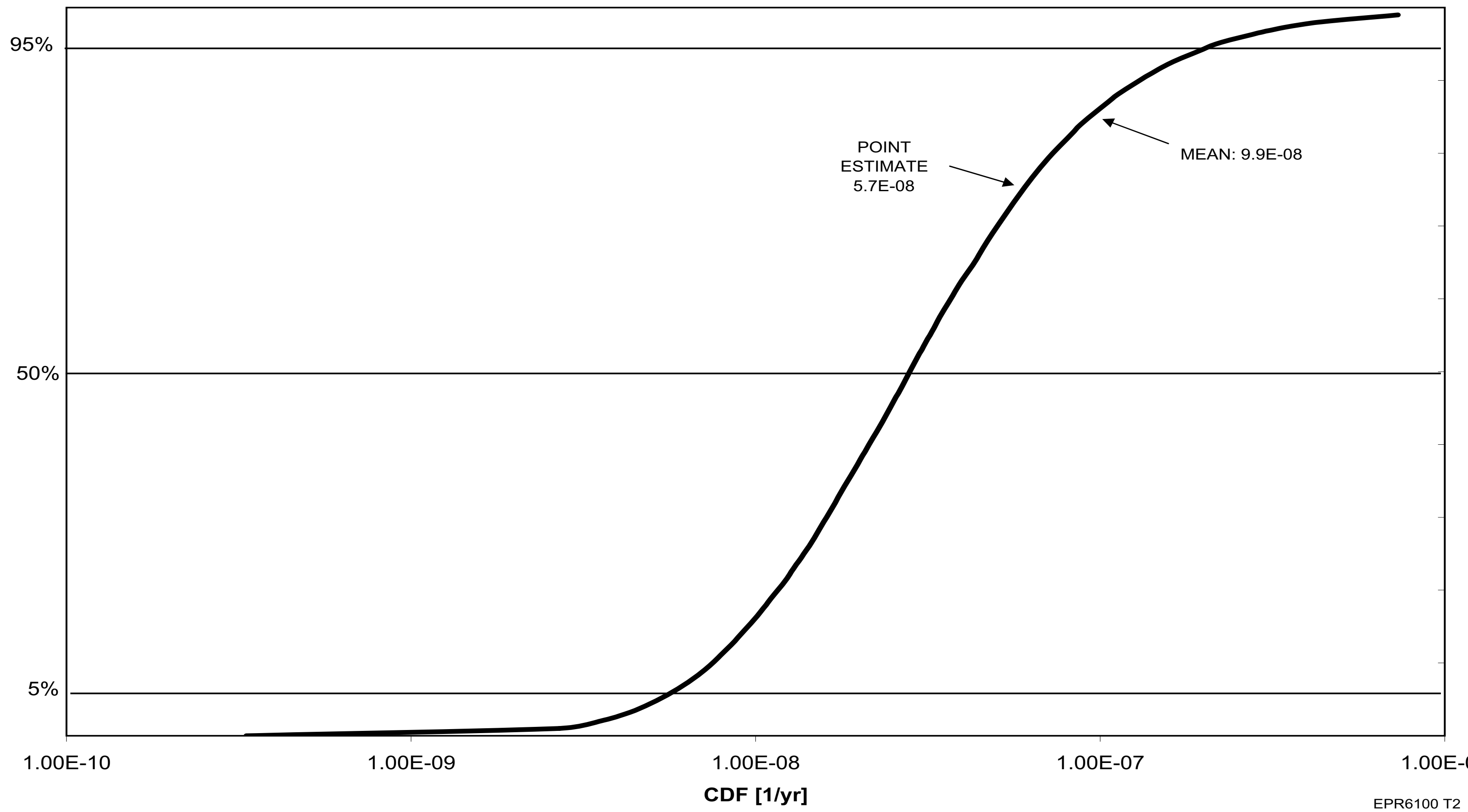


Figure 19.1-29— U.S. EPR Level 1 Internal Events Total Uncertainty Analysis Results – Cumulative Distribution for All Internal, Fire and Flood Events CDF

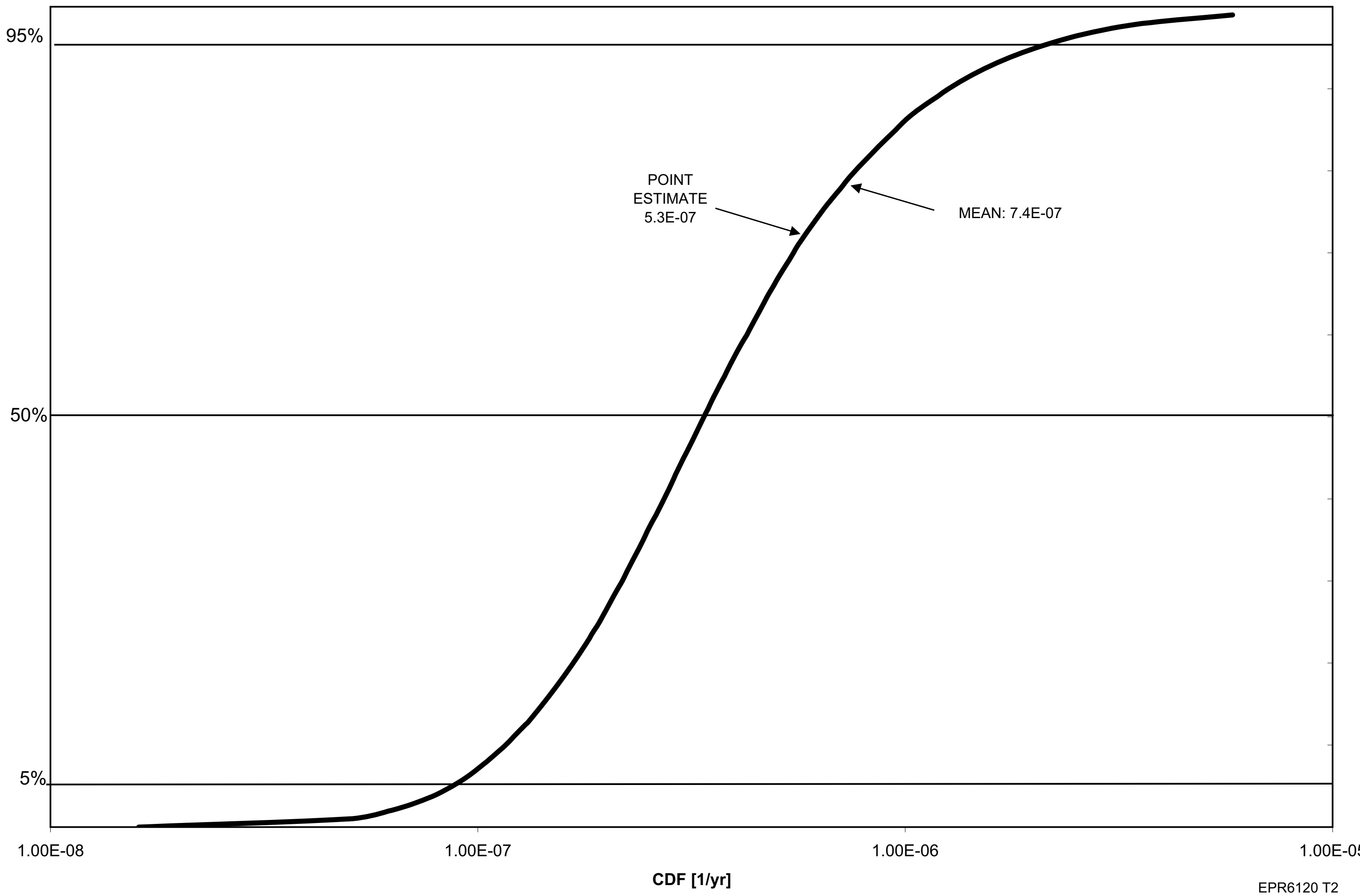
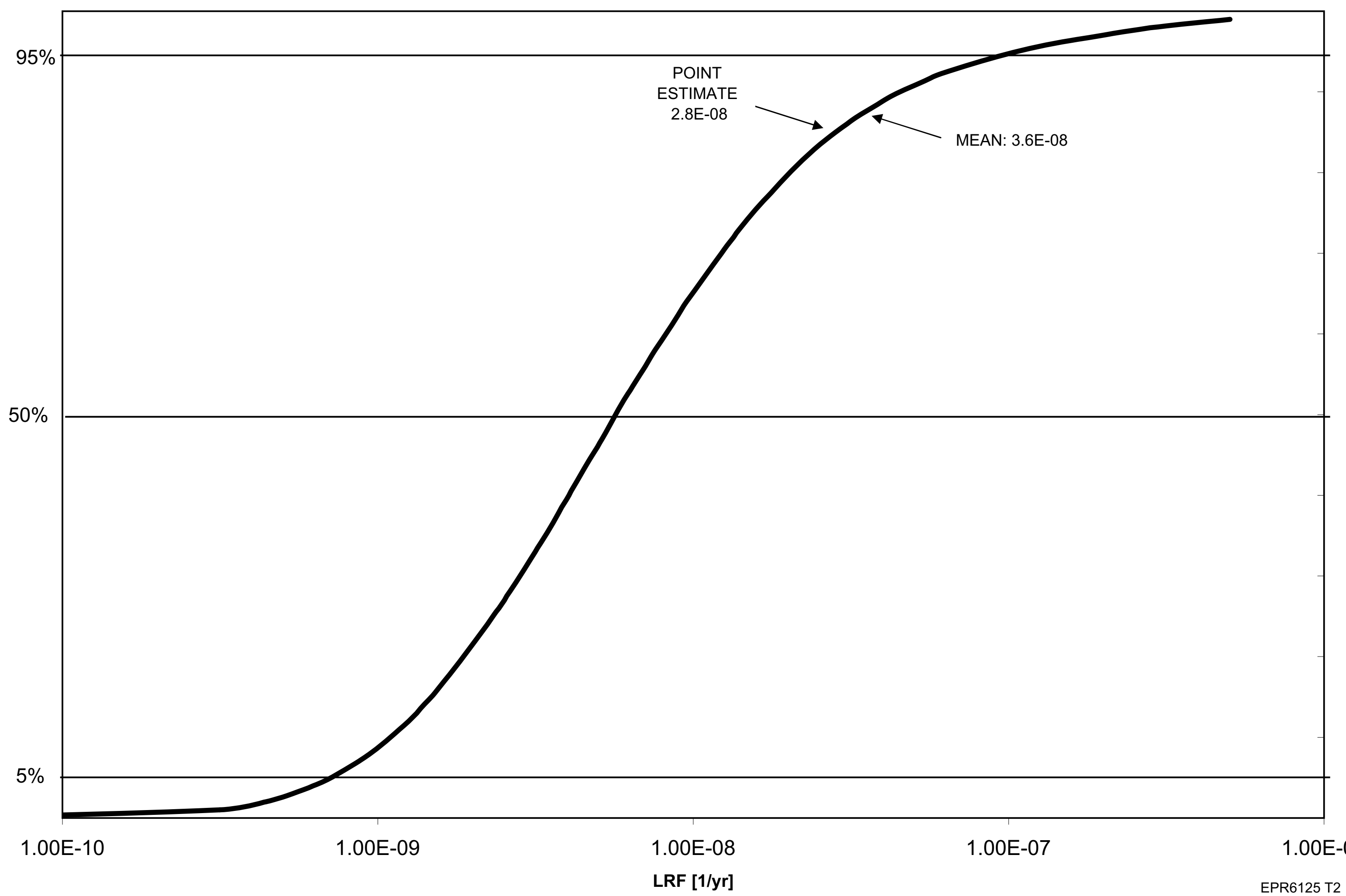


Figure 19.1-30—U.S. EPR Level 2 Internal Events Total Uncertainty Analysis Results – Cumulative Distribution for All Internal, Fire and Flood Events LRF



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