FINAL ENVIRONMENTAL ASSESSMENT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

RELATING TO THE CERTIFICATION OF THE

U.S. ADVANCED BOILING WATER REACTOR DESIGN

DOCKET NO. 52-001

Attachment 2

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1.0 INTRODUCTION AND SUMMARY

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued a design certification for the U.S. advanced boiling water reactor (ABWR) design. Design certification is a rulemaking that amends Part 52 of Title 10 of the Code of Federal Regulations (10 CFR Part 52). To comply with the requirements of the National Environmental Policy Act of 1969 (NEPA), as amended, the NRC must consider the environmental impacts of issuing this amendment to 10 CFR Part 52. In addition, the NRC decided to consider severe accident mitigation design alternatives (SAMDAs) as part of this final environmental assessment (EA) to resolve SAMDAs for NEPA on a generic basis for the U.S. ABWR design. The EA for this rulemaking is contained herein and is prepared in accordance with NEPA and 10 CFR Part 51. This EA only addresses the environmental impacts of issuing a design certification for the U.S. ABWR and SAMDAs for the U.S. ABWR design. The environmental impacts of construction and operation of a facility at a particular site will be evaluated as part of the application(s) for siting, construction, and operation of that facility.

In an application dated September 29, 1987, the GE Nuclear Energy (GE) company applied for certification of the U.S. ABWR standard design by the NRC. The application was made in accordance with the procedures of 10 CFR Part 50, Appendix O, and the Policy Statement on Nuclear Power Plant Standardization, dated September 15, 1987. The application was docketed by the NRC staff on February 22, 1988 (Docket No. STN 50-605). On December 20, 1991, GE requested that its application be considered as an application for design approval and subsequent design certification pursuant to 10 CFR 52.45. Accordingly, the NRC staff assigned a new docket number (52-001) to the application on March 13, 1992.

The NRC has determined that the issuance of this design certification is not a major Federal action significantly affecting the quality of the human environment, and therefore, has decided not to prepare an environmental impact statement (EIS) in connection with this action. The finding of no significant impact is based on the fact that the certification rule itself would not authorize the siting, construction or operation of the U.S. ABWR design; it would only codify the U.S. ABWR design in a rule that could be referenced in a construction permit (CP), early site permit (ESP), combined license (COL), or operating license (OL) application. Further, because the action is a rule, there are no resources involved which would have alternative uses.

The NRC also reviewed, pursuant to NEPA, GE's evaluation of design alternatives to prevent and mitigate severe accidents. Based on the review, the NRC finds that the evaluation provides a sufficient basis to conclude that there is reasonable assurance that an amendment to 10 CFR Part 52 certifying the U.S. ABWR design will not exclude SAMDAs for a future facility that would have been cost beneficial had they been considered as part of the original design certification application. These issues are considered resolved for the U.S. ABWR design certification.

2.0 THE NEED FOR THE PROPOSED ACTION

The NRC has long sought the safety benefits of commercial nuclear power plant standardization, as well as the early resolution of design issues and finality of design issue resolution. The NRC plans to achieve these goals by certification of standard plant designs. Subpart B to 10 CFR Part 52 allows for certification by rule of an essentially complete nuclear plant design.

The proposed action would amend 10 CFR Part 52 to certify the U.S. ABWR design. The amendment would allow prospective applicants for a combined license (COL) under Part 52 or for a CP under Part 50 to reference the certified U.S. ABWR design. Those portions of the U.S. ABWR design included in the scope of the design certification would not be subject to further regulatory review or approval. In addition, the amendment would resolve the issue of consideration of SAMDAs for any future facilities that reference the U.S. ABWR design.

3.0 ALTERNATIVES TO THE PROPOSED ACTION

The alternatives to certifying the U.S. ABWR design in an amendment to 10 CFR Part 52 are either (1) no action approving the design or (2) issuing a final design approval (FDA), but not certifying the design. These alternatives in and of themselves would not have a significant impact affecting the quality of the human environment because they do not authorize the siting, construction, or operation of a facility.

In the first case, the design would not be approved. Therefore, a facility to be built as a U.S. ABWR would be required to be licensed under 10 CFR Part 50 or 10 CFR Part 52, Subpart C, as a custom plant application. All design issues would have to be considered as part of each application to construct and operate a U.S. ABWR facility at a particular site. This alternative would not achieve the benefits of standardization, provide early resolution of design issues, or provide finality of design issue resolution.

In the second case, the U.S. ABWR would be issued an FDA under 10 CFR Part 52, Appendix O, but the design would not be certified in a rulemaking. Therefore, although the NRC would have approved the design, the design could be modified and thus require re-evaluation as part of each application to construct and operate a U.S. ABWR facility at a particular site. This alternative would provide early resolution of issues, but would not achieve the benefits of standardization or provide finality of design issue resolution.

The NRC sees no advantage in either of the alternatives compared to the design certification rulemaking proposed for the U.S. ABWR. Although neither the alternatives nor the proposed design certification rulemaking would have a significant impact affecting the quality of the human environment in and of themselves, the rulemaking provides for standardization, as well as early resolution of design issues and finality of design issue resolution for design issues that are within the scope of the design certification, including SAMDAs. Therefore, the NRC concludes that the alternatives to rulemaking would not achieve the objectives the Commission intended by certification of the U.S. ABWR design pursuant to 10 CFR Part 52, Subpart B.

3.1 Severe Accident Design Alternatives

The Commission decided to evaluate design alternatives for severe accidents as part of the design certification for the U.S. ABWR design, consistent with its objectives of achieving early resolution of issues for the design and standardization. The Commission, in a 1985 policy statement, defined the term "severe accident" as those events which are "beyond the substantial coverage of design basis events" and includes those for which there is substantial damage to the reactor core whether or not there are serious offsite consequences. Design basis events are considered to be those analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the ABWR Design Control Document (DCD).

As part of its design certification application, GE performed a probabilistic risk assessment (PRA) for the ABWR design to (1) identify the dominant severe accident sequences and associated source terms for the design; (2) modify the design, based on PRA insights, to prevent or mitigate severe accidents and reduce the risk of severe accidents; and (3) provide a basis for concluding that all reasonable steps have been taken to reduce the chances of occurrence, and to mitigate the consequences, of severe accidents. GE's analysis is documented in Chapter 19 of the ABWR standard safety analysis report (SSAR).

In addition to considering alternatives to the rulemaking process as discussed in Section 3, applicants for reactor design approvals or CPs must also consider alternative design features for severe accidents based on (1) the requirements of 10 CFR Part 50 and (2) a court ruling relating to NEPA. These requirements can be summarized as follows:

- 10 CFR 50.34(f)(1)(i) requires the applicant to perform a plant/site specific probabilistic risk assessment, 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 include consideration of certain severe-accident-mitigation design alternatives (SAMDAs) in the environmental impact review performed under Section 102(2)(c) of NEPA as part of the OL application.

Although these two requirements are not directly related, the purpose is the same: to consider alternatives to the proposed design, to evaluate potential alternatives for improvements in the plant design for increased safety performance during severe accidents, and to prevent viable alternatives from being foreclosed. It should be noted that the Commission is not required to consider alternatives to the design in this EA on the rulemaking; however, as a matter of discretion, the Commission has determined that consideration of SAMDAs is consistent with the intent of 10 CFR Part 52 for early resolution of issues, finality of design issue resolution, and enhancing the benefits of standardization.
In its decision in *Limerick*, the Court of Appeals for the Third Circuit expressed its opinion that it was likely that evaluation of SAMDAs for NEPA purposes would be difficult to perform on a generic basis. However, the NRC has determined that generic evaluation of SAMDAs for the U.S. ABWR standard design is warranted because (1) the design and construction of all plants referencing the certified U.S. ABWR design will be governed by the rule certifying a single design, and (2) the site parameters specified in the rule and in the "Technical Support Document" (TSD) dated December 1994, establish the consequences for a reasonable set of SAMDAs for the ABWR. The low residual risk of the ABWR and limited potential for further risk reductions provides high confidence that additional cost beneficial SAMDAs wound not be found. Should the actual site parameters for a particular site exceed those assumed in the rule and the TSD, SAMDAs would have to be re-evaluated in the site-specific environmental report and EIS.

GE initially submitted its response to 10 CFR 50.34(f) in SSAR Section 19P as part its application for a final design approval (FDA) and subsequent design certification for the ABWR. The NRC issued an FDA for the ABWR in July 1994, and provided its evaluation of SSAR Section 19P in FSER Section 20.5.1. Subsequently, as part of its preparation of the DCD for the design certification rulemaking, GE updated and relocated Section

19P of the SSAR to Attachment A of the TSD for the ABWR" (see letter from J. Quirk (GE) to R.W. Borchardt (NRC), December 21, 1994). GE submitted the TSD to meet the Commission's requirement to consider SAMDAs as part of the design certification application.

3.2 Estimate of Risk for U.S. ABWR

In response to 10 CFR 50.34(f)(1)(i), GE provided an evaluation of the U.S. ABWR design improvements in SSAR Section 19P. GE's evaluation of risk was based on the risk-reduction potential for internal events only. The limited scope was a consequence of GE's use of alternative analyses for external events. The staff's evaluation of this approach to external events is in FSER Section 19.1.3. The staff's evaluation of design alternatives considering risk from external events is discussed in Section 3.5.5 of this EA.

Risk was defined in terms of person-Sieverts (Sv), and was calculated by multiplying the probability of an event per year by its consequences (the whole body exposure to the population within 50 miles of the release) over 60 years. GE used the CRAC2 code to estimate offsite consequences at five different sites, each representing a different geographic region of the U.S. Offsite consequences were calculated for each release class from the U.S. ABWR Level 2 probabilistic risk assessment (PRA) which contained accident progression analysis and source term analysis following the Level 1 PRA accident sequence analysis. The meteorological and population data were obtained from previously developed information contained in Sandia National Laboratories' "Technical Guidance for Siting Criteria Development" (NUREG/CR-2239, December 1986). The source terms were determined using the MAAP code for each of the release categories as discussed in Chapter 19 of the final safety evaluation report (FSER). The results of the five sets of consequence calculations were averaged together to represent a typical site in the U.S.

GE's estimate of the cumulative offsite risk to the population within 50 miles of the site appears in Table 1 of GE's TSD. GE calculated the total cumulative exposure from all analyzed accidents to be about 0.003 person-Sieverts (Sv) (0.3 person-rem) over a 60-year plant life. The extremely small level of risk calculated by GE is primarily due to the low estimated core-damage frequency for the U.S. ABWR (1.6 x 10^{-7} per reactor-year). This means that even if all core-damage accidents led to the worst release, on the basis of GE's core-damage frequency estimates for internal events, the total exposure would be only about 0.3 person-Sv (30 person-rem). The risk calculated in the analysis supported GE's conclusion that none of the design improvements beyond those already incorporated in the U.S. ABWR design are cost beneficial.

As a result of the low estimated core-damage frequency and associated risk levels for the U.S. ABWR, any modifications costing more than a few dollars would not be cost effective, even if the design modification totally eliminated the severe accidents or their consequences.

3.3 Identification of Potential Design Alternatives

GE's evaluation of potential design improvements in response to the requirements of 10 CFR 50.34(f)(1)(i) also gives a technical basis for the staff to evaluate the SAMDAs, as required by the Limerick decision. The staff's review of GE's evaluation is presented below.

By surveying previous industry- and NRC-sponsored studies of features to prevent and mitigate severe accidents, GE prepared a set of potential severe-accident design alternatives for the U.S. ABWR and developed a composite list of 68 potential design alternatives, organized into 14 categories. The list of potential design alternatives considered for the U.S. ABWR is presented in Table 2 of the TSD.

GE eliminated certain design alternatives from further consideration because they were not applicable to the U.S. ABWR (e.g., post accident inerting system, hydrogen control by venting), were considered as part of another alternative (e.g., diverse injection system, fuel cells), or were already incorporated in the design. Examples of design alternatives already included in the design were improved low-pressure injection system (fire pump), reactor water clean-up decay heat removal, low-flow vent (unfiltered), and combustible gas control (pre-inerted containment). These and additional U.S. ABWR design features that contribute to low core-damage frequency and risk for the U.S. ABWR design are discussed further in FSER Section 19.1. After this screening, 21 potential design alternatives applicable to the design, covering 12 of the 14 categories, remained for further consideration.

3.4 Description of Design Alternatives

The design alternatives selected by GE for cost-benefit evaluation are described in Sections A.3 and A.4 of the TSD. The design alternatives are summarized below.

(1) Emergency procedures guidelines (EPGs) and accident management guidelines (AMGs) for severe accidents — Expand the EPGs and emergency operating procedures (EOPs) to address arrest of a core melt, emergency planning, radiological release assessment, and other areas related to severe accidents. This modification would make manual actions in response to core-damage events more reliable.

(2) Computer-aided instrumentation — Apply expert system-based improvements to plant status monitoring, including human-engineered displays of important variables in the EPGs and AMGs, and displays of procedural options for operators to evaluate during severe accidents. This modification would make manual actions to prevent core damage more reliable.

(3) Improved maintenance procedures and manuals — Improve maintenance manuals and give more information about U.S. ABWR components important to reducing risk. These manuals and this information would make equipment important for preventing and mitigating accidents more reliable.

(4) Passive high-pressure system — Add an isolation condenser-type high- pressure system for removing decay heat from both the core and the containment. The modification would be equivalent to adding another reactor core isolation cooling (RCIC) system and containment heat removal system.

(5) Improved depressurization — Supply manually controlled, seismically protected air operators to permit manual reactor pressure vessel depressurization in the event of loss of dc

control power or control air events. Improved depressurization would reduce the threat of containment failure due to high-pressure melt ejection and allow more reliable access to low-pressure systems.

(6) Suppression pool jockey pump – Add a small, ac-independent makeup pump to allow low-pressure decay heat removal from the reactor pressure vessel (RPV) using suppression pool water as the source. This modification would have the same benefits as the ac-independent "fire-water" addition mode of residual heat removal (RHR), but without the associated long-term containment water inventory buildup concerns.

(7) Safety-related condensate storage tank (CST) – Upgrade the structure of the CST so that it could supply makeup water to the reactor after a large seismic event. This modification would enhance core injection capabilities in seismic events by giving an alternative to the suppression pool as a source of water for injection.

(8) Larger-volume containment – Increase the volume of containment by a factor of two. This modification would reduce the peak pressures associated with an energetic event, making drywell head failure less likely, and would reduce the rate of long-term containment pressurization, thereby delaying fission product release.

(9) Increased containment pressure capacity – Increase the ultimate pressure capacity of containment (including seals) to a level at which all release modes except normal containment leakage are eliminated.

(10) Improved vacuum breakers – Add a second vacuum breaker valve in each of the eight drywell-to-wetwell vacuum breaker lines to make these valves redundant. This modification would reduce the potential for suppression pool bypass due to stuck-open or leaking vacuum breaker valves.

(11) Improved bottom head penetration design – Change the transition piece (used to connect the stainless steel RPV drainline to the RPV) from carbon steel to a material with a higher melting point, such as Inconel. Also establish

external welds or restraints on the control rod drives external to the vessel so that the drives would not be ejected in the event the internal welds fail. This modification would delay reactor vessel failure by several hours, thereby increasing the potential to arrest core damage in vessel, but might also make the lower head more likely to fail grossly on overpressure.

(12) Larger-volume suppression pool — Increase the size of the suppression pool to reduce pool heatup rates. This modification would reduce the frequency of core melt from Class II sequences (loss of containment heat removal) and anticipated transients without scram (ATWS) sequences by giving operators more time to act and heat removal systems more time to recover.

(13) Low-flow filtered vent — Add a filter system external to the containment to further reduce the magnitude of radioactive releases via containment venting. The system would be similar to the multiple-venturi scrubbing systems in some plants in Europe. The system filters would scrub fission products better than the suppression pool at present, but would not affect releases due to drywell head failure and containment bypass sequences.

(14) Drywell head flooding — Provide an additional line to permit intentional flooding of the upper drywell head using the existing firewater addition system. Drywell head flooding would cool the drywell head seal, preventing its failure, and scrub fission products in the event of drywell head leakage. Instrumentation and controls to permit manual control from the control room to accomplish drywell head flooding were included in the evaluation as part of this modification.

(15) Additional service water pump — Add another service water cooling loop (pump and heat exchanger) to make the service water network more reliable. This loop could remove heat from any one of the three ECCS systems, making failure of injection due to loss of component cooling less frequent.

(16) Steam-driven turbine generator — Add a steam-driven turbine generator that uses reactor steam and exhausts to the suppression pool. This modification would reduce the frequency of station blackout sequences in the same way that adding another gas turbine generator would.

(17) Alternate pump power source — Add a separate diesel generator and supporting auxiliaries to power the feedwater or condensate pumps. This modification would remove the reliance of these pumps on offsite power and permit them to be used as a backup to the high-pressure core flooder (HPCF) and the low-pressure core flooder (LPCF).

(18) Dedicated dc power supply — Add a separate, diverse dc power source (fuel cell or separate battery) to supply a dc motor-pump combination for RPV and containment cooling. This modification would further reduce the risk from loss of offsite power and station blackout.

(19) ATWS-sized vent — Provide a wetwell vent line capable of passing the steam flow from an ATWS. The system would be significantly larger than the existing containment overpressure protection system (COPS) design and could be manually initiated from the control room. This system would prevent a containment overpressure failure in ATWS events thus preventing failure of other containment systems and thereby preventing core damage.

(20) Reactor building sprays — Modify the fire-water spray system in the reactor building to spray in areas vulnerable to fission product release. This modification would reduce the risk associated with releases into the reactor building, such as drywell head failures and containment bypass events, but would not affect releases via COPS.

(21) Flooded rubble bed — Provide a bed of refractory pebbles that would be flooded with water. The rubble bed would impede the flow of molten corium to the concrete drywell structures and increase the available heat transfer area, thereby enhancing debris coolability. This modification would further reduce the potential for core-concrete interactions in the U.S. ABWR. A major drawback of the

modification is that additional experimental testing would be necessary to validate the concept for the U.S. ABWR application.

The NRC staff has reviewed the set of potential design alternatives identified by GE in the TSD and finds the set to constitute a reasonable range of design alternatives. The list includes all alternatives identified in the NRC containment performance improvement (CPI) program and in the NRC review of SAMDAs for the Limerick Generating Station, that would be applicable to the U.S. ABWR. Although the list does not include one of the SAMDAs considered as part of the NRC's review of SAMDAs for Comanche Peak, namely, improved instrumentation for containment bypass sequences, this improvement would not significantly reduce risk potential for the U.S. ABWR since the level of residual risk is already low compared to operating plants and in absolute terms. The NRC notes that the set of design alternatives is not all inclusive, since additional, possibly even less expensive, design alternatives can always be postulated. However, the NRC concludes that the benefits of any additional modifications are unlikely to exceed the benefits of the modifications evaluated and that the alternative improvements would not likely cost less than the least expensive alternatives evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered. On this basis, the NRC concludes that the set of potential design alternatives identified by GE is acceptable.

3.5 Risk Reduction Potential of Design Alternatives

3.5.1 GE Evaluation of Risk Reduction Potential

GE used the estimated reduction in cumulative risk of accidents occurring during the life of the plant resulting from the above design changes to estimate the benefits of plant improvements. Estimates of risk reduction were developed by determining the approximate effect of each modification on the frequency of the various release classes in the probabilistic risk assessment (PRA). GE's basis for estimating the risk reduction for each design improvement is given in TSD Section A.4 and summarized in Table 1 of this EA.

The NRC staff has reviewed GE's bases for estimating how much the various design alternatives would reduce risks. The NRC staff notes that GE exercised considerable judgment in estimating the risk reduction potential but that, in general, the rationale and assumptions on which the risk reduction estimates are based (center column of Table 1) are reasonable and in many cases conservative (as described below, the NRC staff did not analyze individual SAMDA potential risk reduction, but made bounding assumptions). However, this is not to say that the estimates of person-Sv averted are conservative, because the staff does not completely agree with GE's characterization of baseline risk. For example, the risk reduction potential of improved vacuum breakers appears to be underestimated in GE's analysis. GE estimates that improved vacuum breakers (addition of a second vacuum breaker valve in series with each of the existing valves) would reduce risk by about 4×10^{-7} person-Sv (4×10^{-5} person-rem). This value is largely due to significant credit for fission-product removal by wetwell sprays (when available) and to the failure to consider the impact of the design improvement on bypass scenarios in which sprays are unavailable. GE's risk reduction estimate for this improvement would increase by at least three orders of magnitude if the latter factor were taken into account. Nevertheless, the risk reduction would remain small since the probability of the events involved is on the order of 1×10^{-10} per reactor-year.

3.5.2 Staff Evaluation of Risk Reduction Potential

In view of the extremely small residual risk for the U.S. ABWR, rather than separately assess risk-reduction potential of each U.S. ABWR design improvement, the NRC staff used a bounding assumption that each improvement would eliminate all of the risk for internal events for the U.S. ABWR (0.01 person-Sv (1 person-rem) for the 60-year plant life). This approach tends to overestimate the benefits of each individual SAMDA because the U.S. ABWR risk profile reflects contributions from several unique types of sequences (e.g., station blackout, containment bypass, loss-of-coolant accidents). An individual design improvement would generally reduce or eliminate some of these contributors but would not be effective on others. Moreover, many different modes of containment failure must be dealt with to ensure containment integrity in a severe accident. Thus, a carefully selected set of plant improvements would be needed, each one acting on particular components of risk, to effectively and significantly reduce total risk.

3.5.3 Costs of SAMDAs

GE determined the approximate costs for each design improvement. The costing methodology and assumptions are described in TSD Section A.1.3.1. The cost of each plant improvement is given in Table 4 of the TSD and in TSD Section A.5 on an item-by-item basis.

GE indicated that the cost estimates represent the incremental costs that would be incurred in a new plant, rather than costs incurred in backfit. GE also stated that it intentionally biased costs on the low side, but that it took all known or reasonably expected costs into account to arrive at a reasonable minimum cost.

For modifications that reduce core-damage frequency, GE reduced the costs of the design alternatives by an amount proportional to the reduction in the present worth of the risk of averted onsite costs. The onsite costs that were considered include replacement power at $0.013/kwh differential cost, direct accident costs including onsite cleanup at $2 billion, and the economic loss of the facility at $1.4 billion. The resulting costs for each of the design alternatives are given in Table 4 of the TSD.

The NRC staff reviewed the bases for GE's cost estimates and finds them acceptable. For certain alternatives, the NRC staff also compared GE's cost estimates with estimates developed elsewhere for similar alternatives, even though the bases for some of these cost estimates were different. The NRC staff considered the cost estimates developed as part of the evaluation of design alternatives for GESSAR II (NUREG-0979, Supplement 4) and the review of SAMDAs for Limerick and Comanche Peak (NUREG-0974 and -0775, respectively).

The NRC staff noted a number of inconsistencies in the cost estimates. For example, GE's cost estimates for improved vacuum breakers ($100,000), modified reactor building sprays ($100,000), and ATWS-sized vent ($300,000) were considerably less than expected, whereas the costs for SAMDAS such as improved bottom head penetration design ($750K) and flooded rubble bed (approximately $19 million) were much higher than expected. As explained in the sensitivity analysis in Section 3.5.5, none of the SAMDAs are within two orders of magnitude of being cost beneficial. Thus, even if those cost estimates that appear high were reduced by a factor of ten, the SAMDAs would still not be cost beneficial. Accordingly, the NRC staff has used GE's cost estimates in the cost/benefit comparison analysis below.

Only rough approximations of the costs of specific alternatives are possible at this time. Large uncertainties exist because detailed designs are not available and because experience with construction and licensing problems that could surface in this type of work is limited. However, even though the U.S. ABWR design is still in the design phase, relatively large costs are anticipated for many of the design alternatives, which would involve first-of-a-kind engineering and would need to be integrated into the existing design. In addition, the introduction of a new system initiates a series of related requirements such as incremental training, procedural changes, and possible licensing requirements. These are all legitimate costs and must be considered in a comprehensive cost estimate.

Therefore, the NRC staff considers GE's approximate cost estimates as adequate, given the uncertainties surrounding the underlying cost estimates, and the level of precision necessary given the greater uncertainty inherent on the benefit side, with which these costs were compared.

3.5.4 Cost/Benefit Comparison

GE compared costs and benefits to determine whether any of the potential severe accident design features were justifiable. GE's estimates of the cost per person-Sv (person-rem) averted for the various design alternatives are presented in Table 2 of this EA. The GE values are based on the risk-reduction estimates reported in Table 1 of this EA, whereas the NRC staff values are based on the conservative assumption that each design improvement would eliminate all of the residual risk (0.01 person-Sv (1 person-rem) over the 60-year plant life).

In accordance with former NRC practice (NUREG-3568), GE used a screening criterion of $100,000 per person-Sv ($1000 per person-rem) averted to determine whether any of the design alternatives could be cost effective. According to GE's evaluation as shown in Table 2, the potential cost per averted person-Sv ranges from about $170 million to $2 billion for the various suggested modifications, far exceeding the former $100,000 per person-Sv ($1000 per person-rem) criterion. On this basis, GE concluded that no additional modifications to the U.S. ABWR design are warranted.

The NRC staff agrees that none of the design alternatives are cost effective. The NRC staff notes that using the least expensive modifications (estimated to cost about $100,000), and conservatively assuming that all risk is averted (0.01 person-Sv (1 person-rem)), the resulting cost/benefit would be $10 million per person-Sv (i.e., $100,000/0.01 person-Sv = $10 million/person-Sv)($100,000/person-rem), which is well in excess of the $100,000 per person-Sv ($1000 per person-rem) criterion. Realistically, individual design alternatives only partly reduce the residual risk for the U.S. ABWR, resulting in a much higher cost/benefit ratio for even the most cost beneficial case.

Therefore, the NRC concludes that, because of the low residual risk for the U.S. ABWR and the $100,000 per person-Sv ($1000 per person-rem) criterion, none of the modifications evaluated would be cost effective.

3.5.5 Further Considerations

The NRC staff has reviewed the assumptions on which this conclusion is based and has considered the effect of uncertainties in estimating core-damage frequency, the use of alternative cost-benefit criteria, and the inclusion of external events within the scope of the analysis.

GE's uncertainty analyses for the Level 1 portion of the PRA (see FSER Section 19.1.3.2.5) showed the 95th-percentile core-damage frequency (CDF) to be 4.5×10^{-7} per reactor-year. This is higher by a factor of three than the mean value on which the cost-benefit analysis is based, but is still very low compared to operating plants (CDF range of 10^{-4}-10^{-5} per reactor-year) and in absolute terms. Even if the benefits of the various design alternatives were requantified on the basis of this upper bound value, none of the alternatives would become cost beneficial. This would remain the case even if the cost-benefit criterion was also increased by a factor of 10 to $1 million per person-Sv ($10,000 per person-rem) averted, since the most cost beneficial design alternative is still at least an order of magnitude greater than this criterion (e.g., cost/benefit = $0.1M/0.00060 person-Sv = $170 million per person-Sv averted).

If external events are included, the estimate of U.S. ABWR risk could be one or possibly two orders of magnitude higher than considered in this analysis. For example, considering the NRC staff review of GE's original seismic PRA, as documented in the draft SER, the total risk from internal and seismic events for the 60-year plant life would range from about 0.4 to 2 person-Sv (40 to 200 person-rem), depending on the site population. The values for the final U.S. ABWR design are actually somewhat less, since these estimates do not consider plant improvements incorporated in the design after the original PRA analysis, including upgrading the seismic capability of the diesel-driven firewater pump. However, even without taking credit for these features, the cost/benefit analysis would not justify incorporation of additional SAMDAs. Because most external event analyses submitted to the NRC show that seismic events dominate risk for external events, the NRC staff assessed the design alternatives using seismic risk as a bounding analysis for other external events, including fires and internal floods.

Even assuming the highest estimate of total risk (2 person-Sv (200 person-rem)) and complete elimination of all risk, any design modifications or combinations costing more than $200,000 would not be cost beneficial (2 person-Sv averted risk x $100,000/person-Sv = $200,000). (This assumption of complete elimination of all risk is very conservative as evidenced by GE's analysis, which shows that modifications estimated to cost less than $200,000 have a relatively low risk-reduction potential and would eliminate less than 10-percent of the residual risk.)

For the four design modifications costing less than $200,000, drywell head flooding appears to be the most cost beneficial at $170 million/person-Sv averted. Conservatively assuming a total residual risk of 2 person-Sv (200 person-rem) for the ABWR, drywell head flooding would have to eliminate 50-percent (1 person-Sv (100 person-rem)) or more of this risk to be considered cost beneficial. However, based on the analysis of internal events, drywell head flooding accounts for only a small reduction (a few percent) in risk. The risk reduction for external events is also expected to be small, since this modification affects only one of the numerous contributors to risk. This design improvement, therefore, would not be cost beneficial. Based on an inspection of Table 2 of this report, the other three design modifications also would not yield significant risk reductions and therefore would not be cost beneficial.

Since the draft EA was issued in April 1995, the NRC has issued "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (NUREG/BR-0058, Revision 2, November 1995). This policy document adopts a $2000 per person-rem conversion factor, subject to present worth considerations and is limited in scope to health effects. Limiting the conversion factor solely to health effects requires that the regulatory analysis include an additional dollar allowance for averted offsite property damage. By adopting the new $2000 per person-rem conversion factor and a $3000 per person-rem supplemental allowance for offsite property (see NUREG/CR-6349, "Cost benefit Considerations in Regulatory Analysis"), and assuming a base case 7% real discount rate as prescribed in NUREG/BR-0058, Revision 2, the present value of the health and safety benefits attributable to the Drywell Head Flooder approximate $233,000. This is a factor of about 1.2 times higher than the earlier $200,000 estimate. A comparable estimate for the health and safety benefits of this SAMDA based on a 3% real discount rate, which is recommended for sensitivity analysis purposes, is $460,000 or 2.3 times greater than the earlier $200,000 estimate. Given that the Drywell Head Flooder is estimated to cost on the order of $100,000, under either the 3% or 7% discount rate scenario, this design alternative would have to eliminate at least 22% or 43% respectively, of the total lifetime risk. Since the drywell head flooder is estimated to only account for less than 10% of the total risk, even for this most cost beneficial SAMDA, the total costs continue to be well in excess of the total benefits.

In summary, the NRC concludes that with the significant margins in the results of the cost-benefit analysis, consideration the new values provided in NUREG/BR-0058 would not change the findings of the analysis.

3.6 Conclusions

As discussed in FSER Chapter 19, GE has extensively used the results of a PRA to arrive at a final U.S. ABWR design. Based on the insights obtained from the PRA for the U.S. ABWR standard design, design features have been incorporated into the design to reduce risk, including risk from severe accidents. Consequently, the estimated core-damage frequency and risk calculated for the U.S. ABWR are very low both relative to operating plants and in absolute terms. The low core-damage frequency and risk for the U.S. ABWR reflects GE's efforts to systematically minimize the effect of initiators and sequences that have contributed to risk in previous BWR PRAs. GE has done so largely by incorporating a number of hardware improvements in the U.S. ABWR design. These include the provision of three separated divisions of the emergency core cooling system (ECCS), a diverse and independent combustion gas turbine capable of providing ac power to any of the three divisions, an ac-independent water addition system, and a fine-motion control rod drive system as a backup to the hydraulic drive system. Several additional design features have also been incorporated in the U.S. ABWR design to mitigate the consequences of a core-damage event, including inerting of the containment atmosphere, a lower drywell flooder system and a containment overpressure protection (vent) system, the use of basaltic concrete in the lower drywell, and an increased containment ultimate pressure capacity.

Because the U.S. ABWR design already includes numerous plant features to reduce core-damage frequency and risk, additional plant improvements would be unable to significantly reduce the risk of either internally or externally initiated events. For example, the U.S. ABWR seismic design basis (0.3 g safe-shutdown earthquake) has been shown to result in an ability to withstand earthquakes well beyond the design basis, as characterized by a high confidence with low probability of failure (HCLPF) value of at least 0.6 g. Moreover, with the features already incorporated in the U.S. ABWR design, the ability to estimate core-damage frequency and risk approaches the limitations of probabilistic techniques. Specifically, when core-damage frequencies of 1 in 100,000 or 1 million years are estimated in a PRA, the areas of the PRA where modeling is least complete or supporting data is sparse or even nonexistent could actually contribute most to risk. Areas not modeled or incompletely modeled include human reliability, sabotage, rare initiating events, construction or design errors, and systems interactions. Although improvements in the modeling of these areas may introduce additional contributors to core-damage frequency and risk estimates, the NRC staff does not expect that they would be significant in absolute terms.

In 10 CFR 50.34(f)(1)(i), the Commission requires the applicant to perform a plant- or site-specific probabilistic risk assessment, 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 NRC evaluated GE's response to this item in Section 20.5.1 of the FSER. In view of the foregoing, the NRC concludes that the PRA and GE's use of the insights of this study to improve the design of the U.S. ABWR meet this requirement for purpose of design certification pursuant to 10 CFR Part 52. The NRC concurs with GE's conclusion that none of the potential design modifications evaluated are justified on cost-benefit considerations. The NRC further concludes that any other design changes are unlikely to be justifiable on the basis of person-Sv exposure considerations because the estimated core-damage frequencies would remain very low on an absolute scale.

4.0 THE ENVIRONMENTAL IMPACT OF THE PROPOSED ACTION

The issuance of an amendment to 10 CFR Part 52 certifying the U.S. ABWR design would not constitute a significant environmental impact. The amendment would only codify the results of the NRC's review and approval of the U.S. ABWR design as defined in the FSER, dated July 1994 (NUREG-1503). Further, because the action is a rule, there are no resources involved that would have alternative uses.

In Section 3 of this EA the NRC reviewed alternatives to the design certification rulemaking and alternative design features related to the prevention and mitigation of severe accidents. Consideration of alternatives under NEPA were necessary for two reasons: (1) to show that the design certification rule is the appropriate course of action, and (2) to ensure that there are no cost-beneficial design changes relating to the prevention and mitigation of severe accidents that were excluded from the design, as codified in the design certification rule. The NRC concludes that the alternatives to design certification did not provide for resolution of issues as did the proposed design certification rulemaking.

This design certification rulemaking is in keeping with the Commission's intent in the Standardization and Severe Accident Policy Statements, and 10 CFR Part 52, to make future plants safer than the current generation plants, to achieve early resolution of licensing issues, and to enhance the safety benefits of standardization. Through its own independent analysis, the NRC also concludes that GE adequately considered an appropriate set of SAMDAs and none were found to be cost-beneficial. Although no design changes resulted from the SAMDAs review, GE did make changes to the U.S. ABWR design based on the results of the PRA. These changes were related to severe accident prevention and mitigation, but were not considered in the SAMDA evaluation because they were already part of the design. See FSER Section 19.1.3.2.2, "PRA as a Design Tool."

The certification rule by itself would not authorize the siting, construction, or operation of an U.S. ABWR design nuclear power plant. The issuance of a CP, ESP, COL, or OL for the U.S. ABWR design will require a prospective applicant to address the environmental impacts of construction and operation at a specific site. At that time, the NRC will evaluate the environmental impacts and issue an EIS in accordance with NEPA. The SAMDAs analysis for the U.S. ABWR, however, has been completed as part of this EA and will not need to be to be evaluated again as part of an EIS related to siting, construction, or operation.

5.0 AGENCIES AND PERSONS CONSULTED, AND SOURCES USED

The NRC concludes that design certification rulemaking does not result in a significant environmental impact because the action does not authorize the construction and operation of a facility at a particular site. Therefore, the NRC staff did not issue this EA for comment by Federal, State, and local agencies. However, the NRC's finding of no significant environmental impact, was published in the Federal Register on April 7, 1995, with the proposed ABWR design certification rule and there were no comments received related to this EA.

The sources for this EA include the "Technical Support Document for the ABWR," Revision 1, December 1994 (Attachment to a letter, J.F. Quirk (GE) to R.W. Borchardt (NRC), December 21, 1994); GE's U.S. "ABWR Standard Safety Analysis Report," as amended, July 1994; and the NRC's "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design" (NUREG-1503, Volumes 1 and 2), July 1994.

6.0 FINDING OF NO SIGNIFICANT IMPACT

The Director, Office of Nuclear Reactor Regulation (NRR), has determined under the National Environmental Policy Act 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, and therefore, an EIS not required.

The basis for the determination, as documented in this final EA, is that the amendment to 10 CFR Part 52 would not authorize the siting, construction, or operation of a facility using the U.S. ABWR design; it would only codify the U.S. ABWR design in a rule. 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.

In addition, as part of this final EA, the NRC reviewed, pursuant to NEPA, GE's evaluation of various design alternatives to prevent and mitigate severe accidents that was submitted in GE's "Technical Support Document for the ABWR." The Director of NRR finds that GE's evaluation provides a sufficient basis to conclude that there is reasonable assurance that an amendment to 10 CFR Part 52 certifying the U.S. ABWR design will not exclude a severe accident design alternative for a facility referencing the certified design that would have been cost beneficial had it been considered as part of the original design certification application. The evaluation of these issues under NEPA is considered resolved for the U.S. ABWR design.

Table 1 Summary of GE's Assessment of Risk Reduction for Candidate Design Improvements

POTENTIAL ABWR DESIGN MODIFICATION	GE's BASIS FOR ESTIMATING RISK REDUCTION	PERSON-SV (PERSON- REM) AVERTED
Accident Management		
Severe accident EPGs/AMGs	10% reduction in failure rates for manually initiated mitigative actions	0.00015 (0.015)
Computer-aided instrumentation	10% reduction in failure rates for manually initiated preventive actions	0.00010 (0.01)
Improved maintenance procedures/manuals	10% improvement in reliability of HPCF, RCIC, RHR, LPCF	0.00016 (0.016)
Decay Heat Removal		
Passive high-pressure system	Equivalent to adding a diverse RCIC and RHR system with 10% unavailability	0.00069 (0.069)
Improved depressurization system	Factor of 2 reduction in depressurization failure rates	0.00042 (0.042)
Suppression pool jockey pump	10% improvement in reliability of low-pressure makeup (resulting in 2% reduction in core damage frequency from low-pressure sequences	0.00002 (0.002)
Safety-related condensate storage tank	Engineering judgement	0.00010 (0.01)
Containment Capability		
Larger-volume containment	Elimination of all containment release modes involving drywell head failure (Cases 3, 6, 7, 8, 9)	0.00150 (0.15)
Increased containment pressure capacity	Elimination of all containment release modes except normal containment leakage	0.0016 (0.16)
Improved vacuum breakers	Elimination of releases from Release Class 2	0.0000004 (0.00004)
Improved bottom head penetration design	Factor of 2 increase in the probability of arresting core damage in vessel	0.00057 (0.057)
Containment Heat Removal		
Larger-volume suppression pool	Elimination of Class II Sequences	0.000002 (0.0002)
Containment Mass Removal		
Low-flow filtered vent	Elimination of the risk associated with releases via COPS	0.00014 (0.014)
Containment Spray Systems		
Drywell head flooding	Elimination of drywell head overtemperature failures and reduction in releases from drywell head overpressure failures	0.00060 (0.06)

Prevention Concepts Additional service water Loop	10% increase in reliability of HPCF, RCIC, RHR, LPCF	0.00016 (0.016)
AC Power Supplies Steam-driven turbine generator Alternate pump power source	 80% reduction in the diesel generator common-mode failure rate Equivalent to adding a diverse RCIC system	 0.00052 (0.052) 0.00069 (0.069)
DC Power Supplies Dedicated dc power supply	Factor of 10 increase in RCIC availability in LOOP and SBO sequences	0.00069 (0.069)
ATWS Capability ATWS-sized vent	Elimination of risk from ATWS (Case 9)	0.00030 (0.03)
System Simplification Reactor building sprays	10% reduction in risk from releases through the reactor building	0.00017 (0.017)
Core Retention Devices Flooded rubble bed	Elimination of sequences with core concrete interactions, except those with failure of containment heat removal (1% of Cases 1, 6, and 7)	0.000010 (0.001)

Table 2

Potential Design Improvements and Associated Costs (GE)

Modification		Estimated Cost ($M)	Person-Sv (Person-Rem) Averted	Cost($M)/ Person-Sv (Person-Rem) Averted
1.	**Accident Management**			
1a.	Severe accident EPGs	0.60	0.00015 (0.015)	4,000 (40)
1b.	Computer-aided instrumentation	0.60	0.00010 (0.01)	>4,000 (>40)
1c.	Improved maintenance procedures and manuals	0.30	0.00016 (0.016)	1,870 (18.7)
2.	**Decay Heat Removal**			
2a.	Passive high-pressure system	1.7	0.00069 (0.069)	2,530 (25.3)
2b.	Improved depressurization	0.60	0.00042 (0.042)	1,430 (14.3)
2c.	Suppression pool jockey pump	0.12	0.00002 (0.002)	>4,000 (>40)
2d.	Safety-related condensate storage tank	1.0	0.00010 (0.01)	>4,000 (>40)
3.	**Containment Capability**			
3a.	Larger-volume containment	8.0	0.00150 (0.15)	>4,000 (>40)
3b.	Increased containment pressure capacity	12.0	0.0016 (0.16)	>4,000 (>40)
3c.	Improved vacuum breakers	0.10	0.0000004 (0.00004)	>4,000 (>40)
3d.	Improved bottom head penetration design	0.75	0.00057 (0.057)	1,320 (13.2)
4.	**Containment Heat Removal**			
4a.	Larger-volume suppression pool	8.0	0.000002 (0.0002)	>4,000 (>40)
5.	**Containment Atmosphere Mass Removal**			
5.a	Low-flow filtered vent	3.0	0.00014 (0.014)	>4,000 (>40)
7.	**Containment Spray Systems**			
7a.	Drywell head flooding	0.10	0.00060 (0.06)	170 (1.7)
8.	**Prevention Concepts**			
8a.	Additional service water loop	6.0	0.00016 (0.016)	>4,000 (>40)
9.	**AC Power Supplies**			
9a.	Steam driven turbine generator	6.0	0.00052 (0.052)	>4,000 (>40)
9b.	Alternate pump power source	1.2	0.00069 (0.069)	1,730 (17.3)
10.	**DC Power Supplies**			
10a.	Dedicated RHR dc power supply	3.0	0.00069 (0.069)	>4,000 (>40)

11.	**ATWS Capability**			
11a.	ATWS-sized vent	0.30	0.00030 (0.03)	1,000 (10)
13.	**System Simplification**			
13a.	Reactor building sprays	0.10	0.00017 (0.017)	590 (5.9)
14.	**Core Retention Devices**			
14a.	Flooded rubble bed	18.8	0.00001 (0.001)	>4,000 (>40)