ENVIRONMENTAL ASSESSMENT BY THE U.S. NUCLEAR REGULATORY COMMISSION RELATING TO THE CERTIFICATION OF THE AP1000 STANDARD PLANT DESIGN

DOCKET NO. 52-006

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

RELATING TO THE CERTIFICATION OF THE

AP1000 STANDARD PLANT DESIGN

DOCKET NO. 52-006

The U.S. Nuclear Regulatory Commission (NRC) has issued a design certification for the Advanced Passive 1000 (AP1000) design in response to an application submitted on March 28, 2002, by Westinghouse Electric Company LLC (hereinafter referred to as Westinghouse). A design certification is a rulemaking that amends Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 52).

The NRC has performed an environmental assessment (EA) of the environmental impacts of the proposed new rule and has documented its findings of no significant impact in accordance with the requirements of 10 CFR 51.21 and the National Environmental Policy Act of 1969 (NEPA), as amended. This EA also addresses the severe accident mitigation design alternatives (SAMDAs), that the NRC has considered as part of this EA for the AP1000 design. This EA does not address the site-specific environmental impacts of constructing and operating a facility, which references the AP1000 design certification at a particular site; such impacts will be evaluated as part of any application or applications for the siting, construction, or operation of a facility.

As discussed in detail in Section 4.0 of this EA, the NRC determined that issuing this design certification does not constitute a major Federal action significantly affecting the quality of the human environment. The basis for this finding of no significant impact is that the design certification would not authorize the siting, construction, or operation of a facility of an AP1000 reactor design. Rather, the certification would merely codify the AP1000 design in a rule that could be referenced in a construction permit (CP), combined license (COL), or operating license (OL) application. Further, because the certification is just a rule, it does not involve any resources that have alternative uses. Therefore, the NRC has not prepared an environmental impact statement (EIS) in connection with this action.

The NRC also reviewed Westinghouse's evaluation of SAMDAs that generically apply to the AP1000 design. On that basis, the NRC found that the evaluation provides reasonable assurance that there are no additional SAMDAs beyond those currently incorporated into the AP1000 design which are cost-beneficial, whether considered at the time of the approval of the AP1000 design certification or in connection with the licensing of a future facility referencing the AP1000 design certification, where the plant referencing this appendix is located on a site whose site parameters are within those specified in Appendix 1B of the AP1000 design control document (DCD). These issues are considered resolved for the AP1000 design.

ENVIRONMENTAL ASSESSMENT

1.0 Identification of the Proposed Action

The proposed action would certify the AP1000 design under Appendix D to 10 CFR Part 52. The new rule would allow prospective licensees to reference the certified AP1000 design as part of a combined license (COL) application under 10 CFR Part 52 or may allow for a construction permit (CP) application under 10 CFR Part 50.

2.0 The Need for the Proposed Action

The NRC has long sought the safety benefits of commercial nuclear power plant standardization and early final resolution of design issues. The NRC plans to achieve these benefits by certifying nuclear plant designs. Subpart B to 10 CFR Part 52 allows for certification in the form of rulemaking of an essentially complete nuclear plant design.

The proposed action would amend 10 CFR Part 52 to certify the AP1000 design. The amendment would allow prospective licensees to reference the certified AP1000 design as part of a COL application under 10 CFR Part 52 or may allow for a CP application under 10 CFR Part 50. Those portions of the AP1000 design included in the scope of the certification rulemaking would not be subject to further safety regulatory review or approval in a COL proceeding. In addition, the design certification rule would eliminate the need to consider SAMDAs for any future facilities that reference the certified AP1000 design.

3.0 The Environmental Impact of the Proposed Action

Issuing an amendment to 10 CFR Part 52 to certify the AP1000 standard plant design would not constitute a significant environmental impact. The amendment would merely codify the NRC's approval of the AP1000 design (refer to NUREG-1793). Furthermore, because the amendment is a rule, it involves no resources that have alternative uses.

As described in Section 4.0 of this EA, the NRC reviewed alternatives to the design certification rulemaking and alternative design features for preventing and mitigating severe accidents. NEPA requires consideration of alternatives to show that the design certification rule is the appropriate course of action and to ensure that the design referenced in the rulemaking does not exclude any cost-beneficial design changes related to the prevention and mitigation of severe accidents. The NRC concludes that, unlike the proposed design certification rule, the alternatives to certification do not provide for resolution of issues.

Design certification is in keeping with the Commission's intent to make future plants safer than the current generation of plants, to achieve early resolution of licensing issues, and to achieve the safety benefits of standardization (refer to the Advanced Reactor (51 FR 24643), Standardization (52 FR 348803), and Severe Accident Policy Statements (50 FR 32138), and to 10 CFR Part 52). Through its own independent analysis, the NRC also concludes that Westinghouse adequately considered an appropriate set of SAMDAs and that none were cost-beneficial. Although Westinghouse made no design changes as a result of reviewing the SAMDAs, Westinghouse had already incorporated certain features in the AP1000 design on the basis of the probabilistic risk assessment (PRA) results. Section 4.2 of this EA gives examples of these features. These design features relate to severe accident prevention and mitigation, but were not considered in the SAMDA evaluation because they were already part of the AP1000 design (refer to Section 19.1.6.2 of NUREG-1793, "AP1000 Design Improvement as a Result of Probabilistic Risk Assessment Studies").

Finally, the design certification rule by itself would not authorize the siting, construction, or operation of a nuclear power plant. The issuance of a CP, early site permit (ESP), COL, or OL which references the AP1000 design will require a prospective applicant to address the environmental impacts of construction and operation at a specific site. The NRC will then evaluate the environmental impacts and issue an EIS in accordance with 10 CFR Part 51. However, the SAMDA analysis has been completed as part of this EA and can be incorporated by reference into an EIS related to siting, construction, or operation of a nuclear plant that references the AP1000 design.

4.0 Alternatives to the Proposed Action

The NRC has an alternative to certifying the AP1000 design: take no action to certify the design under Subpart B of 10 CFR Part 52. As with the proposed action, this alternative would not have a significant impact on the quality of the human environment because it would not authorize the siting, construction, or operation of a facility.

In the alternative, the NRC would not certify the AP1000 design in a rulemaking. The NRC issued a final design approval for AP1000 under Appendix O to 10 CFR Part 52 on September 13, 2004. Therefore, although the NRC has approved the design, the design would not have finality in proceedings under 10 CFR Part 50 or 10 CFR Part 52, Subpart C and could be modified. As a result, the design could require re-evaluation as part of each application to construct and operate a facility of an AP1000 design at a particular site. This alternative would provide for early internal NRC resolution of design issues to the extent that the design would remain unchanged at the facility application stage, but would not obtain all of the benefits of standardization nor permit overall finality for the resolved design issues.

The NRC sees no advantage in this alternative compared to the design certification rulemaking proposed for the AP1000 design. Although neither the alternative nor the proposed action (design certification rulemaking) would significantly affect the quality of the human environment, the proposed action achieves the benefits of standardization, permits early resolution of design issues, and provides finality in licensing proceedings for the resolved design issues (including SAMDAs) that are within the scope of the design certification. Therefore, the NRC concludes that the alternative to rulemaking would not achieve the objectives that the Commission intends by certifying the AP1000 design pursuant to 10 CFR Part 52, Subpart B.

4.1 Severe Accident Mitigation Design Alternatives

Consistent with the objectives of standardization and early resolution of design issues, the Commission decided to evaluate SAMDAs as part of the design certification for the AP1000 design. In a 1985 policy statement, the Commission defined the term “severe accident” as an event that is “beyond the substantial coverage of design-basis events,” including events where there is substantial damage to the reactor core (whether or not there are serious offsite consequences). Design-basis events are events analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the DCD.

As part of its design certification application, Westinghouse performed a PRA for the AP1000 design to achieve the following objectives:

- Identify the dominant severe accident sequences and associated source terms for the design.
- Modify the design, on the basis of PRA insights, to prevent or mitigate and reduce the risk of severe accidents.
- Provide a basis for concluding that all reasonable steps have been taken to reduce the chances of occurrence, and mitigate the consequences, of severe accidents.

Westinghouse's PRA analysis is described in Chapter 19 of the AP1000 DCD.

In addition to considering alternatives to the rulemaking process discussed in Section 3.0, applicants for reactor design certification, COLs, or CPs must also consider alternative design features for severe accidents consistent with the requirements of 10 CFR Part 50, and with a court ruling related to NEPA. These requirements can be summarized as follows:

- 10 CFR 52.79 and 10 CFR 50.34(f)(1)(i)[1] requires the applicant to perform a plant/site-specific PRA, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.
- The U.S. Court of Appeals decision, in *Limerick Ecology Action v. NRC*, 869 F.2d 719 (3rd Cir. 1989), effectively requires the NRC to consider certain SAMDAs in the environmental impact review performed under Section 102(2)(c) of NEPA.

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[1]Although 10 CFR 50.34(f)(1)(i) by its terms does not apply to new construction permits (CP), the Commission's policy is that a CP applicant will be required to comply with 50.34(f)(1)(i).

Although these requirements are not directly related, they share a common purpose to consider alternatives to the proposed design, to evaluate whether potential alternative improvements in the plant design might increase safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed. It should be noted that the Commission is not required to consider alternatives to the design in this EA. However, as a matter of discretion, the Commission has determined that considering SAMDAs concomitant with the rulemaking is consistent with the intent of 10 CFR Part 52 for early resolution of issues, finality for resolved design issues, and achieving the benefits of standardization.

In its decision in *Limerick Ecology Action v. NRC*, the Court of Appeals for the Third Circuit expressed its opinion that it would likely be difficult to evaluate SAMDAs for NEPA purposes on a generic basis. However, the NRC has determined that generic evaluation of SAMDAs for the AP1000 standard design is warranted for two significant reasons. First, the design and construction of all plants referencing the certified AP1000 design will be governed by the rule certifying a single design. Second, the site parameters specified in the rule and the

AP1000 DCD establish the consequences for a reasonable enveloping set of SAMDAs for the AP1000 design. The low residual risk of the AP1000 design and the limited potential for further risk reductions provides high confidence that additional cost-beneficial SAMDAs would not be found for sites within the site parameter envelope assumed for the AP1000 EA of SAMDAs. If the actual parameters for a particular site exceed those assumed in the rule and the DCD, then SAMDAs must be re-evaluated in the site-specific environmental report and the EIS. If the actual parameters for a postulated site are bounded by those assumed in the rule and the DCD, then the SAMDA analysis can be incorporated by reference in the site-specific EIS.

4.2 Potential SAMDAs Identified by Westinghouse

To identify candidate design alternatives, Westinghouse reviewed the design alternatives for other plants including the CE System 80+. Westinghouse also reviewed the results of the AP1000 PRA and design alternatives suggested by AP1000 design personnel.

Westinghouse eliminated the following SAMDAs from further consideration because they are already incorporated in the AP1000 design:

- hydrogen ignition system
- reactor cavity flooding system
- reactor coolant pump seal cooling (AP1000 has canned motor pumps)
- reactor coolant system (RCS) depressurization
- external reactor vessel cooling
- non-safety-grade containment sprays

Several risk-significant enhancements to the AP600 design have also been incorporated in the AP1000 design and were therefore not further considered. These modifications are summarized below and discussed further in DCD Tier 2, Section 1B.1.5, “Summary of Risk Significant Enhancement.”

- a change in the normal position of the two containment motor-operated recirculation valves (in series with squib valves) from closed to open to improve the reliability of opening these flowpaths
- a change in the emergency operating procedures (EOPs) to call for in-containment refueling water storage tank (IRWST) draining earlier in an event to improve the probability of successful operator action
- a change in the design of the IRWST vents to preferentially direct hydrogen releases to the IRWST pipe vents, where diffusion flames will not adversely impact the containment

- incorporation of a low-boron core to reduce the potential contribution of anticipated transient without scram (ATWS) events to plant risk
- addition of a third passive containment cooling system (PCS) drainline with a motor-operated valve (MOV) that is diverse from the air-operated valves (AOVs) used in the other two drainlines, to improve PCS reliability
- specification that two of the four squib valves in the recirculation lines be low-pressure-type valves, and the remaining two squib valves be high-pressure-type valves to reduce the contribution to core damage frequency (CDF) from common-cause failures (CCFs) of recirculation squib valves

On the basis of the screening, Westinghouse retained 14 potential SAMDAs for further consideration. This set of SAMDAs is the same as that considered for the AP600 design. DCD Tier 2, Section 1B.1.3, “Selection and Description of SAMDAs,” describes the 14 design improvements as follows:

(1) Upgrade the chemical and volume control system (CVCS) for small loss-of-coolant accidents (LOCAs): The CVCS is currently capable of maintaining the RCS inventory for LOCAs for effective break sizes up to 0.97 cm (3/8 in.) in diameter. A design alternative involving the upgrade of the CVCS for small LOCAs would increase the capability of the CVCS, enabling it to maintain RCS inventory during small- and intermediate-size LOCAs (up to an effective break size of 15.2 cm (6 in.) in diameter). Implementation of this design alternative would require installation of IRWST and containment recirculation connections to the CVCS, as well as the addition of a second line from the CVCS pumps to the RCS.

(2) Filtered vent: This design alternative would involve the installation of a filtered containment vent, including all associated piping and penetrations. This modification would provide a means to vent containment to prevent catastrophic overpressure

failures and would also provide a filtering capability for source term release. The filtered vent would reduce the risk of late containment failures that might occur after failure of the PCS. Note, however, that even if the PCS fails, it is expected that air cooling will limit the containment pressure to less than the ultimate pressure capacity of the containment under most environmental conditions.

(3) Self-actuating containment isolation valves (CIVs): Self-actuation of CIVs could be used to increase the likelihood of successful containment isolation during a severe accident. This design alternative would involve the addition of a self-actuating valve or the enhancement of the existing CIVs on normally open containment penetrations (i.e., penetrations that provide normally open pathways to the environment during power and normal shutdown conditions). The design alternative would provide for self-actuation in the event that containment conditions are indicative of a severe accident. Closed systems inside and outside containment, such as the normal residual heat removal system (RNS) and component cooling, would be excluded from this design alternative. The actuation of CIVs would be automatically initiated in the event that containment conditions are indicative of a severe accident.

(4) Passive containment sprays: This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 design (in lieu of the non-safety-related active containment spray capability currently incorporated in the AP1000 design). Installation of the safety-grade containment spray system could result in an increase in the following three risk benefits:

- scrubbing of fission products, primarily for containment isolation failure
- alternative means for flooding the reactor vessel (in-vessel retention)
- control of containment pressure if the PCS fails

(5) Active high-pressure safety injection (HPSI) system: A safety-related active HPSI system could be added that would be capable of preventing a core melt for all events except the large-break LOCA and ATWS. Note, however, that this design alternative is not consistent with the AP1000 design objectives. The AP1000 would change from a plant with passive systems to a plant with passive and active systems.

(6) Steam generator (SG) shell-side heat removal system: This design alternative would involve the installation of a passive safety-related heat removal system to the secondary side of the SGs. This enhancement would provide closed-loop secondary-system cooling by means of natural circulation and stored water cooling, thereby preventing the loss of the primary heat sink given the loss of startup feedwater (SFW) and the passive residual heat removal (RHR) heat exchanger (HX).

(7) Direct SG relief flow to the IRWST: To prevent fission product release from bypassing containment during a steam generator tube rupture (SGTR) event (or to reduce the amount released), flow from the SG safety and relief valves could be directed to the IRWST. An alternative, lower cost option would be to redirect flow only from the first-stage safety valve to the IRWST.

(8) Increased SG pressure capability: As an alternative to design alternative (7) above, another method could be used to prevent fission product release from bypassing containment during an SGTR event (or to reduce the amount). This alternative method would involve an increase of the SG secondary-side pressure capability and safety valve pressure setpoint to a level high enough to not allow an SGTR to cause the secondary-system safety valve to open. Although detailed analyses have not been performed, it is estimated that the secondary-side design pressure would have to be increased by several hundred pounds per square inch (psi).

(9) Secondary containment filtered ventilation: This design alternative involves the installation of a passive charcoal and high-efficiency particulate air filter system for the middle- and lower-annulus region of the secondary concrete containment (below elevation 135'-3"). Drawing a partial vacuum on the middle annulus via an eductor with motive power from compressed gas tanks would operate the filter system. This design alternative would reduce particulate fission product release from any failed containment penetrations.

(10) Diverse IRWST injection valves: In the current design, a squib valve in series with a check valve (CV) isolates each of the four IRWST injection paths. To provide diversity, a modification could be made to allow a different vendor to provide the valves in two of the lines. Such diverse IRWST injection valves would reduce the likelihood of CCFs of the four IRWST injection paths.

(11) Diverse containment recirculation valves: In both the AP600 and AP1000 designs, two of the four recirculation lines have a squib valve in series with a CV, and the remaining two recirculation lines have a squib valve in series with an MOV. This SAMDA involves changing the recirculation valve specification to enable two of the four lines to use diverse squib valves. To provide diversity, a modification could be made to allow a different vendor to provide the squib valves in two lines. Alternatively, in the AP1000 design, Westinghouse has specified that two of the four recirculation squib valves be designated as the low-pressure type and the remaining two squib valves as the high-pressure type. The diverse containment recirculation valves incorporated in the AP1000 design are responsive to the intent of this SAMDA and will reduce the frequency of core melt due to CCF of the four containment recirculation lines.

(12) Ex-vessel core catcher: This design alternative would inhibit core concrete interaction (CCI), even if the debris bed dries out. The enhancement would involve the design of a structure in the containment cavity or the use of a special concrete or coating. The current AP1000 design incorporates a wet cavity design in which ex-vessel cooling is used to keep core debris within the vessel. In cases where reactor vessel flooding has failed, the PRA assumes that containment failure occurs from an ex-vessel steam explosion or CCI.

(13) High-pressure containment design: A high-pressure containment design would prevent containment failures from severe accident phenomena such as steam explosions and hydrogen detonation. This proposed containment design would have a design pressure of approximately 2.17 MPa (300 psig) and would include a passive cooling feature similar to the one in the existing containment design. Although the high-pressure containment would not reduce the frequency or magnitude of releases from an unisolated containment, it would reduce the likelihood of containment failures.

(14) Increase reliability of diverse actuation system (DAS): The DAS is a non-safety system that can automatically trip the reactor and turbine and actuate certain engineered safety feature (ESF) equipment if the protection and safety monitoring system (PMS) is unable to perform these functions. The DAS provides diverse monitoring of selected plant parameters to guide manual operation and to confirm reactor trip and ESF actuations. Increasing the reliability of the DAS involves adding a third instrumentation and control (I&C) cabinet and a third set of DAS instruments to allow the use of two-out-of-three logic instead of two-out-of-two logic.

Westinghouse considered an additional SAMDA that would involve relocating the entire normal residual heat removal system (RNS) and piping inside the containment pressure boundary. This would prevent containment bypass due to intersystem loss-of-coolant accidents

(ISLOCAs) in the RNS. However, in the AP1000, the RNS has a higher design pressure than the systems in current pressurized-water reactors (PWRs), and an additional isolation valve is provided. As a result, ISLOCAs do not contribute significantly to the CDF in the AP1000 PRA. Accordingly, Westinghouse did not further investigate this change. The NRC has reviewed the Westinghouse analyses and agrees that further consideration of this change is not warranted because the change would provide virtually no risk reduction.

4.3 NRC Evaluation

The set of potential design improvements considered for the AP1000 is the same as those considered for the AP600. As part of the review for the AP600, the NRC reviewed the set of potential design improvements identified by Westinghouse and found it to be reasonably complete. The activity was accomplished by reviewing design alternatives associated with the following plants: Limerick, Comanche Peak, CE System 80+, Watts Bar, and the advanced boiling water reactor (ABWR). The NRC also reviewed accident management strategies described in (NUREG/CR-5474) and alternatives identified through the Containment Performance Improvement (CPI) Program (NUREG/CR-5567, -5575, -5630, and -5562). The results of this assessment are summarized in Appendix A to "Review of Severe Accident Mitigation Design Alternatives (SAMDAs) for the Westinghouse AP600 Design," Science and Engineering Associates, Inc., (SEA 97-2708-010-A;1, August 29, 1997). Given the similarity between the AP1000 and the AP600 design features and risk profile, the NRC considers this prior evaluation for the AP600 to be applicable to the AP1000 as well.

The NRC notes that the AP1000 design is less tolerant of equipment failures than the AP600 because the large LOCA success criterion for the AP1000 requires operation of two of two accumulators whereas only one of two accumulators is required for the AP600, and because the LOCA success criterion for the AP1000 requires operation of three of four automatic depressurization system (ADS) Stage 4 valves whereas only two of four ADS Stage 4 valves are required for the AP600. At the NRC's request, Westinghouse performed an evaluation of the two additional design alternatives:

(1) Larger accumulators: An increase in the size of the accumulators sufficient to change the large LOCA success criterion from two of two accumulators to one of two accumulators. Westinghouse estimates that the accumulator tanks would have to increase in size from 56.6 m^3 to 113.2 m^3 (2000 ft^3 to 4000 ft^3). This increase would likely require a change to the design of the direct vessel injection (DVI) piping subsystem and significant reanalysis of the DVI piping.

(2) Larger ADS Stage 4 valves: Increasing the size of the ADS Stage 4 (ADS-4) valves sufficient to change the LOCA success criterion from three of four valves to two of four valves. Westinghouse estimates that the valves would have to increase in size from 35.6 cm to 45.7 cm (14 in. to 18 in.) and that common fourth stage piping that connects to the hot leg would have to increase in size from 45.7 cm to 50.8 cm (18 in. to at least 20 in.). This increase would require a significant redesign of the squib valve and the ADS-4 piping, which in turn would impact the design of the reactor coolant loop piping. Such a redesign would necessitate additional confirmatory testing to verify that the behavior of the passive safety systems was not adversely impacted.

For both of these alternatives, Westinghouse estimated that the redesign and reanalysis cost of the changes would be significantly greater than the benefits of completely eliminating all severe accident risk for the AP1000. Therefore, these design changes were not pursued further.

Although Westinghouse's analysis omitted several design alternatives, in most instances these design alternatives are either already included in the AP1000 design or bounded in terms of risk reduction by one or more of the design alternatives that were included in Westinghouse's analysis. In some other cases, design alternatives were pertinent only to boiling-water reactors (BWRs). The NRC's review did not reveal any obvious additional design alternatives that should have been considered by Westinghouse. Westinghouse considered some of the potential design alternatives identified in the above references as appropriate for accident management strategies, rather than as design alternatives. The NRC notes that the set of design improvements is not all inclusive in that additional, perhaps less expensive design improvements could be postulated. However, the benefits of any additional modifications would not likely exceed the costs of the modifications evaluated. Also, the costs of alternative improvements are not expected to be less than the costs of the least expensive improvements evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The discussions in DCD Tier 2, Appendix 1B, do not provide Westinghouse's basis or process for screening the many possible design alternatives to arrive at the final list of 14. Although the information provided does not demonstrate that the search for design alternatives was comprehensive, the NRC's review of the more than 120 candidate design alternatives considered for the AP600 did not identify any new alternatives more likely to be cost-beneficial than those included in the AP1000 design alternative evaluations. The NRC notes that Westinghouse has incorporated several risk significant enhancements within the AP1000 design, as discussed in Section 19.4.3.1 of NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," (AP1000 FSER), and has considered potential design changes to improve the AP1000 success criteria. On this basis, the NRC concludes that the set of potential design improvements evaluated by Westinghouse is acceptable.

4.4 Risk Reduction Potential of SAMDAs

4.4.1 Westinghouse Evaluation

In its evaluation, Westinghouse assumed that each design alternative would work perfectly to completely eliminate all severe accident risk from evaluated internal, external, and shutdown events. This assumption is conservative, since it maximizes the benefit of each design alternative. The design alternative benefits were estimated on the basis of the reduction of risk expressed in terms of whole body person-rem per year received by the total population within a 80.5-km (50-mile) radius of the AP1000 plant site, as discussed in Section 19.4.2 of the AP1000 FSER.

Westinghouse used the cost-benefit methodology of NUREG/BR-0184 to calculate the maximum attainable benefit of completely eliminating all risk for the AP1000. This methodology includes consideration of replacement power costs. The applicant estimated the present worth of eliminating all risk to be $21,000. Even if the AP1000 CDF and large release frequency (LRF) were a factor of 10 higher, this value would only increase to about $200,000.

4.4.2 NRC Evaluation

NRC reviewed Westinghouse's bases for estimating the risk reduction for the various SAMDAs, and concluded that Westinghouse used bounding and conservative assumptions as the bases for the risk reduction estimates for each design alternative.

Westinghouse's risk reduction estimates are based on point-estimate (mean) values, and do not consider uncertainties in CDF or offsite consequences. Although this is consistent with the approach taken in previous design alternative evaluations, further consideration of these factors could lead to significantly higher risk reduction values, given the extremely small CDF and risk estimates in the baseline PRA. In assessing the risk reduction potential of design improvements for the AP1000, the NRC has based its evaluation on the applicant's risk reduction estimates for the various design alternatives, in conjunction with an assessment of the potential impact of uncertainties on the results. This assessment is discussed further in Section 19.4.6 of the AP1000 FSER and in Section 4.6 of this EA.

4.5 Cost Impacts of Candidate SAMDAs

4.5.1 Westinghouse Evaluation

DCD Tier 2, Section 1B.1.8, “Evaluation of Potential Improvements,” discusses capital cost estimates for the design alternatives evaluated by Westinghouse for the AP1000. DCD Tier 2, Table 1B-5, presents the results of the cost evaluations. The cost evaluations did not account for the costs of design engineering, testing, and maintenance for each design alternative. Including these costs would increase the overall costs and decrease the benefits of each alternative. Thus, the Westinghouse approach is conservative.

4.5.2 NRC Evaluation

As mentioned previously, the set of SAMDAs considered for the AP1000 is the same as the set considered for the AP600. As part of the AP600 review, the NRC compared the capital costs for the AP600 design alternatives with those evaluated for the ABWR and CE System 80+ designs. The purpose of this comparison was to determine the reasonableness of the cost estimates presented by the applicant. The design alternatives among the reactor designs, did not exactly match, so only rough comparisons were possible. Based on these comparisons, the NRC concluded that the cost estimates for the AP600 design alternatives are in reasonable agreement with the costs for roughly similar design alternatives evaluated for other plants. Given the similarity between the AP1000 and the AP600 design features and risk profile, the NRC considers this prior evaluation for the AP600 to be applicable to the AP1000 as well. This is reasonable, considering uncertainties in the cost estimates, and the level of precision necessary, given the greater uncertainty inherent on the benefit side with which these costs were compared.

4.6 Cost-Benefit Comparison

4.6.1 Westinghouse Evaluation

After considering the risk reduction potential and cost impact of the various SAMDAs, Westinghouse did a cost-benefit comparison to determine whether any of the potential severe accident design features would be justified. To do so, Westinghouse evaluated the benefits of each design alternative in terms of potential risk reduction, which was defined as the reduction in whole body person-rem per year received by the total population within a 80.5-km (50-mile) radius of the AP1000 plant site. Westinghouse used the cost-benefit methodology of NUREG/BR-0184 to calculate the maximum attainable benefit of completely eliminating all risk for the AP1000. This methodology includes consideration of replacement power costs. Westinghouse estimated the present worth of eliminating all risk to be $21,000. This value is an upper bound because in practice no design alternative, if implemented, would reduce the plant CDF to zero. Westinghouse also provided additional sensitivity analyses of the impacts of the following:

- a 3-percent discount rate rather than the 7-percent discount rate assumed in the base case
- a factor of 10 increase in the population dose used in the base case
- a more realistic reduction in CDF (i.e., each SAMDA reduces CDF by 50 percent rather than 100 percent, as assumed in the base case)
- a factor of 2 increase in the base case CDF
- a factor of 10 increase in the maximum attainable benefit

DCD Tier 2, Table 1B-4, summarizes the results for these cases. With the exception of the last sensitivity case, the calculated maximum attainable benefit was no more than $43,000. Even when the AP1000 CDF and LRF were increased by a factor of 10, the maximum attainable benefit of eliminating all risk for the AP1000 would only increased to about $200,000.

The applicant found that none of the 14 design alternatives and neither of the two additional alternatives related to the PRA success criteria would be cost beneficial. Only one alternative has an implementation cost close to $21,000, namely, SAMDA 3, self-actuating CIVs, which has an estimated cost of $33,000. All of the remaining alternatives have estimated implementation costs at least a factor of 20 greater than the maximum attainable benefit of $21,000. On this basis, the applicant concluded that only SAMDA 3 warranted further evaluation.

SAMDA 3 consists of improved containment isolation provisions on all normally open containment penetrations. The design alternative would involve either adding a self-actuating valve or enhancing the existing inside CIV to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Westinghouse noted that even if this SAMDA completely eliminated all releases associated with containment isolation failures (i.e., release category containment isolation (CI)) and reduced the CDF to zero, the benefit of the SAMDA would be on the order of $1000. More realistically, the CDF would not be impacted, and elimination of all containment isolation failures would only have a benefit on the order of $100. Thus, even the lowest cost SAMDA would not be cost beneficial.

On the basis of the cost-benefit comparison, the applicant concluded that no additional modifications to the AP1000 design were warranted.

4.6.2 NRC Evaluation

The applicant’s estimates of risk do not account for uncertainties either in the CDF or in the offsite radiation exposures resulting from a core damage event. The uncertainties in both of these key elements are fairly large because key safety features of the AP1000 design are unique and their reliability has been evaluated through analysis and testing programs rather than operating experience. In addition, the estimates of CDF and offsite exposures do not account for the added risk from earthquakes.

As part of the AP600 review, the NRC did detailed analyses to assess design alternative benefits, taking into account the uncertainties in estimated CDF, offsite releases of radioactive materials from a severe accident, and the effects of external events. Given the similarities between the AP1000 and AP600 design features and risk profiles and the sets of SAMDAs relevant to each design, the NRC considers this prior evaluation for the AP600, summarized below, to be applicable to the AP1000 as well.

The staff estimated the maximum benefits that could be achieved with the AP600 design alternatives, assuming that a design alternative can either completely eliminate all core damage events or completely eliminate offsite releases of radioactive materials in the event of a severe accident. The estimates of benefits were calculated using the NRC-developed FORECAST code (NUREG/CR-5595, Revision 1, "FORECAST: Regulatory Effects Cost Analysis Software Manual, Version 4.1," Science and Engineering Associates, Inc., July 1996). FORECAST allows the use of uncertainty ranges for all key parameters and provides a means for combining uncertainties in these parameters. For the purposes of estimating the maximum potential benefit from the AP600 design alternatives, the staff assumed that external events and accident sequences not yet accounted for in the PRA increased the reference CDF by two orders of magnitude (i.e., a factor of 100).

The results of the analysis indicated that design alternatives which prevent accidents (i.e., reduce the accident frequency to zero) are much more cost effective than design alternatives which reduce or eliminate offsite releases, but which have no effect on accident frequency. This is because of the fairly large benefits of averting onsite cleanup and decontamination costs and avoiding replacement energy costs. Neither of these costs are assumed to be impacted by design alternatives which do not reduce accident frequency. The staff divided the design alternatives into two groups: those that impact the CDF and those that impact containment performance, but not CDF. Benefits were estimated by taking the fractional reduction in risk for each design alternative (compared to the AP600 baseline risk as defined by the applicant) and applying that fraction to the mean benefits.

Design alternatives that were within a decade of meeting a benefit-cost criterion of $5000/person-rem were subjected to further probabilistic and deterministic considerations. None of the design alternatives had a cost-benefit ratio of less than $5000/person-rem. The only design alternatives which came within a decade of the $5000/person-rem criterion were SAMDA 10, diverse IRWST injection valves, and SAMDA 3, self-actuating CIVs. The NRC concludes, on the basis of further probabilistic and deterministic evaluations, that these design alternatives are not cost beneficial and need not be further pursued.

Given the similarities between the AP1000 and the AP600 design features and risk profiles and the sets of SAMDAs relevant to each design, the NRC considers the results of this prior evaluation for the AP600 to be applicable to the AP1000 as well. Accordingly, the NRC further evaluated these two SAMDAs for the AP1000, as discussed below.

4.7 Further Considerations

4.7.1 Self-Actuating Containment Isolation Valves

This design alternative would reduce the likelihood of containment isolation failure by adding self-actuating valves or enhancing the existing CIVs for automatic closure when containment conditions indicate a severe accident has occurred. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions. For example, a fusible link would melt in response to elevated ambient temperatures, venting the air operator of a fail-closed valve, thus providing the self-actuating function. This design alternative is estimated to impact releases from containment by less than 10 percent.

This improvement to the containment isolation capability would appear to be effective in reducing offsite releases for accidents involving external and internal events. The addition of this design alternative would impose minor operational disadvantages to the plant because the operations and maintenance staff would require some additional training. These automatic features would also require periodic testing to assure that they were functioning properly.

The most important question regarding this design alternative is whether it can be implemented for a cost of only $33,000. The cost estimate appears not to include the first-time engineering and qualification testing that would be required to demonstrate that the valve would perform its intended function in a timely and reliable manner. The costs of periodic testing and maintenance appear not to have been included. The NRC believes that the actual costs of this design alternative would be substantially higher than the applicant's estimate (by a factor of 10 or more) when all related costs are realistically considered. On the basis of the unfavorable cost-benefit ratio and the expectation that actual costs would be even higher than the applicant estimated, the NRC concludes that this design alternative is not cost beneficial and need not be further evaluated.

4.7.2 Diverse IRWST Injection Valves

In the current AP1000 design, a squib valve in series with a CV isolates each of four IRWST injection paths. This design alternative would reduce the likelihood of CCFs of IRWST injection to the reactor by utilizing diverse valves in two of the four lines. The complete elimination of the CCFs of IRWST injection squib valves would lead to a moderate (up to 10 percent) reduction of the at-power internal events CDF. In the absence of a comprehensive external events PRA for the AP1000 plant, it is difficult to estimate the effectiveness of this design alternative in reducing the risk from external events such as seismic events. However, it appears likely that failure to inject coolant to the reactor would remain a contributor to the CDF from external events, in which case diversity in the IRWST injection valves should help to reduce the risk from both external and internal events.

Alternate vendors are available for the CVs. However, it is questionable if CVs of different vendors would be sufficiently varied to be considered diverse unless the type of CV was changed from the current swing-disk check valve type to another type. The swing-disk type is preferred for this application and other types are considered less reliable.

Adding diversity to the injection line squib valves would require additional spares at the plant and some additional training for plant operations and maintenance staff, but would not appear to add significantly to the operational aspects of the AP1000. However, a greater issue concerns the availability and costs of acquiring diverse valves from a second vendor. Squib valves are specialized valve designs for which there are few vendors. The applicant claimed that a vendor might not be willing to design, qualify, and build a reasonable squib valve design for this application, considering that the vendor would only supply two valves per plant. The cost estimate for this design alternative assumes that a second squib valve vendor exists and that the vendor only provides the two diverse IRWST squib valves per plant. The cost estimate does not include the additional first-time engineering and qualification testing costs that will be incurred by the second vendor. The applicant estimated that those costs could be more than $1 million dollars. As a result, the applicant concluded that this design alternative would not be practicable because of the uncertainty in the availability of a second squib valve design/vendor and the uncertainty about the reliability of another type of CV. The NRC considers the rationale set forth by the applicant regarding the potential reductions in reliability and high costs associated with obtaining diverse valves to be reasonable. On the bases of these arguments, the NRC concludes that this design alternative need not be further pursued.

4.8 Conclusions

As discussed in Section 19.1 of the AP1000 FSER, Westinghouse used the PRA results extensively to arrive at the final AP1000 design. As a result, the estimated CDF and risk calculated for the AP1000 design are very low, both relative to existing operating plants and in absolute terms. Moreover, the low CDF and risk for the AP1000 plant reflect Westinghouse's efforts to systematically minimize the effect of initiators/sequences that have been important contributors to CDF in previous PWR PRAs. This minimization has been done largely through the incorporation of a number of design improvements. Section 19.1 of the AP1000 FSER discusses these improvements and the additional AP1000 design features which contribute to low CDF and risk for the AP1000.

Because the AP1000 design already has numerous plant features designed to reduce CDF and risk, the benefits and risk reduction potential of additional plant improvements is significantly reduced. This reduction is true for both internally and externally initiated events. Moreover, with the features already incorporated in the AP1000 design, the ability to estimate CDF and risk approaches the limitations of probabilistic techniques. Specifically, when CDFs are estimated to be on the order of 1,000,000 years, it is possible that the areas of the PRA where modeling is least complete, or supporting data are sparse or even nonexistent, may actually be the more important contributors to risk. Areas not modeled or incompletely modeled include human reliability, sabotage, rare initiating events, construction and design errors, and systems interactions. Although improvements in the modeling of these areas may introduce additional contributors to CDF and risk, the NRC does not expect that additional contributions would change the conclusions in absolute terms.

The NRC concludes that none of the potential design modifications evaluated are justified on the basis of cost-benefit considerations. The NRC further concludes that it is unlikely that any other design changes would be justified in the future on the basis of person-rem exposure because the estimated CDFs are very low on an absolute scale.

5.0 Alternative Use of Resources

No resources, such as land, water, or physical materials, will be affected by the promulgation of this proposed rule. This proposed rule would codify the AP1000 design in the *Code of Federal Regulations* but would not authorize the siting, construction, or operation of any nuclear power plant.

6.0 States Consulted and Sources Used

The NRC has sent a copy of the proposed rule and draft EA to the State Liaison Officers and requested their comments on the EA.

The Commission has determined under the NEPA of 1969, as amended, and the NRC's regulations in 10 CFR Part 51, Subpart A, that this rule is not a major Federal action significantly affecting the quality of the human environment. Therefore, the NRC has determined that preparation of an environmental impact statement for this rulemaking is not required. The basis for this determination, as documented in this EA, is that the amendment to 10 CFR Part 52 would not authorize the siting, construction, or operation of a facility referencing the AP1000 design; it would only codify the AP1000 design in a rule. Therefore, the NRC staff did not issue this EA for comment by Federal, State, and local agencies. The NRC's finding of no significant environmental impact was published in the *Federal Register* on xxxxxxxx, with the proposed design certification rule and draft EA for the AP1000 design. The NRC will evaluate the environmental impacts and issue an EIS as appropriate in accordance with NEPA as part of the application(s) for the siting, construction, or operation of a facility.

FINDING OF NO SIGNIFICANT IMPACT:

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has decided not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the proposed design certification rule. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents in ADAMS should contact the NRC PDR reference staff at 1-800-397-4209 or 301-415-4737 or send an e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 18th day of March, 2005

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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