SYSTEM 80+™ STANDARD PLANT DESIGN REDUCED OFF-SITE RADIOLOGICAL IMPACTS

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ABSTRACT

The minimization of potential radiological consequences to the public is one of the most important international issues for nuclear power plant design and operation. While specific criteria and emphasis on long-term vs. short-term effects may vary from country to country, there is a clear international desire to ensure that potential post-accident radiological impacts are minimized. This can be seen through a review of requirements proposed in programs such as the US Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) program, the European Utility Requirements program, the Korean Standard Requirements program, and the People's Republic of China nuclear power program. This paper identifies and compares specific criteria from the various programs and summarizes related numerical results for the System 80+™ Standard Nuclear Power Plant.

For the first two hours after a postulated large-break LOCA, the conservatively-calculated design basis dose to an individual at the site boundary for System 80+ is less than 1.72 Sv to the thyroid and less than 0.026 Sv to the whole body. Corresponding acceptance criteria are 3.0 Sv and 0.25 Sv, respectively. The System 80+ Total Effective Dose Equivalent (TEDE) dose for the same LOCA and two-hour time frame is 0.09 Sv, significantly less than the NRC's acceptance criterion of 0.25 Sv TEDE.

The core damage frequency (CDF) for System 80+ is calculated to be between 1.9E-6 and 3.5E-6 events/year, depending on the assumptions used and reviewed by the US Nuclear Regulatory Commission (NRC). The probability of a large release (0.25 Sv whole body in the first 24 hours) is 5E-8 events/year for a site boundary radius of 800 meters. The probability of a corresponding small release (0.01 Sv whole body in the first 24 hours) is 3E-7 events/year. The potential dose at any distance from the plant is reduced by more than two orders of magnitude for System 80+ relative to the plants analyzed when the NRC established current emergency planning requirements. In fact, the dose at the System 80+ site boundary is less than that at the outer limit of the 10-mile emergency planning zone for the plants originally analyzed. These significantly lower predictions of potential offsite doses suggest that a re-evaluation of current emergency planning requirements should be initiated. The System 80+ analysis has also shown that the size of the food ingestion emergency planning zone justifiably could be reduced in area by a factor of about ten.

BACKGROUND

Protection of the health and safety of the public has always been paramount in the design and operation of nuclear power plants. The release of radioactive material during normal operation is strictly controlled by government regulations and the predicted release of radioactive fission products for a hypothetical accident is carefully analyzed during the design and licensing process to gain confidence that, even in the event of a serious accident, radioactive releases to the public would be minimal. One example of the success of this approach is evidenced by the Three Mile Island (TMI) accident, during which there was severe core damage but only a nominal off-site release yielding maximum individual whole body doses of about 0.001 Sv (US NRC, 1996). Even though the TMI radioactive releases were low, subsequent emphasis was placed on understanding severe accident phenomena and on improved prevention and mitigation design features. This emphasis has impacted operating plants through hardware backfits and additional regulations in countries throughout the world. Examples are the use of filtered vents on European reactors and increased regulatory scrutiny with respect to reactor operation in the US and Pacific Basin countries such as the Republic of Korea and Japan.

The emphasis on the minimization of radioactive releases for ALWRs is clearly demonstrated through design requirements established by utilities, through increased regulatory review criteria for prevention and mitigation of severe accidents, and through features incorporated into ALWR designs by plant vendors. Examples of these new designs are the ABB System 80® and System 80+™ Standard Plant designs, the General Electric ABWR design, the Westinghouse AP-600 design, and the European Pressurized Reactor.

The universal emphasis on development of internationally accepted guidelines for the control of radioactive releases can be seen through a comparison of the design guidance provided in various requirements programs (Table 1). The criteria listed for the design basis LOCA have been applied to operating reactors as well as to ALWRs. The CDF goal specified in Table 1 for ALWRs (1.0E-5 events/year) is more stringent than that which has been used as a goal for reactors in the US (1.0E-4 events/year). The severe accident off-site release criteria have evolved and are continuing to be discussed by various organizations. While these criteria are not the same in all countries, it is clear that all of them provide emphasis on reduced radiological releases and protection of the public.

The design basis LOCA analysis methods used to demonstrate compliance with the above criteria include very conservative computer codes and input assumptions. This conservatism ensures that the licensing decision covers the worst case and that the health and safety of the public is protected with a high degree of confidence. These analysis methods are deterministic in nature because the course of the event is determined by specific models in the computer codes and by specific input assumptions. The analyses used to show compliance with severe accident criteria, however, are based on probabilistic methods due to (1) the large uncertainties in determining the event sequence that leads to core damage, and (2) the uncertainties in input assumptions (e.g., valve operability, human reliability). The probabilistic methods are not as conservative as those for the design basis safety analyses and, therefore, the severe accident analyses are more typical of “best-estimate” analyses.

The minimization of predicted radioactivity releases starts at the beginning of the design process. Conservatism and safety are built into nuclear plant designs using the “defense in depth” approach which includes the conservative selection of design and analytical methods for the design of basic plant structures, systems, and components as well as for safety systems and design basis safety analysis. Experience with equipment procurement and plant operations leads to component design improvements for either increased safety, improved performance or both. A few examples of component design improvements to enhance safety are: (1) the use of larger components, and (2) the use of higher capacity safety systems; both improvements enable the plant to more easily ride through plant transients and prevent damage to the core should the actuation of safety systems be necessary.

Table 1 Comparison of International Requirements for Reduced Radiological Release Potential

Event	US ALWR Utility Requirements	European Requirement	Korean Standard Requirement	China
Design Basis LOCA Limits	• 0.25 Sv whole body and 3.0 Sv thyroid for first 2 hours (operating plants) • 0.25 Sv TEDE for 2 hours from release initiation (for new applications)	• Finland: 0.005 Sv external dose plus ingestion dose for one year • Sweden: << 0.25 Sv whole body and 3.0 Sv thyroid	• 0.25 Sv whole body and 3.0 Sv thyroid for first 2 hours	Not available.
Core Damage Frequency	< 1.0 E-5 events/year	Same	Same	Same
Severe Accident Radiological Release	• 0.25 Sv, acute 24-hour, large release frequency < 1.0 E-6 events/year	• Large release freq. < 1.0 E-6 events/yr • ICRP-63: 24-hour TEDE dose < 0.5 Sv (no evacuation required) • Finland: < 100 TBq Cs-137 released	• Large release freq. < 1.0 E-6 events/year	• Large release freq. < 1.0 E-6 events/year • 8-hour TEDE dose < 0.25 Sv (no evacuation required)

Defense-in-depth also leads to increased safety as a result of increased redundancy, diversity, and simplicity of plant safety systems. One example of such an improvement is the use of four completely independent high-pressure safety injection trains rather than using two high-pressure pumps with headers to feed the four cold legs of the reactor coolant system. Another example is the addition of a third source of AC electrical power (e.g., a gas-turbine generator) that is independent and diverse from the emergency diesel generators. Plant safety margins have also been improved through severe accident prevention and mitigation. This includes the above design improvements that lead to lower CDFs and new mitigation features such as the rapid depressurization system, hydrogen igniter system, cavity flooding system, and a reactor cavity designed for debris retention. The added margin with respect to preventing and mitigating core damage results in a lower likelihood of off-site radiological releases.

SYSTEM 80+ REDUCED RADIOLOGICAL IMPACTS

Design advancements incorporated into the System 80® and System 80+ plant designs have resulted in lower off-site dose predictions for the large-break design basis LOCA, lower core damage frequencies, and lower large release frequencies for severe accidents. The evolutionary improvement process began with ABB-CE's 1300 MWe System 80 plant, continued with the 1050 MWe System 80 plants currently operating and under construction in the Republic of Korea, and culminated in the System 80+ Standard Plant design for which Design Certification was granted by the US NRC in May 1997. The analytical results demonstrate a reduction in the predicted off-site doses for the worst case design basis LOCA (Table 2). The doses are improved most for the System 80+ design due to the incorporation of the NRC's new radiological source term technology. This improvement was achieved at the same time design improvements were made. For example, the large charcoal filters on the containment ventilation system were re-classified as non-safety (i.e., no credit for them in the analysis) in order to decrease plant maintenance costs, and a higher maximum containment leakage rate was assumed in order to provide margin for containment leak testing. Also, much improved CDFs are shown for the 1050 MWe version of System 80 and for System 80+ as a result of ALWR design improvements.

Table 2 Reduced Radiological Impacts for System 80® and System 80+™

Criteria	System 80 (1300 MWe)	System 80 (1050 MWe)	System 80+ (1350 MWe)
Design Basis LOCA Off-site Doses (2-hour dose)	• Whole Body: 0.04 Sv • Thyroid: 1.33 Sv (X/Q=3E-4 s/m3 @ 900m, cont. leakage = 0.1% vol/day, with charcoal filters)	• Whole Body:0.026 Sv • Thyroid: 2.47 Sv (X/Q=2E-4 s/m3 @ 300m, cont. leakage = 0.1% vol/day, with charcoal filters)	• Whole Body:0.026 Sv • Thyroid: 1.72 Sv (X/Q=1E-3 s/m3 @ 800m, cont. leakage = 0.5% vol/day, no charcoal filters) • TEDE: 0.09 Sv
Core Damage Frequency (events/year)	Internal Events:< 8E-5 External Events: 5E-5 Shutdown Risk: 4E-5 •Total CDF: < 1.7E-4	Internal Events: 7.7E-6 External Events: 2.5E-5 Shutdown Risk: 1.5E-6 • Total CDF: 3.4E-5	Internal Events: 1.7E-6 to 1E-7 External Events: 1E-6 Shutdown Risk: 0.8E-6 • Total CDF: 1.9E-6 to 3.5E-6
Large Off-site Release Frequency	For 0.25 Sv/24 hr: < 1.1E-5 events/year @ 900m	8E-6 events/year @ 300m (dose not available)	For 0.25 Sv/24 hr: 5E-8 events/year @ 800m and 6E-8 events/year @ 300m

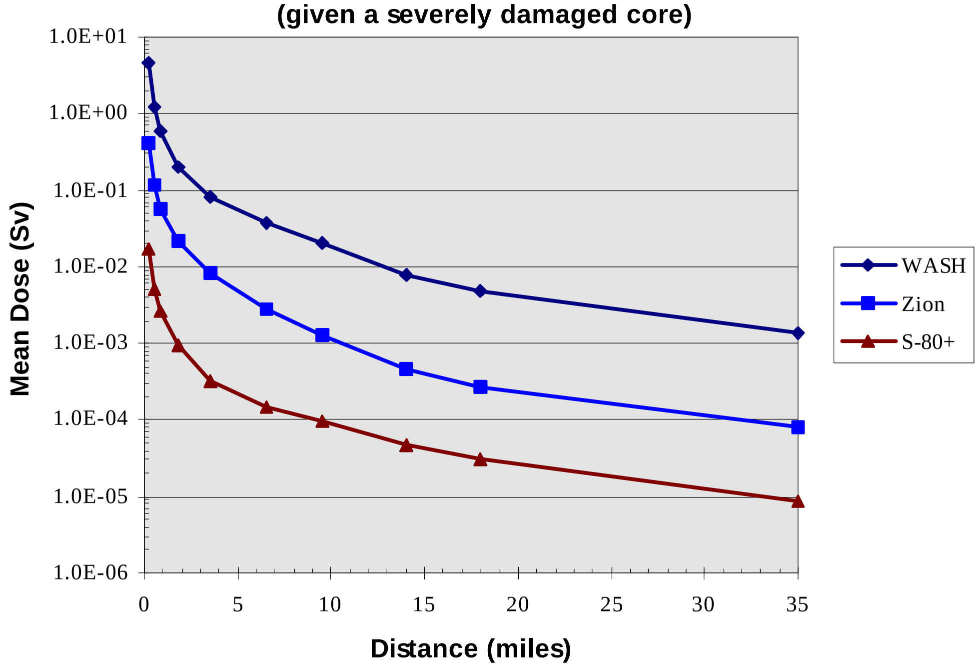
The Probabilistic Safety Assessment methodology used to predict the core damage frequency and large off-site release frequency results reported above follows the assumptions and ground-rules of the US ALWR URD and the requirements of the US NRC. When comparing results for different designs it is critical to use the same methods of calculation and consistent input assumptions. The total CDF decreased from 1.7E-4 events/year to 3.5E-6 events/year as the System 80 design evolved from the 1300 MWe version to System 80+. This improvement was the result of implementing advanced design features such as the rapid depressurization system, a diverse source of on-site AC electrical power, the in-containment refueling water storage tank, and the arrangement of safety equipment into physically separated quadrants. Furthermore, uncertainty analysis reviewed by the US NRC showed that the CDF for internal events would decrease by more than one order of magnitude (to 1E-7 events/year) as the result of using more realistic, but still reasonable, assumptions for items such as the likelihood of reactor vessel failure or common mode failure of plant components (ABB-CE, 1994).

Severe accident mitigation features such as the use of hydrogen igniters, reactor cavity flooding, and a very conservative reactor cavity design (strong walls, thick basemat, and “tortuous” vent path to the upper containment region) resulted in a large off-site release (0.25 Sv) frequency of 5E-8 events/year for a site boundary radius of 800 meters and 6E-8 events/year for a site boundary radius of 300 meters. In addition, the mean individual TEDE (weighted over all core damage events) for an 800 meter site boundary is only 0.0052 Sv. The frequency-weighted Cs-137 activity release to the environment for events contributing to over 89% of the total CDF (i.e., over 97% of the early releases) is only 0.051 TBq. The maximum Cs-137 activity release for an event contributing only 0.3% to the total CDF was 17 TBq. These results are significantly less than the corresponding criteria in Table 1, confirming the expectation that System 80+ releases have large margins to international criteria.

The Nuclear Energy Institute and the Electric Power Research Institute (EPRI) have also conducted a program to develop guidance for: (1) assessing the effectiveness of the containment mitigation function during a severe accident in the context of emergency planning, and (2) predicting the off-site doses consistent with the US Protective Action Guidelines (PAGs) for initiation of emergency response. The PAG dose guidelines are 0.01 Sv TEDE and 0.05 Sv to the thyroid (US EPA, 1991). For the System 80+ design, a severe LOCA was analyzed assuming: (1) a severely damaged core and consequent reactor vessel failure, (2) only one train of containment spray operating, but crediting of spray hygroscopicity, (3) a maximum containment leakage rate of 0.5% volume/day, and (4) a median dose analysis using limiting US meteorological conditions. The resulting 24-hour median doses at the site boundary for System 80+ plants are 0.003 Sv TEDE and 0.027 Sv to the thyroid. These very low off-site doses are the result of improved severe accident mitigation systems and containment design. It is believed, therefore, that the design improvements and the very low off-site doses justify a relaxation in the US emergency planning requirements.

A review of dose as a function of distance from the plant also provides confidence that ALWR designs would provide improved protection of the public health and safety in the event of a severe accident. The MACCS code (US NRC/Sandia, 1990) was used to analyze the 24-hour mean dose to an individual as a function of distance from the plant following a severe accident. Using the same methodology and input assumptions that were used for the System 80+ analysis, doses were predicted for the Zion plant (US NRC/Brookhaven, 1997) and the “WASH-1400” PWR that was analyzed when current emergency planning requirements (US NRC, 1978) were established. The results show that for System 80+ the potential dose to an individual is reduced by more than two orders of magnitude at all distances from the plant relative to the WASH-1400 PWR (Figure 1). The improvement relative to the Zion plant is more than one order of magnitude at all distances. Figure 1 also shows that the dose at the System 80+ site boundary (800m radius) is less than that for the WASH-1400 PWR at the outer boundary of the 10-mile emergency planning zone. It is believed that such improved results (even without credit for the System 80+ reduced CDF) provide the basis for initiating a re-evaluation of the US requirements for emergency planning within a 10-mile radius for ALWR designs.

Figure 1
Mean Individual Dose vs. Distance
(given a severely damaged core)



Radiation exposure to an individual can also result from ingestion of foodstuffs. In the US, the emergency response for the ingestion pathway is triggered by projected ground concentrations

of several radionuclides and associated projected radiation doses. The "limiting" pathway (i.e., that used to establish the current US requirement for a 50-mile radius Ingestion-Pathway Emergency Planning Zone, or EPZ) is the dose to the infant thyroid via the drinking of contaminated milk.

Using the WASH-1400 source term set (US NRC, 1975), it was shown in NUREG-0396 that only ~30% of the core melt accidents would be expected to produce doses exceeding 1.5 rem to the infant thyroid beyond 50 miles. This was the basis for limiting the Ingestion-Pathway EPZ to that radius. In the PAG Manual, it is explained that the 1.5 rem infant thyroid dose PAG may be exceeded whenever the post-accident I-131 ground concentration exceeds approximately 4800 Bq/m2 (0.13 $\mu Ci/m^2$). Therefore, the likelihood of exceeding the infant thyroid dose ingestion PAG for the System 80+ design can be determined by predicting the center-line (i.e., maximum) ground concentration as a function of distance from the plant for a range of severe accidents. In such an evaluation, the objective would be to determine the distance at which the post-severe accident I-131 ground concentration exceeds 4800 Bq/m^2 30% of the time, using source terms reflecting the severe accident behavior of System 80+ and the advances in severe accident source term understanding since WASH-1400.

Such an evaluation can be approximated by selecting a source term which is not exceeded by more than 30% of the severe accidents from the System 80+ PRA (i.e., the accident source term discussed above), and then using the median center-line concentration resulting from that source term as the basis for establishing the required distance. Although this approach simplifies the evaluation, the distance remains a function of both weather and deposition velocity. Using the 80-90th percentile meteorological data set from the US ALWR Program (which tends to produce high concentrations in the air and on the ground), it was found that the required distance would be only about 18 miles with a dry deposition velocity (DDV) of 1 cm/sec and about 12 miles with a deposition velocity of 0.3 cm/sec. The former DDV value was used in NUREG-0396 and the latter value was used in NUREG/CR-4551 (US NRC, 1990). This indicates that a substantial reduction in land area subject to detailed emergency planning for the ingestion pathway (about a factor of ten) is warranted for System 80+. Note that this analysis of the ingestion pathway addresses only the change in ingestion dose vs. distance *given that a severe accident has occurred*, and says nothing about the substantial reduction in core damage frequency discussed above.

CONCLUSION

The emphasis that international ALWR programs have placed on reducing the potential for off-site radiological consequences has been successfully demonstrated for the System 80+ Standard Nuclear Power Plant design. For design basis LOCAs, reduced off-site dose predictions result from use of the US NRC's new radiological source term technology. For severe accident analyses, reduced off-site dose predictions result from lower core damage frequency, improved safety system performance, and improved containment reliability. The potential dose at any distance from the plant is reduced by more than two orders of magnitude for System 80+ relative to the plants analyzed when the NRC established current emergency planning requirements. Similarly, long-term ingestion pathway contamination is also greatly reduced. It is believed, therefore, that the significantly lower potential for off-site radiological consequences justifies a re-evaluation and relaxation of current emergency planning requirements.

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KEY WORDS

System 80+™, ALWR, Off-site, Dose, Source Term, Emergency Planning, Ingestion Pathway.