# US-APWR

# APPLICANT'S ENVIRONMENTAL REPORT - STANDARD DESIGN CERTIFICATION

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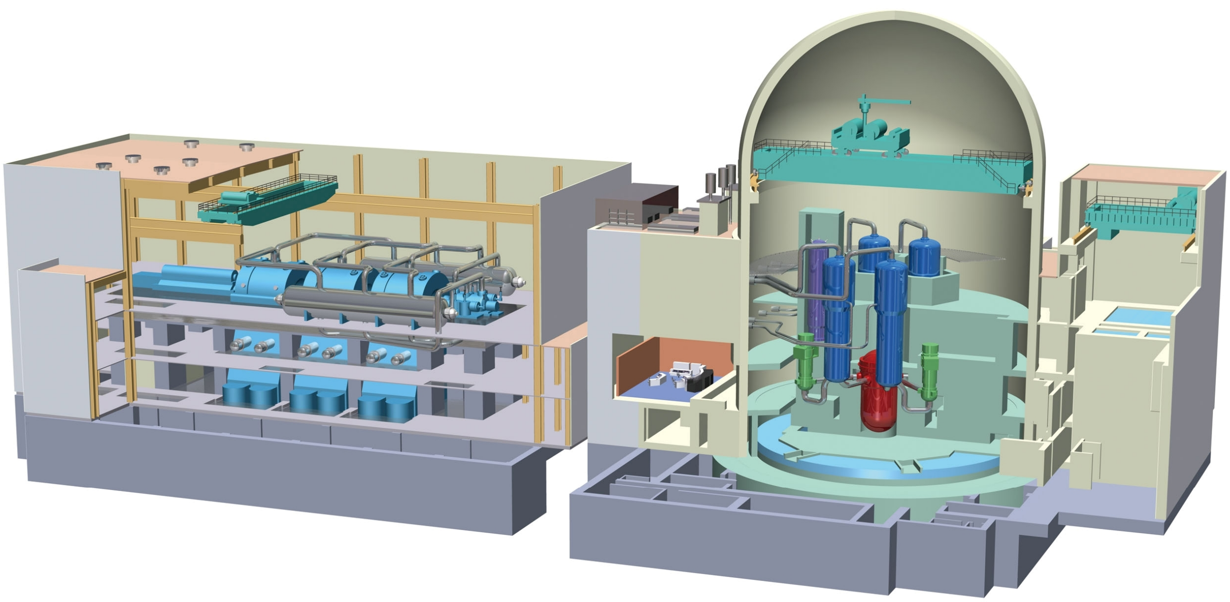

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## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| A/B | auxiliary building |
| AAC | alternate alternating current |
| ac | alternating current |
| ATWS | anticipated transient without scram |
| CCDF | complementary cumulative distribution function |
| CCDP | conditional core damage probability |
| CCW | component cooling water |
| CCWS | component cooling water system |
| CDF | core damage frequency |
| COL | Combined License |
| CPET | containment phenomenological event tree |
| CS/RHR | containment spray/residual heat removal |
| CSET | containment system event tree |
| CSS | containment spray system |
| dc | direct current |
| DCD | Design Control Document |
| ECCS | emergency core cooling system |
| EDG | emergency diesel generator |
| EFW | emergency feedwater |
| EFWS | emergency feed water system |
| ESW | essential service water |
| ESWS | essential service water system |
| FSS | fire protection water supply system |
| GTG | gas turbine generator |
| HPI | high pressure injection |
| HX | heat exchanger |
| ISLOCA | interfacing system loss-of-coolant accident |
| LOCA | loss-of-coolant accident |
| LOCCW | loss of component cooling water |
| LOOP | loss of offsite power |
| LPSD | low-power and shutdown |

**ACRONYMS AND ABBREVIATIONS (Continued)**

| | |
|---|---|
| LRF | large release frequency |
| M/D | motor-driven |
| MCCI | molten core concrete interaction |
| MLOCA | medium pipe break LOCA |
| NEI | Nuclear Energy Institute |
| NEPA | National Environmental Policy Act |
| NPP | nuclear power plant |
| NRC | U.S. Nuclear Regulatory Commission |
| PDS | plant damage state |
| PRA | probabilistic risk assessment |
| R/B | reactor building |
| RC | release category |
| RCP | reactor coolant pump |
| RCS | reactor coolant system |
| RV | reactor vessel |
| RWSP | refueling water storage pit |
| SAMA | severe accident mitigation alternative |
| SAMDA | severe accident mitigation design alternative |
| SBLOCA | small break loss of coolant accident |
| SBO | station blackout |
| SIS | safety injection system |
| SLBO | steam line break/leak outside containment |
| SLOCA | small pipe break LOCA |
| SW | service water |
| T/B | turbine building |
| T/D | turbine driven |

# 1. INTRODUCTION

This document provides an evaluation of severe accident mitigation design alternatives (SAMDA) for the US-APWR design. This evaluation is performed to address the costs and benefits of severe accident mitigation design alternatives, and the bases for not incorporating severe accident mitigation design alternatives in the US-APWR design. This document has been developed in accordance with applicable regulatory requirements as identified below.

The National Environmental Policy Act (NEPA), Section 102.(C)(iii) requires, in part, that:

> ... all agencies of the Federal Government shall ... (C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on ... (iii) alternatives to the proposed action.

10 CFR 52.47(b)(2) requires the submittal of an environmental report as required by 10 CFR 51.55.

10 CFR 51.55 requires each applicant for a standard design certification to submit with its application a separate document entitled, "Applicant's Environmental Report—Standard Design Certification." The environmental report must address the costs and benefits of severe accident mitigation design alternatives, and the bases for not incorporating severe accident mitigation design alternatives in the design to be certified.

10 CFR 50.34(f)(1)(i) also requires performance of a plant/site specific probabilistic risk assessment (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.

In a 1985 policy statement, the U.S. Nuclear Regulatory Commission (NRC) 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 . While design-basis events are considered to be those analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800), severe accidents are considered in a PRA analysis. Accordingly, the US-APWR PRA has been prepared to achieve the following objectives:

1. Identify the dominant severe accident sequences
2. Modify the design, on the bases of PRA insights, to prevent or mitigate severe accidents and reduce the risk of severe accidents.
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.

For the US-APWR, an evaluation of potential design improvements, or SAMDAs, was performed to meet these objectives. The information provided in this report complies with applicable parts of NUREG-1555(Reference1).

## 2. SUMMARY

The cost-benefit methodology follows the current guidance for regulatory analysis contained in NUREG/BR-0184 and NUREG/BR-0058 (References 2 and 3). Industry implementation guidance (NEI 05-01, Rev. A) is applied to identify and screen SAMDAs (Reference 4). Review of potential design alternatives will consider those of current PWR plant designs, PRA information on US-APWR, and design alternatives identified by US-APWR design personnel. Both onsite and offsite costs will be included in a manner consistent with SECY-99-169 (Reference 5).

There are five principal component costs to be considered using the NRC handbook methodology when the proposed action changes either accident frequencies or consequences.

- Offsite Exposure Cost
- Onsite Exposure Cost
- Offsite Property Damage
- Cleanup and Decontamination Cost
- Replacement Power Cost

The total maximum averted cost is the sum of these five component costs. It represents the maximum averted cost risk if risks are eliminated. Cost results are summarized in Table 1 for the four categories of analyzed events. The total averted cost is $289k.

A ranking of categories of events to this cost benefit is as shown:

- Internal fire, at power (39%)
- Internal flooding, at power (30%)
- Internal events, at power (26%)
- Shutdown events (4%).

A ranking by type of cost is as follows:

- Replacement power cost (65%)
- Cleanup and Decontamination costs (24%)
- Offsite Exposure cost (10%)
- Onsite Exposure cost (1%)
- Offsite Property Damage (<1%).

# 3. SELECTION AND DESCRIPTION OF SAMDAS

## 3.1 Screening Method

The approach for identifying potential design improvements followed NEI 05-01, Rev. A (Reference 4). SAMDA candidates were selected primarily from two sources: screening from the NEI 05-01 for pressurized water reactors (PWRs, Table 14); and US-APWR specific candidates considering the design and insights from the core damage frequency (CDF) and population dose risk profile. The process used for SAMDA identification follows Section 5 of NEI 05-01, and resulted in the SAMDA candidate list shown in Table 2 of 156 SAMDA candidates.

Two phases of evaluation were performed with the first being a Phase I qualitative screening analysis following section 6 of NEI 05-01. This screening was done to eliminate SAMDAs from further consideration, and was done to reduce the number of SAMDA for which quantitative analysis in a later phase (Phase II) was necessary.

## 3.2 Screening Criteria

The following are screening criteria identified in NEI 05-01 and applied for the US-APWR design.

- **Not applicable:** If a SAMDA candidate does not apply to the plant design, it is not retained. For example, installation of accumulators for turbine-driven feedwater pump flow control valves would not require further analysis at a plant with motor operated turbine-driven feedwater pump flow control valves.
- **Already implemented:** If a SAMDA candidate has already been implemented at the plant, it is not retained. For example, installation of motor generator set trip breakers in the control room to reduce the frequency of core damage due to an anticipated transient without scram (ATWS) would not require further analysis at a plant with a control room actuated diverse scram system.
- **Not a design alternative:** These candidates do not involve a design change. Typically, an item of this type is a Combined License (COL) action item.
- **Combined:** If a SAMDA candidate is similar in nature and can be combined with another SAMDA candidate to develop a more comprehensive or plant-specific SAMDA candidate, only the combined SAMDA candidate is retained. For example, addition of an independent reactor coolant pump seal injection system and use of an existing hydro test pump for reactor coolant pump seal injection provide similar risk-reduction benefits. If the lower-cost alternative is not cost-beneficial, the higher-cost alternative also will not be cost-beneficial. Therefore, the higher-cost alternative would not require further analysis.
- **Excessive implementation cost:** If a SAMDA requires extensive changes that will obviously exceed the maximum benefit, even without an implementation cost estimate, it is not retained. For example, the cost of installing an additional, buried off-site power source would exceed the maximum benefit and would not require further analysis.
- **Very low benefit:** If a SAMDA from an industry document is related to a non-risk

significant system for which change in reliability is known to have negligible impact on the risk profile, it is not retained. For example, if the instrument air system is not a risk-significant system at the plant, and failure of the air compressors is not on the PSA importance list, the plant risk profile would be unchanged if the air compressors were made perfectly reliable. Therefore, an improvement to replace the current air compressors with a more reliable model would not require further analysis.

The Phase I screening was performed in the two right-hand columns of Table 2. The list of 156 items was analyzed to determine if there are cost-beneficial design alternatives what should be considered for the US-APWR. The screening analysis identified 20 alternatives that are not applicable. There were 22 design alternatives that were already incorporated into the US-APWR design. Twenty-nine (29) items were screened since they are not design alternatives. There are 3 items that were not feasible because their cost would clearly outweigh any risk-benefit consideration. In addition, there were 3 items that were similar in nature to other items and have been combined. Finally, there were 69 Issues that were considered to have very low benefit due to their insignificant contribution to reducing risk. In summary, of the 156 total items analyzed, 10 were not screened out using the previously mentioned screening criteria.

The 10 SAMDAs for which a Phase II, quantitative cost analysis was performed are:

1. Provide additional direct current (dc) battery capacity (At least one train emergency dc power can be supplied more than 24 hours.)
2. Provide an additional diesel generator (At least one train emergency alternating current (ac) power can be supplied more than 24 hours.)
3. Install an additional, buried off-site power source
4. Provide an additional high pressure injection pump with independent diesel (With dedicated pump cooling)
5. Add a service water pump (Add independent train)
6. Install an independent reactor coolant pump seal injection system, with dedicated diesel (With dedicated pump cooling)
7. Install an additional component cooling water pump (Add independent train)
8. Add a motor-driven feedwater pump (With independent room cooling)
9. Install a filtered containment vent to remove decay heat
10. Install a redundant containment spray system (Add independent train)

SAMDA cost evaluation results are described with their assumptions in Table 3. The lowest cost SAMDA is SAMDA #10, Install a redundant containment spray system, at $870k. The second lowest cost SAMDA is SAMDA #4, Provide an additional high pressure injection pump with independent diesel, at $1,000k.

# 4. METHODOLOGY

## 4.1 Risk Reduction Potential of Design Improvements

Guidance contained in NUREG/BR-0184 (Reference 2) and NEI 05-01, Rev. A (Reference 4) provide the methodology for value-impact (benefit-cost) analysis, which is a central part of regulatory analysis. Values and impacts are characterized in monetary terms when feasible. The analysis balances benefits (values) with costs (impact) related to a proposed NRC action.

There are five principal component costs considered using the NRC handbook methodology when the proposed action changes either accident frequencies or consequences.

- Offsite Exposure Cost
- Onsite Exposure Cost
- Offsite Property Damage
- Cleanup and Decontamination Cost
- Replacement Power Cost

The proposed design modifications evaluated in this section are associated with Severe Accident Mitigation Design Alternatives (SAMDA). The estimation of the component costs are found in the probabilistic risk assessment (PRA) for US-APWR. The output data from the risk assessment that serve as input data to the value-impact analysis of this section are presented in Table 4. The risk assessment covers four categories of events: (1) internal events; (2) internal fire; (3) internal flood; and (4) low-power and shutdown (LPSD).

## 4.2 Algorithms for Calculating Value of Risk Avoided

### 4.2.1 Offsite Exposure Cost

For nuclear power plants, expected changes in radiation exposure should be measured over a 50-mile radius from the plant site. The product of the radiation exposure expressed in annual offsite exposure (person-rem/year) and the monetary conversion factor (R) is 2000 (dollars/person-rem) gives the monetary equivalent risk value (dollars/year). This value is converted to present-day dollars by a factor (C) that discounts future losses to the present value as shown below.

$C = [1-e^{(-rtf)}]/r$

Where:

C = present-value discount factor

r = real discount rate

$t_f$ = years remaining until end of facility life (years)

$Z_{pha} = D_{pa} \times R$

Where:

$Z_{pha}$ = monetary value of public health (accident) risk per year before discounting (dollars/year)

$D_{pa}$ = annual offsite exposure from risk assessment (person-rem/year)

R = monentary equivalent of unit dose (dollars/person-rem)

$W_{pha} = C \times Z_{pha}$

Where:

$W_{pha}$ = monetary value of public health risk after discounting (dollars/year)

### 4.2.2 Offsite Property Damage

Offsite property damage is based on an estimate that covers costs related to emergency response and long-term protective action that are associated with a 50-mile radius from the plant site. This value is converted to present-day dollars by the factor (C) that discounts future losses to the present value.

$D = E \times C$

Where:

D = present value of offsite property damage (dollars/year)

E = undiscounted cost of offsite property damage (dollars/year)

### 4.2.3 Onsite Exposure Cost

The estimation of occupation health costs involves evaluating costs associated with both immediate and long-term exposures. Immediate exposures occur during the accident and immediate management phase of the emergency. Long-term exposure occurs during the phase of cleanup and refurbishment or decommissioning of the damaged facility. The product of the accident dose (person-rem/event) and accident frequency (event/year) provides the risk estimate. The same monetary conversion factor (R) of 2000 (dollars/person-rem) that was used in the offsite exposure cost analysis is used for the onsite analysis to provide the monetary equivalent risk value. Immediate and long-term costs are converted to present-day dollars by the factor (C) that discounts future losses to the present value and summed.

Immediate dose:

$W_{IO} = R \times CDF \times D_{IO} \times C$

Where:

$W_{IO}$ = monetary value of accident risk avoided due to immediate doses, after discounting (dollars/year)

CDF = core damage frequency (events/year)

$D_{IO}$ = immediate occupational dose (person-rem/event)

<u>Long-term dose</u>:

$W_{LTO} = R \times CDF \times D_{LTO} \times C \times \{[1-e^{(-rm)}]/rm\}$

Where:

$W_{LTO}$ = monetary value of accident risk avoided due to long-term doses, after discounting (dollars/year)

$D_{LTO}$ = long-term occupational dose (person-rem/event)

m = years over which long-term doses accrue (e.g., cleanup period)

<u>Total dose</u>:

$W_O = W_{IO} + W_{LTO}$

Where:

$W_O$ = total monetary value of accident risk avoided, after discounting (dollars/year)

### 4.2.4 Cleanup and Decontamination Cost

Cleanup and decontamination costs of a nuclear facility (and possibly auxiliary building) are based on postulated physical damage and contamination levels from a medium or severe accident. Costs are integrated over the life of reactor power and converted to present-day dollars by the factor (C) that discounts future losses to the present value. Multiplication by the accident frequency provides the risk-based cost estimate for cleanup and decontamination.

$PV_{CD} = C_{CD} \times \{[1-e^{(-rm)}]/rm\}$

Where:

$PV_{CD}$ = net present value of cleanup and decontamination for a single event (dollars)

$C_{CD}$ = total undiscounted cost for single accident with constant year basis (dollars)

$U_{CD} = PV_{CD} \times C$

Where:

$U_{CD}$ = net present value of cleanup and decontamination over the life of the facility (dollars)

$W_{CD} = U_{CD} \times CDF$

Where:

$W_{CD}$ = cleanup and decontamination costs (dollars/year)

### 4.2.5 Replacement Power Cost

Following a severe accident, replacement power costs are considered for the remaining reactor lifetime. The single event costs are adjusted to account for all years of reactor service. Multiplication by the accident frequency provides the risk-based cost estimate for replacement power.

$PV_{RP} = B \times [1\text{-}e^{(-rtf)}]^2/r$

Where:

$PV_{RP}$ = net present value of replacement power for a single event (dollars)

$B$ = a constant representing a string of replacement power costs that occur over the lifetime of a reactor after an event (dollars). For a 910 MWe "generic" reactor NUREG/BR-0184 uses a value of 1.2E+08. For the analyzed reactor system, this value is scaled upward for a 1610 MWe reactor {i.e., B = 1.2E+08 x (1610/910)}.

$U_{RP} = PV_{RP} \times [1\text{-}e^{(-rtf)}]^2/r$

Where:

$U_{RP}$ = net present value of replacement power over life of facility (dollars)

$W_{RP} = U_{RP} \times CDF$

Where:

$W_{RP}$ = replacement power costs (dollars/year)

In the present analysis, the assumption is made that the population dose risk from internal events at power is applicable to internal fire events at power, internal flooding events at power, and shutdown events. A CDF scaling factor is applied to adjust from the population dose risk from internal events to the subject event dose risk. The same argument is also applied to the property damage risk from internal events at power and scaling property damage risk for internal fire events at power, internal flooding events at power, and shutdown events.

## 4.3 Estimate of Risk for Design

The SAMDA analysis uses two distinct analyses to form the basis for the baseline design risk. The first analysis is the Level 1 and 2 PRA of the US-APWR design, as summarized in Section 19.1 of the Design Control Document (DCD) (Reference 6). The second analysis is a Level 3 PRA analysis that integrates the Level 2 source terms from the first analysis to quantify the consequences based on a reference site. The SAMDA analysis is supported by the US-APWR Level 3 PRA report (Reference 7).

The Level 1 analyses resulted in a set of CDFs from events at power and also from LPSD, and are listed in Table 5. The CDF from at power internal events, fire and flood events is 4.4E-06 per year and from LPSD events is 2.0E-07 per year. The total CDF is therefore 4.6E-06 per year. Table 6 shows the major contributors to internal events at power, fire events at power, flooding events at power, and low-power and shutdown events. Additional details on the leading contributors to CDF are described in sections 19.1.4 through 19.1.6 of the DCD.

The Level 2 PRA accepts accident class input information from the Level 1 PRA and develops plant damage states (PDSs) using a containment system event tree logic model (Level 2 PRA CSET). The PDS information is used in a containment phenomenological event tree model (Level 2 PRA CPET) to calculate US-APWR release categories and their respective frequency. The MAAP code is used to develop the fission product source term corresponding to each release category (RC). Additional detail on the Level 2 PRA is found in section 19.1 of the US-APWR DCD and in the US-APWR PRA report (Reference 8). Table 7 shows the resulting release categories from internal at power events that are developed from the Level 2 PRA, a brief description of the release sequence and the release category frequency. The fission product group and inventory fraction released, timing and duration for each of the six release categories are shown in Table 8. The constituent elements for each fission product group are listed in Table 9.

The Level 3 PRA methodology is shown in the Appendix A, the MACCS2 code, Version 1.13.1 (Reference 9) is used to estimate the population dose for each release category source term. MACCS2 was developed by the NRC to model offsite consequences due to postulated atmospheric release of radionuclides. The MACCS2 code includes three modules: ATMOS, EARLY, and CHRONC. The ATMOS and the EARLY modules have been used for the offsite dose risk quantification during the emergency phase for the first 24 hours following the onset of core damage. The CHRONC module, has been used for offsite property damage risk evaluation in this PRA quantification.

In the offsite dose risk quantification, the meteorological data of the Surry site has been used as “typical”. The 50-mile population distribution data for the Surry site in the MACCS2 code sample input file has been adjusted to be in exceedance of about 80% of the U.S. nuclear plant sites, as described in NUREG/CR-2239, “Technical Guidance for Siting Criteria Development” (Reference 10).

The MACCS2 offsite dose and property damage risk quantification is executed for each release category source term. Consequences are calculated by MACCS2 for the first 24-hour period following onset of core damage. The code provides a complementary cumulative distribution function (CCDF) and the mean value results for each user-specified consequence. The CCDF presents the probability that a level of consequence is exceeded and can be used to evaluate the relative likelihood of a result compared to a mean, or average value. In the SAMDA analysis, the mean values are used for the baseline risk profile. The population dose risk is calculated by multiplying the release category frequency by the mean value of the consequence result. Therefore, the overall population dose risk is the sum of the six release category risks and is reported in terms of person-sievert/year (person-Sv/y). Similarly, the offsite property damage risk is calculated based on the sum of the six individual mean property damage risks and reported in dollars/per reactor-year, ($/reactor-year).

Table 10a shows the population dose, the population dose risk, and the percentage contribution for each release category from internal at power events. RC5 is not evaluated because there is no release within 24 hours after the onset of core damage. The total risk is 0.27 person-rem/y, and the largest contributor is from RC3 – Containment overpressure failure due to loss of heat removal (86%).

Table 10b shows the offsite property damage cost, the associated property damage risk, and the percentage contribution for each release category from internal at power events. RC5 is not evaluated because there is no release within 24 hours after the onset of core damage. The total risk in this category is $8.9/yr, with the largest contributors being: RC3 – Containment overpressure failure due to loss of heat removal (58%), RC4 – Early containment failure (20%), and RC1 – Containment Bypass (18%).

# 5. SUMMARY OF RISK SIGNIFICANT ENHANCEMENTS

This section summarizes the design enhancements already incorporated into the US-APWR plant due to probabilistic risk assessment insights and results. Major enhanced design features adopted in the US-APWR to reduce or eliminate weaknesses in current reactor design is summarized in Table 11.

# 6. REFERENCE SITE CHARACTERISTICS

In the offsite dose risk quantification, the meteorological data of the Surry site has been used as “typical”, which is accessible in the MACCS2 code sample input file attached to the MACCS2 code. Since the Surry site is not very near the ocean, the sea-breeze effects are not included (which is good) and the terrain is flat which is more typical than sites with complex river valley effects. Average wind speeds are also not excessively high (i.e.: they are not too high which would be non-conservative).

The 50-mile population data of the Surry site in the MACCS2 code sample input file has been adjusted to be representative of about 80% of the U.S. nuclear plant sites in NUREG/CR-2239, “Technical Guidance for Siting Criteria Development” (Reference 10).

# 7. VALUE OF ELIMINATING RISK

In the present analysis, the assumption is made that the population dose risk from internal events at power is applicable to internal fire events at power, internal flooding events at power, and shutdown events. A CDF scaling factor is applied to adjust from the population dose risk from internal events to the subject event dose risk. The same argument is also applied to the property damage risk from internal events at power and scaling property damage risk for internal fire events at power, internal flooding events at power, and shutdown events.

The total maximum averted cost benefit is the sum of the five component cost benefits for the four events discussed above and resulted to be \$289k. Cost results are summarized in Table 1

As an uncertainty analysis, Table 12 shows the outcome of each SAMDA benefit sensitivity analysis. Each SAMDA benefit is derived by multiplying each ratio of contribution to decrease CDF or large release frequency (LRF) and the maximum averted benefit is \$289k.together. The baseline benefit involves a real discount rate, r, of 7%/year (0.07/year), as recommended in NUREG/BR-0184, the sensitivity cases of 5% and 3% discount rate are specified in NEI 05-01 and NUREG/BR-0058 respectively. The last column shows the SAMDA benefit using a monetary equivalent of population dose of \$3,000 per person-rem (instead of the \$2,000 per person-rem value used in the baseline analysis).

# 8. EVALUATION OF POTENTIAL IMPROVEMENTS

This section discusses the cost impacts of candidate design improvements (Phase II evaluation of SAMDA candidate items), and consists of performing a cost estimate for the ten items not screened in the Phase I analysis and followed NEI 05-01 Section 7 methods. For those SAMDAs involving hardware modifications, the cost estimation process was to find "standard" costs from the following:

- NEI 05-01, Rev. A
- Severe accident mitigation alternative (SAMA) analyses for current U.S. power plants
- SAMDA analyses for other reactor designs.

Cost estimates that were derived independent of earlier precedents included procurement and installation, and where applicable, long-term maintenance, surveillance, calibration and training. These factors are allowable under NEI 05-01. A measure of conservatism was retained in the cost estimates to allow a reasonable examination of cost vs. benefit. As an example, older studies were used for cost examples of a SAMA or SAMDA candidate without attempting to adjust to present-day dollars. Additionally, in only one case (Containment Spray System, SAMDA #10) was the alternative cost scaled from a lower-power, current plant to the larger power (1610 Mwe) appropriate for US-APWR. The costs of other SAMDA candidates was maintained without adjustment by power scaling factor.

SAMDA cost evaluation results are described with their assumptions in Table 3. The lowest cost SAMDA is SAMDA #10, *Install a redundant containment spray system*, at $870k. The second lowest cost SAMDA is SAMDA #4, *Provide an additional high pressure injection pump with independent diesel*, at $1,000k. These potential improvements are substantially more costly than the calculated maximum averted cost of $289k.

# 9. RESULTS

There are no additional design alternatives that are shown to be cost-beneficial in severe accident mitigation design. As shown in Table 12, all the SAMDA costs are above the benefit, and therefore, are not cost-beneficial.

This SAMDA analysis inherently involves various uncertainties; the evaluation is therefore performed considering a variety of conservative assumptions. Wide range of sensitivity analyses are also performed, and consequently the result of this SAMDA analysis shows that it is far below cost-beneficial to consider applying any design alternatives. Therefore, the influence from the PRA update can be sufficiently covered in a safety margin and conservatisms.

# 10. REFERENCES

1 NUREG-1555, Revision 1, Environmental Standard Review Plan, Section 7.3, Severe Accident Mitigation Alternatives, U.S. Nuclear Regulatory Commission, Washington, DC, July 2007.

2 NUREG/BR-0184, Regulatory Analysis Technical Evaluation Handbook, U.S. Nuclear Regulatory Commission, Washington, DC, 1997.

3 NUREG/BR-0058, Revision 4, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, U.S. Nuclear Regulatory Commission, Washington, DC, August 2004.

4 NEI 05-01, Revision A, Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document, Nuclear Energy Institute, Washington, D.C., 2005.

5 Treatment of Averted Onsite Costs in Regulatory Analysis, SECY-99-169, U.S. Nuclear Regulatory Commission, Washington, DC, letter issued July 1, 1999.

6 Design Control Document for the US-APWR, Rev. 1, Mitsubishi Heavy Industries, August 2008.

7 Takechi, Y., et al., US-APWR Probabilistic Risk Assessment (Level 3) Rev.1, MUAP-08004, Mitsubishi Heavy Industries, September 2008.

8 Tanaka, F., et al., US-APWR Probabilistic Risk Assessment, MUAP-07030, Rev. 1, Mitsubishi Heavy Industries, September 2008.

9 NUREG/CR-6613, Vol. 1 (SAND97-0594), Code Manual for MACCS2 User's Guide, U.S. Nuclear Regulatory Commission, Washington, DC, and Sandia National Laboratories, May 1998.

10 NUREG/CR-2239, Technical Guidance for Siting Criteria Development, U.S. Nuclear Regulatory Commission, Washington, DC, and Sandia National Laboratories, December 1982.

11 Applicant's Environmental Report Operating License Renewal Stage for Vogtle Electric Generating Plant Units 1 and 2, Southern Nuclear Operating Company Docket No. 50-424, License No. NPF-068 and Docket No. 50-425, License No. NPF-081 June 2007 Attachment F.

# Tables

### Table 1 – US-APWR Value of Risk Avoided

| Cost Component | Internal Events | Internal Fire | Internal Flood | LPSD | Totals for All Events |
|---|---|---|---|---|---|
| Offsite Exposure Cost | \$7.6k | \$11.4k | \$8.9k | \$1.3k | \$29.1k |
| Offsite Property Damage Cost | \$0.1k | \$0.2k | \$0.1k | \$0.02k | \$0.5k |
| Onsite Exposure Cost | \$0.6k | \$0.9k | \$0.7k | \$0.1k | \$2.3k |
| Cleanup and Decontamination Cost | \$18.2k | \$27.3k | \$21.3k | \$3.0k | \$69.8k |
| Replacement Power Cost | \$48.9k | \$73.4k | \$57.1k | \$8.2k | \$187.6k |
| Total (Maximum Averted Cost Benefit) | \$75.5k | \$113.2k | \$88.1k | \$12.6k | \$289.3k |

## Table 2 – SAMDA Candidate List and Phase I Screening (sheet 1 of 9)

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| **dc and ac power** | | | |
| **1** | **001** | **Provide additional dc battery capacity (At least one train emergency dc power can be supplied more than 24 hours)** | **(Not screened)** |
| | 002 | Replace lead-acid batteries with fuel cells | Very low benefit |
| | 003 | Add additional battery charger or portable, diesel-driven battery charger to existing dc system | Very low benefit |
| | 004 | Improve dc bus Load shedding | Very low benefit |
| | 005 | Provide dc bus cross-ties | Very low benefit |
| | 006 | Provide additional dc power to the 120/240V vital ac system | Very low benefit |
| | 007 | Add an automatic feature to transfer the 120V vital ac bus from normal to standby power | Very low benefit |
| | 008 | Increase training on response to loss of two 120V ac buses which causes inadvertent actuation signals | Very low benefit |
| **2** | **009** | **Provide an additional (diesel) gas turbine generator (At least one train emergency ac power can be supplied more than 24 hours)** | **(Not screened)** |
| | 010 | Revise procedure to allow bypass of diesel generator trips. | Not a design alternative |
| | 011 | Improve 4.16-kV bus cross-tie ability | Very low benefit |
| | 012 | Create ac power cross-tie capability with other unit (multi-unit site) | Not applicable |
| **3** | **013** | **Install an additional, buried off-site power source** | **(Not screened)** |
| | 014 | Install a gas turbine generator. | Already implemented |
| | 015 | Install tornado protection on gas turbine generator | Very low benefit |
| | 016 | Improve uninterruptible power supplies | Not a design alternative |
| | 017 | Create a cross-tie for diesel fuel oil (multi-unit site) | Not applicable |
| | 018 | Develop procedures for replenishing diesel fuel oil | Not a design alternative |
| | 019 | Use fire water system as a backup source for diesel cooling | Not applicable |
| | 020 | Add a new backup source of diesel cooling | Not applicable |
| | 021 | Develop procedures to repair or replace failed 4 KV breakers | Not a design alternative |
| | 022 | In training, emphasize steps in recovery of off-site power after an station blackout (SBO) | Not a design alternative |

**Table 2 – SAMDA Candidate List and Phase I Screening (sheet 2 of 9)**

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| | 023 | Develop a severe weather conditions procedure | Not a design alternative |
| | 024 | Bury off-site power lines | Very low benefit |
| **Core Cooling** | | | |
| | 025 | Install an independent active or passive high pressure injection system | Combined with No. 026 |
| **4** | **026** | **Provide an additional high pressure injection pump with independent diesel (With dedicated pump cooling)** | **(Not screened)** |
| | 027 | Revise procedure to allow operators to inhibit automatic vessel depressurization in non-ATWS scenarios | Not a design alternative |
| | 028 | Add a diverse low pressure injection system. | Very low benefit |
| | 029 | Provide capability for alternate injection via diesel-driven fire pump | Not applicable |
| | 030 | Improve emergency core cooling system (ECCS) suction strainers | Already implemented |
| | 031 | Add the ability to manually align emergency core cooling system recirculation. | Not applicable |
| | 032 | Add the ability to automatically align emergency core cooling system to recirculation mode upon refueling water storage tank depletion | Not applicable |
| | 033 | Provide hardware and procedure to refill the reactor water storage tank once it reaches a specified low level | Not applicable |
| | 034 | Provide an in-containment reactor water storage tank. | Already implemented |
| | 035 | Throttle low pressure injection pumps earlier in medium or large-break loss-of-coolant accident (LOCA) to maintain reactor water storage tank inventory. | Not applicable |
| | 036 | Emphasize timely recirculation alignment in operator training. | Not a design alternative |
| | 037 | Upgrade the chemical and volume control system to mitigate small break loss of coolant accident (SBLOCA). | Very low benefit |
| | 038 | Change the in-containment reactor water storage tank suction from four check valves to two check and two air-operated valves. | Very low benefit |
| | 039 | Replace two of the four electric safety injection pumps with diesel-powered pumps | Combined with No. 026 |

## Table 2 – SAMDA Candidate List and Phase I Screening (sheet 3 of 9)

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| | 040 | Provide capability for remote, manual operation of secondary side pilot-operated relief valves in a station blackout. | Already implemented |
| | 041 | Create a reactor coolant depressurization system | Already implemented |
| | 042 | Make procedure changes for reactor coolant system depressurization | Not a design alternative |
| **Cooling water** | | | |
| | 043 | Add redundant dc control power for service water (SW) pump | Very low benefit |
| | 044 | Replace ECCS pump motors with air-cooled motors | Very low benefit |
| | 045 | Enhance procedural guidance for use of cross-tied component cooling or service water pumps | Not a design alternative |
| **5** | **046** | **Add a service water pump (Add independent train)** | **(Not screened)** |
| | 047 | Enhance the screen wash system | Very low benefit |
| | 048 | Cap downstream piping of normally closed component cooling water drain and vent valves | Very low benefit |
| | 049 | Enhance loss of component cooling water (or loss of service water) procedures to facilitate stopping the reactor coolant pumps | Not a design alternative |
| | 050 | Enhance loss of component cooling water procedure to underscore the desirability of cooling down the reactor coolant system prior to seal LOCA | Not a design alternative |
| | 051 | Additional training on loss of component cooling water | Not a design alternative |
| | 052 | Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals | Already implemented |
| | 053 | On loss of essential raw cooling water, proceduralize shedding component cooling water loads to extend the component cooling water heat-up time | Very low benefit |
| | 054 | Increase charging pump lube oil capacity | Very low benefit |
| **6** | **055** | **Install an independent reactor coolant pump seal injection system, with dedicated diesel (With dedicated pump cooling)** | **(Not screened)** |
| | 056 | Install an independent reactor coolant pump seal injection system, without dedicated diesel | Very low benefit |
| | 057 | Use existing hydro test pump for reactor coolant pump seal injection | Not applicable |

**Table 2 – SAMDA Candidate List and Phase I Screening (sheet 4 of 9)**

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| | 058 | Install improved reactor coolant pump seals | Very low benefit |
| **7** | **059** | **Install an additional component cooling water pump (Add independent train)** | **(Not screened)** |
| | 060 | Prevent makeup pump flow diversion through the relief valves | Very low benefit |
| | 061 | Change procedure to isolate reactor coolant pump seal return flow on loss of component cooling water, and provide (or enhance) guidance on loss of injection during seal LOCA | Not a design alternative |
| | 062 | Implement procedures to stagger high pressure safety injection pump use after a loss of service water | Not a design alternative |
| | 063 | Use fire prevention system pumps as a backup seal injection and high pressure makeup source | Not applicable |
| | 064 | Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie | Already implemented |
| **Feedwater and Condensate** | | | |
| | 065 | Install a digital feed water upgrade | Very low benefit |
| | 066 | Create ability for emergency connection of existing or new water sources to feedwater and condensate systems | Already implemented |
| | 067 | Install an independent diesel for the condensate storage tank makeup pump | Very low benefit |
| **8** | **068** | **Add a motor-driven feedwater pump (With independent room cooling)** | **(Not screened)** |
| | 069 | Install manual isolation valves around auxiliary feedwater turbine-driven steam admission valves | Very low benefit |
| | 070 | Install accumulators for turbine-driven auxiliary feedwater pump flow control valves | Not applicable |
| | 071 | Install a new condensate storage tank (auxiliary feedwater storage tank) | Very low benefit |
| | 072 | Modify the turbine-driven auxiliary feedwater pump to be self-cooled | Very low benefit |
| | 073 | Proceduralize local manual operation of auxiliary feedwater system when control power is lost | Not a design alternative |
| | 074 | Provide hookup for portable generators to power the turbine-driven auxiliary feedwater pump after station batteries are deleted | Very low benefit |

**Table 2 – SAMDA Candidate List and Phase I Screening (sheet 5 of 9)**

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| | 075 | Use fire water system as a backup for steam generator inventory | Very low benefit |
| | 076 | Change failure position of condenser makeup valve fails open on loss of air or power | Very low benefit |
| | 077 | Provide a passive, secondary-side heat-rejection loop consisting of a condenser and heat sink | Combined with No. 068 |
| | 078 | Modify the startup feedwater pump so that it can be used as a backup to the emergency feedwater system, including during a station blackout scenario | Very low benefit |
| | 079 | Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed | Already implemented |
| **Heating, Ventilation, and Air Conditioning** | | | |
| | 080 | Provide a redundant train or means of ventilation | Very low benefit |
| | 081 | Add a diesel building high temperature alarm or redundant louver and thermostat | Not applicable |
| | 082 | Stage backup fans in switchgear rooms | Very low benefit |
| | 083 | Add a switchgear room high temperature alarm | Very low benefit |
| | 084 | Create ability to switch emergency feedwater room fan power supply to station batteries in a station blackout | Very low benefit |
| **Instrument Air and Nitrogen Supply** | | | |
| | 085 | Provide cross-unit connection of uninterruptible compressed air supply | Very low benefit |
| | 086 | Modify procedure to provide ability to align diesel power to more air compressors | Not a design alternative |
| | 087 | Replace service and instrumental air compressors with more reliable compressors witch have self-contained air cooling by shaft driven fans | Very low benefit |
| | 088 | Install nitrogen bottles as backup gas supply for safety relief valves | Not applicable |
| | 089 | Improve SRV and MSIV pneumatic components | Very low benefit |
| **Containment** | | | |
| | 090 | Create a reactor cavity flooding system | Already implemented |
| | 091 | Install a passive containment spray system | Not applicable |
| | 092 | Use the fire water system as a backup source for the containment spray system | Already implemented |
| | 093 | Install an unfiltered, hardened containment vent | Not applicable |

## Table 2 – SAMDA Candidate List and Phase I Screening (sheet 6 of 9)

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| **9** | **094** | **Install a filtered containment vent to remove decay heat<br>Option 1: Gravel bed Filter<br>Option 2: Multiple Venturi Scrubber** | **(Not screened)** |
| | 095 | Enhance fire protection system and standby gas treatment system hardware and procedures | Very low benefit |
| | 096 | Provide post-accident containment inerting capability | Very low benefit |
| | 097 | Create a large concrete crucible with heat removal potential to contain molten core debris | Already implemented |
| | 098 | Create a core melt source reduction system | Very low benefit |
| | 099 | Strengthen primary/secondary containment (e.g., add ribbing to containment shell) | Very low benefit |
| | 100 | Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur | Very low benefit |
| | 101 | Provide a reactor vessel exterior cooling system | Very low benefit |
| | 102 | Construct a building to be connected to primary/secondary containment and maintained at a vacuum | Very low benefit |
| | 103 | Institute simulator training for severe accident scenarios | Not a design alternative |
| | 104 | Improve leak detection procedure | Very low benefit |
| | 105 | Delay containment spray actuation after a large LOCA | Not applicable |
| | 106 | Install a automatic containment spray pump header throttle valves | Not applicable |
| **10** | **107** | **Install a redundant containment spray system<br>(Add independent train)** | **(Not screened)** |
| | 108 | Install an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing ac/dc independent power supplies, such as the security system diesel | Already implemented |
| | 109 | Install a passive hydrogen control system | Very low benefit |
| | 110 | Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure | Already implemented |

## Table 2 – SAMDA Candidate List and Phase I Screening (sheet 7 of 9)

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| **Containment Bypass** | | | |
| | 111 | Install additional pressure or leak monitoring instruments for detection of interfacing system loss-of-coolant accident (ISLOCA) | Very low benefit |
| | 112 | Add redundant and diverse limit switches to each containment isolation valve | Very low benefit |
| | 113 | Increase leak testing of valves in ISLOCA paths | Very low benefit |
| | 114 | Install self-actuating containment isolation valves | Very low benefit |
| | 115 | Locate residual heat removal (RHR) inside containment | Very low benefit |
| | 116 | Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water | Very low benefit |
| | 117 | Revise EOPs to improve ISLOCA identification | Not a design alternative |
| | 118 | Improve operator training on ISLOCA identification | Not a design alternative |
| | 119 | Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage | Not a design alternative |
| | 120 | Replace steam generator with a new design | Not applicable |
| | 121 | Increase the pressure capacity of the secondary side so that a steam generator tube rupture would not cause the relief valves to lift | Very low benefit |
| | 122 | Install a redundant spray system to depressurize the primary system during a steam generator tube rupture | Very low benefit |
| | 123 | Proceduralize use of pressurizer vent valves during steam generator tube rupture sequences | Not a design alternative |
| | 124 | Provide improved instrumentation to detect steam generator tube rupture, such as Nitrogen-16 monitors | Already implemented |
| | 125 | Route the discharge from the main steam safety valves through a structure where a water spray would condense the steam and remove most of the fission products | Excessive Implementation cost |
| | 126 | Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources | Very low benefit |
| | 127 | Revise emergency operating procedures to direct isolation of a failure steam generator | Not a design alternative |

**Table 2 – SAMDA Candidate List and Phase I Screening (sheet 8 of 9)**

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| | 128 | Direct steam generator flooding after a steam generator tube rupture, prior to core damage | Not a design alternative |
| | 129 | Vent main steam safety valves in containment | Excessive Implementation cost |
| **ATWS** | | | |
| | 130 | Add an independent boron injection system | Very low benefit |
| | 131 | Add a system of relief valves to prevent equipment damage from pressure spikes during ATWS | Very low benefit |
| | 132 | Provide an additional control system for rod insertion (e.g., AMSAC) | Already implemented |
| | 133 | Install an ATWS sized filtered containment vent to remove decay heat | Very low benefit |
| | 134 | Revise procedure to bypass MSIV isolation in turbine trip ATWS scenarios | Very low benefit |
| | 135 | Revise procedure to allow override of low pressure core injection during an ATWS event | Not applicable |
| | 136 | Install motor generator set trip breakers in control room | Very low benefit |
| | 137 | Provide capability to remove power from the bus powering the control rod | Already implemented |
| **Internal Flooding** | | | |
| | 138 | Improve inspection of rubber expansion joints on main condenser | Very low benefit |
| | 139 | Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment | Already implemented |
| **Seismic** | | | |
| | 140 | Increase seismic ruggedness of plant components | Very low benefit |
| | 141 | Provide additional restraints for $CO_2$ tanks | Very low benefit |
| **Fire** | | | |
| | 142 | Replace mercury switches in fire protection system | Very low benefit |
| | 143 | Upgrade fire compartment barriers | Already implemented |
| | 144 | Install additional transfer and isolation switches | Very low benefit |
| | 145 | Enhance fire brigade awareness | Not a design alternative |

## Table 2 – SAMDA Candidate List and Phase I Screening (sheet 9 of 9)

| Phase II ID | Phase I SAMDA ID | Potential Enhancement (NEI-05-01) | Screening |
|---|---|---|---|
| | 146 | Enhance control of combustibles and ignition sources | Not a design alternative |
| **Others** | | | |
| | 147 | Install digital large break LOCA protection system | Very low benefit |
| | 148 | Enhance procedures to mitigate large break LOCA | Very low benefit |
| | 149 | Install computer aided instrumentation system to assist the operator in assessing post-accident plant status | Already implemented |
| | 150 | Improve maintenance procedures | Not a design alternative |
| | 151 | Increase training and operating experience feedback to improve operator response | Not a design alternative |
| | 152 | Develop procedures for transportation and nearby facility accidents | Not a design alternative |
| | 153 | Install secondary side guard pipes up to the main steam isolation valves | Very low benefit |
| **US-APWR Specific: Internal Flooding** | | | |
| | AP001 | Enclose electrical equipment room of turbine building (T/B) with waterproof walls (Rating fire barrier with water tight) | Already implemented |
| | AP002 | Provide enough surface area at reactor building (R/B) B1F | Excessive Implementation cost |
| **US-APWR Specific: Fire** | | | |
| | AP003 | Divide the electrical room of T/B into two fire compartments | Already implemented |

**Table 3 – US-APWR SAMDA Candidates (sheet 1 of 3)**

| SAMDA # | Design Alternative | Description | Estimated Cost ($) | Basis for Cost Estimate |
|---|---|---|---|---|
| 1. | Provide additional dc battery capacity | At least one train emergency dc power can be supplied more than 24 hours. Specifically, battery capacity corresponding to 11 of present emergency dc batteries is added. | 2.00E+06 | Peach Bottom License Renewal, ER, Appendix F, (Table G.4-2, Phase II SAMA) Phase 2 SAMA ID# 11, (Phase 1 SAMA ID# 57)page E.G-91, $200k per battery. Assume $200k x 11 = $2.2M. |
| 2. | Provide an additional gas turbine generator | At least one train emergency ac power can be supplied more than 24 hours. Specifically another gas turbine generator same as the emergency gas turbine generator (GTG) is added. | 1.00E+07 | Farley nuclear power plant (NPP) License Renewal, ER, Appendix F, Phase 1 SAMA ID# 70, page F-40 estimated $16,100,000. |
| 3. | Install an additional, buried off-site power source | This candidate comes from the NEI 05-01 and generic information is adopted. | 1.00E+07 | NEI 05-01, Rev. 1: Table 14 (STANDARD List of PWR SAMA Candidates),SAMA ID 013,(Note 1). |
| 4. | Provide an additional high pressure injection pump with independent diesel | In addition to the 4 train high pressure injection (HPI) system, $5^{th}$ HPI train is added. The $5^{th}$ train has the same capacity with others and independent pump cooling system. | 1.00E+06 | Palisades License Renewal, Table G-4 (SAMA Cost Benefit Screening Analysis for Palisades) NUREG 1437 Supplement 27, page G-20, estimated cost of $1,620,000 |

**Table 3 – US-APWR SAMDA Candidates (sheet 2 of 3)**

| SAMDA # | Design Alternative | Description | Estimated Cost ($) | Basis for Cost Estimate |
|---|---|---|---|---|
| 5. | Add a service water pump | In addition to the 4-train essential service water system, 5th essential service water system (ESWS) train is added. The 5th train has the same capacity with others. | 5.90E+06 | NEI 05-01, Rev. 1: Table 12 (SAMPLE Sensitivity Analysis Results), SAMA ID#1 estimated cost |
| 6. | Install an independent reactor coolant pump seal injection system, with dedicated diesel | 1 set seal injection pump and injection line with dedicated or independent pump cooling system. | 3.80E+06 | Farley NPP License Renewal, ER, Appendix F, (Table F-10, Disposition of initial SAMAs Investigated) Phase 1 SAMA ID# 10, page F-33 estimated $3,800,000. |
| 7. | Install an additional component cooling water pump | In addition to the 4 train component cooling water system, 5th component cooling water system (CCWS) train is added. The 5th train has the same capacity with others. | 1.50E+06 | Farley NPP License Renewal, ER, Appendix F, (Table F-10, Disposition of initial SAMAs Investigated) Phase 1 SAMA ID# 14, page F-33 estimated $1,500,000. |
| 8. | Add a motor-driven feed-water pump | In addition to the 4 train emergency feed water system, 5th emergency feed water system (EFWS) train is added. The 5th train has the same capacity with others and independent system of air conditioning for pump room. | 2.00E+06 | Farley NPP License Renewal, ER, Appendix F, (Table F-10, Disposition of initial SAMAs Investigated) Phase 1 SAMA ID# 107, page F-44 estimated $2,200,000. |

**Table 3 – US-APWR SAMDA Candidates (sheet 3 of 3)**

| SAMDA # | Design Alternative | Description | Estimated Cost ($) | Basis for Cost Estimate |
|---|---|---|---|---|
| 9. | Install a filtered containment vent to remove decay heat | This candidate comes from the NEI 05-01 and generic information is adopted. | 3.00E+06 | NEI 05-01, Rev. 1: Table 12 (SAMPLE Sensitivity Analysis Results), SAMA ID#5 estimated cost |
| 10. | Install a redundant containment spray system | In addition to the 4 train containment spray system, 5th containment spray system (CSS) train is added. The 5th train has the same capacity with others. | 8.70E+05 | Farley NPP License Renewal, ER, Appendix F, (Table F-10, Disposition of initial SAMAs Investigated) Phase 1 SAMA ID# 46, page F-37 estimated $450,000. Based on electric power ratio =1610/830=1.9. Similar scaling would expected with thermal power ratio.<br>Estimated cost = 1.9 x $450k = $8.7E+05 |

**Note 1:** The implementation of an alternate alternating current (AAC) power source would most likely take the form of an additional emergency diesel generator (EDG). This SAMA would help mitigate loss of offsite power (LOOP) events and would reduce the risk during time frames of on-line EDG maintenance. The cost of installing an additional EDG has been estimated to be greater than $20 million in the Calvert Cliffs Application for License Renewal [BGE 1998]. It was similarly estimated to be about $26 M for both units at VEGP [Reference 11]. A per unit cost of ~$10M is therefore estimated for US-APWR.

**Table 4 – US-APWR Value of Risk Avoided Input Data**

| Parameter | Explanation | Baseline |
|---|---|---|
| r | real discount rate [/year] (based on NUREG/BR-0184) | 0.07 |
| $t_f$ | years remaining until end of facility life [year] | 60 |
| R | monetary equivalent of unit dose [$/person-rem] (based on NUREG/BR-0184) | 2,000 |
| $D_{IO}$ | immediate on-site occupational dose [person-rem/event] (based on NUREG/BR-0184) | 3,300 |
| $D_{LTO}$ | long-term on-site occupational dose [person-rem/event] (based on NUREG/BR-0184) | 20,000 |
| $m_{LT}$ | years over which long-term dose accrue [years] (based on NUREG/BR-0184) | 10 |
| $C_{CD}$ | The total cost of cleanup and decontamination of a power reactor facility subsequent to a severe accident is estimated in NUREG/BR-0184 to be $1.5E+9. [$] | 1.50E+09 |
| $m_{CD}$ | cleanup period [years] (based on NUREG/BR-0184) | 10 |
| P | rated electrical power output [MWe] | 1610 |
| B | A constant representing a string of replacement power costs that occur over the lifetime of a reactor after an event (dollars). For a 910 MWe “generic” reactor NUREG/BR-0184 uses a value of 1.2E+08. (based on NUREG/BR-0184)<br>B = 1.2E+08 x (1610/910) | 2.1E+08 |
| CDF | core damage frequency from internal at power events[/reactor-year] (See Table 5.) | 1.2E-06 |
| $D_{pa}$ | offsite exposure risk from internal at power events [person-rem/year] (See Table 10a.) | 0.27 |
| E | offsite property damage risk from internal at power events [dollars/year] (See Table 10b.) | 8.9 |

**Table 5 – CDF and LRF from PRA Analysis for US-APWR**

| Event | CDF (1/ry) | LRF (1/ry) |
|---|---|---|
| At-power Internal Events | 1.2E-06 | 1.0E-07 |
| At-power fire | 1.8E-06 | 2.3E-07 |
| At-power flood | 1.4E-06 | 2.8E-07 |
| LPSD | 2.0E-07 | 2.0E-07 |
| Total | 4.6E-06 | 8.1E-07 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 1 of 9)**

| No. | Description | | |
|---|---|---|---|
| | Core Damage Frequency – Internal Events at Power | CDF (1/ry) | Percentage (%) |
| 1. | LOOP with reactor trip. The emergency power supply system (emergency power generator) and alternative ac power source fail to operate and loss of total ac power occurs. EFWS (turbine-driven pumps) succeeds. Offsite power does not recover within 1 hour, and RCP seal LOCA occurs since RCP seal cooling and RCP seal injection is lost during loss of total ac power. In addition, functions to mitigate RCP seal LOCA are also unavailable due to loss of power. Liquid level in the RCS decreases, and 2 hours after initiation of RCP seal LOCA (3 hours after LOOP), core is uncovered. | 5.0E-07 | 42.1 |
| 2. | LOCCW with reactor trip: EFWS successfully functions, but RCP seal LOCA occurs due to failure of the alternate component cooling of the charging pump utilizing FSS or non-essential chilled water system. In addition, functions to mitigate RCP seal LOCA are also unavailable due to loss of CCW. RCS inventory gradually decreases, and finally the core is damaged. | 2.6E-07 | 21.6 |
| 3. | Reactor vessel rupture occurs. This event directly leads to core damage since the reactor vessel can no longer maintain RCS coolant inside. | 1.0E-07 | 8.5 |
| 4. | LOOP with reactor trip: Emergency power supply and EFWS successfully function, but CCWS pumps fail to restart and loss of CCW flow occurs. Alternate component cooling of charging pump utilizing FSS or non-essential chilled water system fails and eventually RCP seal LOCA occurs. In addition, functions to mitigate RCP seal LOCA are unavailable due to loss of CCW. RCS inventory gradually decreases, and finally the core is damaged. | 6.2E-08 | 5.3 |
| | Large Release Frequency – Internal Events at Power | LRF (1/ry) | Percentage (%) |
| 1. | LOOP with reactor trip. Emergency ac power supply system and AAC power source fail and lead to SBO. EFWS (turbine-driven pumps) succeeds. However, RCP seal LOCA occurs due to RCP seal cooling failure. Also, recovery of power systems within 3 hours fails and results in core damage. The containment isolation before core damage succeeds. However, RCS depressurization fails due to loss of emergency power supply. Also reactor cavity flooding fails due to loss of electrical power after core damage. The recovery of power system by the commencement of MCCI fails and results in containment failure. | 1.4E-08 | 13.1 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 2 of 9)**

| No. | Description | | |
|---|---|---|---|
| 2. | LOCCW with reactor trip. EFWS succeeds. Both alternate CCW supply by the non-essential chilled water and by the FSS fail to operate and result in RCP seal LOCA due to RCP cooling failure. Consequently, it results in core damage. The containment isolation, RCS depressurization, and reactor cavity flooding succeeds. However firewater injection to the spray header fails to operate due to human error. Recovery of CCWS fails and therefore results in containment failure. | 9.8E-09 | 8.9 |
| 3. | SLOCA with reactor trip. EFWS, SIS, and CSS succeed. Therefore, core cooling succeeds. However, containment heat removal by the CS/RHR HX fails. Also, the alternative containment cooling by containment fan cooler system fails to operate and results in containment failure before core damage. | 7.7E-09 | 6.9 |
| 4. | LOCCW with reactor trip. This is the same as (2) until core damage. The containment isolation, RCS depressurization, and reactor cavity flooding succeeds. However, firewater injection to the spray header fails. Recovery of CCWS fails and therefore results in containment failure. | 6.0E-09 | 5.5 |
| 5. | LOCCW with reactor trip. This is the same as (2) until core damage. The containment isolation and RCS depressurization succeeds. However both firewater injections to the reactor cavity and to the spray header fail. Recovery of CCWS fails and therefore results in containment failure. | 6.0E-09 | 5.5 |
| 6. | LOCCW with reactor trip. This is the same as (2) until core damage. The containment isolation, RCS depressurization, and reactor cavity flooding succeeds. Also firewater injection to the spray header succeeds. However, recovery of CCWS fails and results in containment failure. | 3.6E 09 | 3.3 |
| 7. | LOCCW with reactor trip. This is the same as (2) until core damage. The containment isolation, RCS depressurization, and reactor cavity flooding succeeds. Also firewater injection to the spray header succeeds, and recovery of CCWS succeeds. However, the containment fails due to some severe accident phenomenon. | 2.9E 09 | 2.6 |
| 8. | SLOCA with reactor trip. EFWS and SIS succeed. Therefore, core cooling succeeds. However, CSS fails. Also, the alternative containment cooling by containment fan cooler system fails to operate and results in containment failure before core damage. | 1.7E-09 | 1.6 |
| 9. | RV rupture. This initiating event is assumed to directly result in core damage. All systems are functional. The containment isolation and reactor cavity flooding succeeds. Also the containment heat removal succeeds. However, the containment fails due to severe accident phenomena such as steam explosion and hydrogen burning. | 1.2E 09 | 1.1 |
| 10. | MLOCA with reactor trip. This is the same as (3) except for initiating events. | 1.1E-09 | 1.0 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 3 of 9)**

| No. | Description | | |
|---|---|---|---|
| | Internal Fire Events at Power | CDF (1/ry) | Percentage (%) |
| 1. | Yard Fire (Switchyard)<br>This area contains main transformer and reserve auxiliary transformer. Fire ignition source postulated in Switchyard are catastrophic fire, non-catastrophic fire and other fires of transformer (it has been referred to NUREG/CR-6850, attachment C, table 6-1, item 27, 28 and 29), whose fire ignition frequency is 2.0E-02/year.<br>The fire in this switchyard may cause LOOP (loss of offsite power), and it also may make the recovery of all power sources. CCDP of this fire scenario has been estimated to 6.0E-05/RY.<br>Fire scenario postulated is as follows:<br>- Fire may cause LOOP because main transformer and reserve auxiliary transformer located in switchyard may be damaged by the fire.<br>- Offsite power cannot be recovered because the fire may damage both of main transformer and reserve auxiliary transformer.<br>- All four class 1E gas turbine generators could not be operated by the random failure.<br>- Operator may fail to connect the emergency power bus to auxiliary alternative current by the human error.<br>- Reactor has the potential to cause the loss of all power supplies of safety systems | 1.2E-06 | 67.0 |
| 2. | FA6-101-01 (T/B other floor) fire<br>FA6-101-01 consists of many compartments in T/B and occupies large floor area, and many fire ignition sources are contained in this fire compartment. Fire ignition frequency of this fire compartment is 5.6E-02/year.<br>This fire compartment contains turbine bypass valves whose spurious operation due to fire leads to reactor transient, but any mitigation system such as EFW and ECCS are not damaged by this fire. Therefore, CCDP of this fire scenario is low, and has been estimated to 1.9E-06. | 1.0E-07 | 5.6 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 4 of 9)**

| No. | Description | | |
|---|---|---|---|
| 3. | FA6-101-04 (FA6-101-04 zone) fire<br>FA6-101-04 has the potential of transient combustibles fire and cable fire caused by welding or cutting and so forth, whose fire ignition frequency is 1.4E-03/year.<br>This area also contains all four-train cables to safety bus ducts from offsite power sources. Therefore, the fire in this area may cause LOOP, and it may make the recovery of every power sources impossible. And, CCDP of this fire scenario has been estimated to 6.0E-05/RY.<br>Fire scenario is as follows:<br>- Fire may cause LOOP because it may damage all four train cables to safety bus ducts from offsite power located in FA6-101-04.<br>- Offsite power cannot be recovered because fire may damage all four train of safety bus duct cable from offsite power sources.<br>- All four class 1E gas turbine generators could not be operated by the random failure.<br>- Operator may fail to connect the emergency power bus to auxiliary alternative current by the human error.<br>- Reactor has the potential to cause the core damage by causing the loss of all power supplies of safety systems. | 8.4E-08 | 4.7 |
| 4. | FA4-101(Auxiliary building) fire<br>FA4-101 consists of all compartments in A/B, and many fire ignition sources are contained in this area. Fire ignition frequency of this area is 2.5E-02/year.<br>FA4-101 contains turbine bypass valves whose spurious operation due to fire leads to SLBO (Steam Line Break), but does not contain mitigation systems (and their associated cable) like EFWS and ECCS. Therefore, CCDP of this fire scenario is low; and has been estimated to 1.9E-06.<br>In this fire scenario, human error of following operator actions has been postulated.<br>- Isolation of safety injection system by containment isolation valve (MOV-001A (B, C, D))<br>- Isolation of RWSP discharge line of CS/RHR by Isolation valve(MOV-001A (B, C, D))<br>- Isolation of CCW tie-line by manual valve | 5.1E-08 | 2.6 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 5 of 9)**

| No. | Description | | |
|---|---|---|---|
| 5. | FA2-205 (D class 1E electrical room) fire<br>FA2-205 contains D-train class 1E electrical cabinets of mitigation system and their cables, and those have the potential of fire ignition sources in this fire area. Fire ignition frequency of FA2-205 is 2.3E-03/year.<br>A fire in FA2-205 has the potential to cause the spurious operation of turbine bypass valve due to the control cable damage, and it may result in SLBO. Fire also have the potential to damage D-train mitigation system function of metal clad switch gear and control center. In addition, feedwater isolation valves to steam generator-C and D have the potential of spurious closure due to their control cables damaged and it results in loss of emergency feed water supply to 2 steam generators. CCDP of this fire scenario is low; and has been estimated to 2.0E-05.<br>Postulated fire scenario is as follows.<br>- Spurious opening of turbine bypass valves results in SLBO..<br>- Closing of main steam line isolation valve may fail by the random failure, and it may result in loss of Secondary system cooling.<br>- Moreover, if feed and bleed becomes unavailable by the operator error or the failure of safety depressurization valve, reactor has the potential to cause core damage and large release. | 4.6E-08 | 2.6 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 6 of 9)**

| No. | Description | | |
|---|---|---|---|
| 6. | FA2-202 (A class 1E electrical room) fire<br>FA2-202 contains A-train class 1E electrical cabinets of mitigation system and their cables, and those have the potential of fire ignition sources in this fire area. Fire ignition frequency of FA2-202 is 2.3E-03/year.<br>A fire in FA2-202 has the potential to cause the spurious operation of turbine bypass valve due to the control cable damage, and it may result in SLBO. Fire also have the potential to damage A-train mitigation system function of metal clad switch gear and control center. In addition, feedwater isolation valves to steam generator-A and B have the potential of spurious closure due to their control cables damaged and it results in loss of emergency feed water supply to 2 steam generators. CCDP of this fire scenario is low; and has been estimated to 1.9E-05.<br>Postulated fire scenario is as follows.<br>- Spurious opening of turbine bypass valve results in SLBO.<br>- Closing of main steam line isolation valve may fail by the random failure, and it may result in loss of secondary system cooling.<br>- Moreover, if feed and bleed becomes unavailable by the operator error or the failure of safety depressurization valve, reactor has the potential to cause core damage and large release. | 4.4E-08 | 2.5 |
| 7. | FA3-104 (A class 1E gas turbine room) fire<br>FA3-104 contains A-train gas turbine generator, emergency generator control board and fuel oil drain tank, and those have the potential of fire ignition sources in this fire area. Fire ignition frequency of FA3-104 is 5.4E-03/year.<br>It has been postulated that a fire in FA3-104 has the potential to cause the reactor transient. Fire has the potential to damage mitigation system function of A-train gas turbine generator, dc control center and their cables. CCDP of this fire scenario is low; and has been estimated to 6.9E-06.<br>Postulated fire scenario is as follows.<br>- It is assumed that the fire may cause reactor transient.<br>- Emergency feedwater line-B may fail by the random failure or the failure of support system like ESWS.<br>- Operator may fail to connect emergency feedwater system to EFW pit, and it may result in the loss of secondary system cooling.<br>- Moreover, if feed and bleed becomes unavailable by the operator error or the failure of safety depressurization valve, reactor has the potential to cause core damage and large release. | 3.7E-08 | 2.1 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 7 of 9)**

| No. | Description | | |
|---|---|---|---|
| 8. | FA2-205-M-05 (Propagation from FA2-205 to FA2-206) fire<br>This is the fire scenario which the fire in FA2-205 propagates to FA2-206. FA2-205 contains D-train class 1E electrical cabinets of mitigation system and their cables, and those have the potential of fire ignition sources in this fire area. Fire ignition frequency of FA2-205 is 2.3E-03/year.<br>FA2-205 contains the cables of safety depressurization valve, and FA2-206 contains safety depressurization valve isolation valve. This fire has the potential to cause the spurious opening of both valves due to the control cable damage, and it may result in SLOCA. Fire also has the potential to damage D-train mitigation system function of metal clad switch gear, control center and dc control center, and it results in C and D-trains mitigation functions. The fire also damages the control cables of accumulator outlet valves or nitrogen line isolation valves of every accumulator, and it may result in loss of accumulator function. CCDP of this fire scenario is low; and has been estimated to 2.2E-05.<br>Postulated fire scenario is as follows.<br>- Spurious opening of safety depressurization valve and safety depressurization valve isolation valve, and it may result in SLOCA.<br>- Moreover, if safety Injection system becomes unavailable by the random failure or the failure of support system like ESW and so forth, reactor has the potential to cause core damage and large release. | 3.7E-08 | 2.1 |
| | Internal Flooding at Power | CDF (1/ry) | Percentage (%) |
| 1. | Major flood due to the rupture of piping in the A-EFW pump (T/D) room on the B1F of R/B causes loss of function of both A and B trains of component cooling water pumps, essential chiller pumps, and batteries by the effect of flooding propagation. Also A and B EFW pumps lose the function. This scenario causes partial loss of component cooling water systems. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line and operator failure for feed and bleed operation result in core damage. | 1.7E-07 | 12.7 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 8 of 9)**

| No. | Description | | |
|---|---|---|---|
| 2. | Flood due to the rupture of piping in the D-EFW Pump (T/D) room on the B1F of R/B causes loss of function of both C and D trains of component cooling water pumps, essential chiller pumps, and batteries by the effect of flooding propagation. Also C and D EFW Pumps lose the function. This scenario causes partial loss of component cooling water systems. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line and operator failure for feed and bleed operation result in core damage. | 1.7E-07 | 12.2 |
| 3. | Flood due to the rupture of piping in the A-EFW pump (T/D) room on the B1F of R/B causes loss of function of both A and B trains of component cooling water pumps, essential chiller pumps, and batteries by the effect of flooding propagation. Also A and B EFW pumps lose the function. This scenario causes partial loss of component cooling water systems. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line and operator failure for feed and bleed operation result in core damage. | 1.5E-07 | 11.2 |
| 4. | Major flood due to the rupture of piping in the D-EFW pump (T/D) room on the B1F of R/B causes loss of function of both C and D trains of component cooling water pumps, essential chiller pumps, and batteries by the effect of flooding propagation. Also C and D EFW pumps lose the function. This scenario causes partial loss of component cooling water systems. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line and operator failure for feed and bleed operation result in core damage. | 1.5E-07 | 11.1 |
| 5. | Major flood due to the rupture of piping in the east side main steam line piping room on the 3F of R/B causes secondary line break. Secondary cooling by A and B steam generators are also not available. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line and operator failure for feed and bleed operation result in core damage. | 1.4E-07 | 10.5 |
| 6. | Major flood due to the rupture of piping in the west side main steam line piping room on the 3F of R/B causes secondary line break. Secondary cooling by C and D steam generators are also not available. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line and operator failure for feed and bleed operation result in core damage. | 1.3E-07 | 9.8 |
| 7. | Spray due to the leak from piping in the east side main steam line piping room on the 3F of R/B causes secondary line break. This scenario assumed plant shutdown by operators. Simultaneously operators fail to open the valve of EFW pit discharge cross tie line. Also operators fail to feed and bleed operation. | 7.3E-08 | 5.4 |

**Table 6 – Risk Profile from the Level 1 and Level 2 PRA Analysis for US-APWR (sheet 9 of 9)**

| No. | Description | | |
|---|---|---|---|
| 8. | Flood due to the rupture of piping on the 4F of R/B east side steam generator blowdown water radiation monitor room causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes partial loss of component cooling water systems. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line, random failure of one EFW pump and operator failure for feed and bleed operation result in core damage. | 3.7E-08 | 2.7 |
| 9. | Flood due to the rupture of piping on the 4F of R/B east side corridor causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes partial loss of component cooling water systems. Simultaneous operator failure to open the valve of EFW pit discharge cross tie line, random failure of one EFW pump and operator failure for feed and bleed operation result in core damage. | 3.7E-08 | 2.7 |
| 10. | Spray due to the leak from piping in the west side main steam line piping room on the 3F of R/B causes secondary line break. This scenario assumed plant shutdown by operators. Simultaneously operators fail to open the valve of EFW pit discharge cross tie line. Also operators fail to feed and bleed operation. | 3.3E-08 | 2.4 |
| | LPSD | CDF (1/ry) | Percentage (%) |
| 1. | Loss-of-coolant accident (LOCA) initiating event | 1.3E-7 | 65.0 |
| 2. | Loss of offsite power (LOOP) initiating event | 3.8E-8 | 19.0 |
| 3. | Loss of CCWS or ESWS initiating event | 2.3E-8 | 11.5 |

**Table 7 – Release Category Description for US-APWR**

| Designator | Description | Release Frequency (per reactor-year) |
|---|---|---|
| RC1 | Containment Bypass – Containment bypass which includes both core damage after SGTR and thermal induced SGTR after core damage | 7.5E-09 |
| RC2 | Containment Isolation Failure | 2.1E-09 |
| RC3 | Containment Failure before Core Damage – This category is for an overpressure failure before core damage due to loss of heat removal | 2.0E-08 |
| RC4 | Early containment failure – This is containment failure condition due to dynamic loads which includes hydrogen combustion before or just after reactor vessel failure, in-vessel or ex-vessel steam explosion, rocket-mode RV failure and containment direct heating. | 1.1E-08 |
| RC5 | Late Containment Failure – This failure of the containment includes overpressure failure after core damage, hydrogen combustion failure after core damage, hydrogen combustion long after reactor vessel failure and basemat melt-through | 6.5E-08 |
| RC6 | Intact Containment – This condition assumes an intact containment throughout the sequence and fission products are released at the design leak rate. | 1.1E-06 |
| Total | | 1.2E-06 |

## Table 8 – Source Term Input Data for US-APWR Release Categories

| Release category | Plume No. | START TIME (seconds) | END TIME (seconds) | Duration (seconds) | Release Fraction (MACCS2 Group) | | | | | | | | | Plume position |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | 1 Kr/Xe | 2 I | 3 Cs | 4 Te/Sb | 5 Sr | 6 Ru | 7 La | 8 Ce | 9 Ba | |
| RC1 | 1 | 1.0E+05 | 1.2E+05 | 1.5E+04 | 6.9E-01 | 2.0E-01 | 1.6E-01 | 8.6E-02 | 3.5E-04 | 1.5E-02 | 1.5E-05 | 4.3E-05 | 2.9E-03 | Leading |
| | 2 | 1.2E+05 | 1.5E+05 | 3.6E+04 | 2.5E-01 | 8.7E-02 | 3.9E-02 | 3.9E-02 | 4.5E-03 | 3.9E-03 | 2.2E-04 | 2.4E-04 | 8.8E-03 | Midpoint |
| | 3 | 1.5E+05 | 2.4E+05 | 8.6E+04 | 2.7E-03 | 4.0E-03 | 8.5E-03 | 7.9E-03 | 3.7E-03 | 4.2E-03 | 2.1E-03 | 1.4E-03 | 3.5E-03 | Midpoint |
| | 4 | 2.4E+05 | 3.3E+05 | 8.6E+04 | 4.9E-03 | 2.3E-03 | 2.7E-03 | 6.1E-04 | 1.9E-04 | 7.6E-05 | 6.2E-04 | 5.3E-04 | 9.7E-05 | Midpoint |
| RC2 | 1 | 9.0E+03 | 4.2E+04 | 3.3E+04 | 7.3E-01 | 3.6E-02 | 2.1E-02 | 3.6E-02 | 5.1E-03 | 1.5E-02 | 3.6E-03 | 1.9E-03 | 8.1E-03 | Leading |
| | 2 | 4.2E+04 | 9.5E+04 | 5.3E+04 | 2.4E-01 | 3.2E-02 | 4.2E-03 | 7.2E-03 | 2.6E-04 | 7.1E-04 | 4.0E-04 | 3.6E-04 | 4.4E-04 | Midpoint |
| | 3 | 9.5E+04 | 1.6E+05 | 6.8E+04 | 2.2E-02 | 1.6E-01 | 1.2E-02 | 2.9E-02 | 1.2E-03 | 4.0E-05 | 5.2E-05 | 1.6E-04 | 1.5E-03 | Midpoint |
| | 4 | 1.6E+05 | 2.5E+05 | 8.6E+04 | 5.4E-03 | 4.7E-02 | 5.5E-03 | 5.9E-03 | 1.1E-03 | 6.1E-05 | 5.6E-05 | 2.5E-04 | 1.1E-03 | Midpoint |
| RC3 | 1 | 1.7E+05 | 2.1E+05 | 4.1E+04 | 9.4E-01 | 4.7E-01 | 4.6E-01 | 4.2E-01 | 4.2E-02 | 2.7E-01 | 1.5E-03 | 6.3E-03 | 1.0E-01 | Leading |
| | 2 | 2.1E+05 | 2.6E+05 | 4.4E+04 | 4.7E-02 | 8.4E-03 | 6.5E-03 | 6.4E-03 | 1.8E-03 | 4.9E-03 | 6.6E-05 | 8.7E-05 | 3.5E-03 | Leading |
| | 3 | 2.6E+05 | 3.4E+05 | 8.3E+04 | 1.5E-03 | 1.0E-03 | 1.1E-03 | 2.8E-03 | 4.4E-04 | 1.8E-04 | 6.4E-06 | 6.0E-05 | 2.2E-04 | Midpoint |
| | 4 | 3.4E+05 | 4.2E+05 | 8.6E+04 | 5.5E-04 | 2.5E-04 | 1.8E-05 | 1.5E-03 | 5.4E-05 | 0.0E+00 | 2.3E-07 | 2.8E-06 | 2.4E-05 | Midpoint |
| RC4 | 1 | 7.8E+04 | 9.4E+04 | 1.6E+04 | 1.0E+00 | 3.8E-02 | 3.3E-02 | 4.9E-02 | 4.5E-03 | 2.4E-02 | 1.2E-04 | 3.7E-04 | 2.3E-02 | Leading |
| | 2 | 9.4E+04 | 1.3E+05 | 3.2E+04 | 1.6E-03 | 1.7E-02 | 8.6E-03 | 3.8E-03 | 3.0E-04 | 2.8E-03 | 6.8E-07 | 3.5E-06 | 5.6E-04 | Midpoint |
| | 3 | 1.3E+05 | 2.1E+05 | 8.6E+04 | 2.7E-04 | 7.5E-03 | 3.4E-03 | 7.8E-03 | 1.3E-03 | 1.1E-05 | 1.5E-05 | 4.7E-04 | 4.7E-04 | Leading |
| | 4 | 2.1E+05 | 3.0E+05 | 8.6E+04 | 1.0E-04 | 6.3E-03 | 1.1E-03 | 2.8E-03 | 1.5E-06 | 0.0E+00 | 3.0E-08 | 9.6E-07 | 1.0E-06 | Midpoint |
| RC5 | 1 | 1.9E+05 | 2.0E+05 | 1.0E+04 | 9.3E-01 | 2.7E-03 | 1.1E-03 | 6.4E-03 | 8.1E-05 | 9.9E-05 | 3.0E-05 | 1.9E-05 | 6.6E-05 | Leading |
| | 2 | 2.0E+05 | 2.6E+05 | 6.0E+04 | 3.5E-02 | 2.2E-02 | 4.2E-03 | 2.5E-03 | 1.4E-06 | 1.9E-06 | 5.3E-07 | 3.4E-07 | 1.6E-06 | Midpoint |
| | 3 | 2.6E+05 | 3.4E+05 | 8.3E+04 | 1.8E-02 | 6.0E-02 | 8.0E-03 | 3.1E-03 | 5.1E-07 | 1.7E-06 | 5.7E-08 | 4.6E-08 | 1.3E-06 | Midpoint |
| | 4 | 3.4E+05 | 4.3E+05 | 8.6E+04 | 6.5E-03 | 5.7E-02 | 6.4E-03 | 4.6E-03 | 1.6E-06 | 9.2E-07 | 2.1E-09 | 1.3E-08 | 3.7E-06 | Midpoint |
| RC6 | 1 | 1.3E+03 | 1.5E+04 | 1.4E+04 | 1.2E-04 | 1.7E-06 | 1.7E-06 | 1.3E-06 | 1.6E-07 | 6.3E-07 | 3.2E-09 | 5.3E-09 | 2.4E-07 | Leading |
| | 2 | 1.5E+04 | 8.8E+04 | 7.3E+04 | 6.5E-04 | 1.5E-09 | 0.0E+00 | 7.0E-09 | 1.8E-08 | 6.5E-09 | 2.9E-10 | 2.8E-10 | 2.5E-08 | Midpoint |
| | 3 | 8.8E+04 | 1.7E+05 | 8.6E+04 | 6.9E-04 | 1.9E-09 | 0.0E+00 | 5.1E-10 | 0.0E+00 | 0.0E+00 | 0.0E+00 | 0.0E+00 | 0.0E+00 | Midpoint |
| | 4 | 1.7E+05 | 2.6E+05 | 8.6E+04 | 6.5E-04 | 0.0E+00 | 0.0E+00 | 8.9E-11 | 6.5E-11 | 4.4E-11 | 4.6E-13 | 1.2E-12 | 6.4E-11 | Midpoint |

**Table 9 – Elements Included in Each MACCS2 Fission Product Group**

| Group | Release Class (Isotopes Included) |
|---|---|
| 1 | Noble Gases (Kr, Xe) |
| 2 | Iodine (I) |
| 3 | Cesium (Cs, Rb) |
| 4 | Tellurium (Te, Sb) |
| 5 | Strontium (Sr) |
| 6 | Ruthenium (Co, Mo, Tc, Ru, Rh) |
| 7 | Lanthanum (Y, Zr, Nb, La, Pr, Nd, Am, Cm) |
| 8 | Cerium (Ce, Np, Pu) |
| 9 | Barium (Ba) |

**Table 10a – Internal Events at Power – Population Dose Risk**

| Release Category | Release Frequency (per reactor-year) | Population Dose (person-Sv) | Population Dose (person-rem) | Risk (person-rem/ reactor-year) | Percentage Contribution to Total Risk, (%) |
|---|---|---|---|---|---|
| RC1 | 7.5E-09 | 2.4E+04 | 2.4E+06 | 1.8E-02 | 6.7 |
| RC2 | 2.1E-09 | 2.3E+04 | 2.3E+06 | 4.9E-03 | 1.8 |
| RC3 | 2.0E-08 | 1.2E+05 | 1.2E+07 | 2.4E-01 | 86.3 |
| RC4 | 1.1E-08 | 1.3E+04 | 1.3E+06 | 1.4E-02 | 5.2 |
| RC5 | 6.5E-08 | 0.0E+00 | 0.0E+00 | 0.0E+00 | 0.0 |
| RC6 | 1.1E-06 | 4.9E-01 | 4.9E+01 | 5.4E-05 | 0.0 |
| | | | Total Risk | 2.7E-01 | 100.0 |

Note:RC5 is not evaluated because there is no release within 24 hours after the onset of core damage.

**Table 10b – Internal Events at Power – Offsite Property Damage Risk**

| Release Category | Release Frequency (per reactor-year) | Property Damage cost, ($) | Risk ($/reactor-year) | Percentage Contribution to Total Risk, (%) |
|---|---|---|---|---|
| RC1 | 7.5E-09 | 2.2E+08 | 1.6E+00 | 18.2 |
| RC2 | 2.1E-09 | 1.8E+08 | 3.7E-01 | 4.1 |
| RC3 | 2.0E-08 | 2.6E+08 | 5.2E+00 | 57.7 |
| RC4 | 1.1E-08 | 1.6E+08 | 1.8E+00 | 19.8 |
| RC5 | 6.5E-08 | 0.0E+00 | 0.0E+00 | 0.0 |
| RC6 | 1.1E-06 | 9.9E+03 | 1.1E-02 | 0.1 |
| | | Total Risk | 8.9E+00 | 100.0 |

Note : RC5 is not evaluated because there is no release within 24 hours after the onset of core damage.

**Table 11 – Uses of PRA in the Design Process (Sheet 1 of 6)**

| Cause of core damage or large release | | Features and requirements to reduce or eliminate weaknesses in current reactor design |
|---|---|---|
| 1 | Loss of ECCS function | Redundancy<br><br>・ Highly redundant safety injection system design with four advanced accumulators and independent four train HHIS enhances the reliability of safety injection function. In addition, feed and bleed operation is available with one of four HHIS.<br><br>Diversity<br><br>・ Alternate core cooling/injection utilizing CSS/RHRS is available in case all safety injection fail. |
| 2 | Loss of ECCS recirculation function | Simplicity<br><br>・ In-containment RWSP is incorporated which results in elimination of switchover to recirculation operation. Reliability of core cooling is enhanced due to simplified operation mode. |
| 3 | Loss of containment cooling | Redundancy<br><br>・ Independent four-train design adapted to the CSS/RHRS enhances reliability of containment spray and RHR function.<br><br>Diversity<br><br>・ Alternate containment cooling operation utilizing containment fan cooler unit and CCWS enhances the reliability of containment cooling function. |
| 4 | Loss of secondary side cooling | Redundancy<br><br>・ Highly redundant EFWS design with two turbine driven EFW pumps and two motor driven EFW pumps enhances the reliability of secondary side cooling. |

**Table 11 – Uses of PRA in the Design Process (Sheet 2 of 6)**

| Cause of core damage or large release | | Features and requirements to reduce or eliminate weaknesses in current reactor design |
|---|---|---|
| 5 | Loss of support system function | <u>Redundancy</u><br>• Four-train CCWS/ESWS design enhances the reliability of CCWS. Furthermore, CCWS is physically separated into two subsystems to reduce the dependency between trains.<br>• Independent four-train electrical system design with four gas turbine emergency generators provides emergency power to each dedicated safety systems. High redundancy and independency enhances the reliability of power supply to safety systems.<br><u>Diversity</u><br>• Alternate component cooling water utilizing fire protection water supply system or the non-essential chilled water system enables to maintain CCW supply to charging pump during loss of CCW events. Thus RCP seal injection function is available under loss of CCW and occurrence of RCP seal LOCA is reduced.<br>• Alternate ac power supported by two non-Class 1E GTGs is incorporated as a countermeasure against SBO. Alternate ac power can supply power to any two of the four safety buses in case class 1E GTGs fail during loss of offsite power. |
| 6 | Failure of reactor trip | <u>Redundancy</u><br>• Independent four-train design of reactor protection systems enhances reliability of plant trip. Four redundant measurements using sensors from the four separate trains are made for each variable used for reactor trip.<br><u>Diversity</u><br>• The DAS, which has functions to prevent ATWS, is installed as a countermeasure to CCF of the digital I&C systems. |

**Table 11 – Uses of PRA in the Design Process (Sheet 3 of 6)**

| Cause of core damage or large release | | Features and requirements to reduce or eliminate weaknesses in current reactor design |
|---|---|---|
| 7 | Interfacing systems LOCA | <u>Prevention</u><br>・ Higher rated piping of residual heat removal systems reduces the occurrence of interfacing systems LOCA. Even if residual heat removal system isolation valves open due to malfunction during normal operation, reactor coolant from main coolant pipe would flow to refueling water storage pit without pipe break outside containment. |
| 8 | Loss of RHR function during plant shutdown | <u>Redundancy</u><br>・ Independent four-train design of RHRS is adapted to enhance reliability of RHR function.<br><u>Diversity</u><br>・ As a countermeasure for loss of RHR, RCS makeup by gravity injection from spent fuel pit is available when the RCS in atmospheric pressure.<br><u>Prevention</u><br>・ To prevent over-drain during mid-loop operation, a loop water level gage and an interlock (actuated by the detection of water level decrease), is provided to isolate water extraction. |
| 9 | Internal fire | <u>Physical separation</u><br>・ Safety related SSCs are physical separated into four independent divisions and thus fire propagation through trains is minimized.<br>・ Divide the electrical room of T/B with two fire compartments. |
| 10 | Internal flood | <u>Physical separation</u><br>・ R/B is divided to two divisions (e.g. east side and west side) and thus flood propagation to all four trains is prevented. |

**Table 11 – Uses of PRA in the Design Process (Sheet 4 of 6)**

| Cause of core damage or large release | | Features and requirements to reduce or eliminate weaknesses in current reactor design |
|---|---|---|
| 11 | Hydrogen combustion | High reliability<br><br>• Reliability of combustible gas control is enhanced by providing Igniters that automatically start with the safety injection signal. Power supply from two non-Class 1E buses with alternative ac generators also enhances reliability of combustible gas control.<br><br>Inherent margin of safety<br><br>• Large volume containment provides combustible gas mixing and protection against hydrogen burns. |
| 12 | Steam explosion | Inherent margin of safety<br><br>• There are no mitigation features against in- and ex-vessel steam explosions. However, robust structure of the containment vessel reduces the possibility of containment failure following steam explosions. |
| 13 | High pressure melt ejection | High reliability<br><br>• A series of depressurization valves which is independent of safety depressurization valves enhances reliability of RCS pressure reduction and reduces possibility of high pressure melt ejection.<br><br>Defense in depth<br><br>• Even if high pressure melt ejection occurs, mitigation features against the challenges to containment failure are provided.<br><br>Diversity<br><br>• For direct containment heating, core debris trap enhances capturing of ejected molten core in the reactor cavity. Debris entrainment is also prevented by reactor cavity flooding systems such as drain line injection from SG compartment and firewater injection. |

**Table 11 – Uses of PRA in the Design Process (Sheet 5 of 6)**

| Cause of core damage or large release | | Features and requirements to reduce or eliminate weaknesses in current reactor design |
|---|---|---|
| 13 | High pressure melt ejection (cont.) | Inherent margin of safety<br><br>・ There are no mitigation features against containment failure accompanied by rocket-mode reactor vessel failure. However, robust structure of the containment vessel reduces the possibility of containment failure following steam explosions. |
| 14 | Temperature-induced SGTR | High reliability<br><br>・ A series of depressurization valves which is independent of safety depressurization valves enhances reliability of RCS pressure reduction and reduces possibility of temperature-induced SGTR. |
| 15 | MCCI | High reliability<br><br>・ Diverse cavity flooding system enhances heat removal from molten core ejected into the reactor cavity where sufficient floor area and appropriate depth ensure spreading debris bed for better coolability. Reactor cavity floor concrete is also provided to protect against challenge to liner plate melt through.<br><br>Diversity<br><br>・ Diverse cavity flooding system consists of drain line injection from SG compartment and firewater injection.<br><br>Inherent margin of safety<br><br>・ Basemat concrete protects against fission products release to the environment. |

**Table 11 – Uses of PRA in the Design Process (Sheet 6 of 6)**

| Cause of core damage or large release | | Features and requirements to reduce or eliminate weaknesses in current reactor design |
|---|---|---|
| 16 | Long-term containment overpressure | <u>Diversity</u><br><br>・ Containment spray mitigates overpressure in the containment. Alternate containment cooling also removes decay heat accumulated in the steam. Firewater injection to spray header, which dose not have a function of heat removal, delays containment failure and ensure the time to recovery of containment spray.<br><br><u>Inherent margin of safety</u><br><br>・ Large volume containment provides sufficient capability to withstand overpressure. |
| 17 | Containment isolation failure | <u>High reliability</u><br><br>・ Main penetrations are isolated automatically even when SBO occurs and alternative ac generators are not available.<br><br><u>Diversity</u><br><br>・ Manual closure of isolation valves is available using DAS even when automatic isolation fails due to software common cause failure. |

## Table 12 – SAMDA Benefit Sensitivity Analyses

| | Design Alternative | Cost Impact | Maximum Averted Benefit | Sensitivity of each SAMDA benefit | | | |
|---|---|---|---|---|---|---|---|
| | | | | Baseline | Discount rate | | Monetary equivalent of unit dose ($3000/person-rem) |
| | | | | | 5% | 3% | |
| 1 | Provide additional dc battery capacity | $2,000k | $289k | $116k | $188k | $304k | $124k |
| 2 | Provide an additional gas turbine generator | $10,000k | | $116k | $188k | $304k | $124k |
| 3 | Install an additional, buried off-site power source | $10,000k | | $118k | $193k | $312k | $127k |
| 4 | Provide an additional high pressure injection pump with independent diesel | $1,000k | | $150k | $244k | $395k | $161k |
| 5 | Add a service water pump | $5,900k | | $72k | $118k | $190k | $78k |
| 6 | Install an independent reactor coolant pump seal injection system, with dedicated diesel | $3,800k | | $136k | $221k | $357k | $146k |
| 7 | Install an additional component cooling water pump | $1,500k | | $72k | $118k | $190k | $78k |
| 8 | Add a motor-driven feed-water pump | $2,000k | | $101k | $165k | $266k | $109k |
| 9 | Install a filtered containment vent to remove decay heat | $3,000k | | $173k | $282k | $455k | $186k |
| 10 | Install a redundant containment spray system | $870k | | $14k | $22k | $36k | $15k |

# APPENDIX A OFFSITE POPULATION DOSE AND PROPERTY DAMAGE RISK QUANTIFICATION METHODOLOGY

## A.1 METHODOLOGY OVERVIEW

For the US-APWR internal events at full power, offsite population total effective dose equivalent (TEDE) and property damage risk quantification within 50 miles are performed with the MACCS2 version 1.13.1 (Reference A-1). The purpose of the offsite population dose and property damage risk quantification prepares input data for the severe accident mitigation design alternative (SAMDA) analysis.

On the design certification application stage, site specific data is not prepared, therefore the meteorological data of the Surry site has been used as “typical”, which is accessible in the MACCS2 code sample input file attached to the MACCS2 code. Additionally, the population data of the Surry site in the MACCS2 code sample input file has been adjusted to be representative of about 80% of the U.S. nuclear plant sites in NUREG/CR-2239, “Technical Guidance for Siting Criteria Development” (Reference A-2).

The population dose risk is calculated by multiplying the release category frequency by the mean value of the consequence result. Therefore, the overall population dose risk is the sum of the six release category risks and is reported in terms of person-sievert/year (person-Sv/y). Similarly, the offsite property damage risk is calculated based on the sum of the six individual mean property damage risks and reported in dollars/per reactor-year, ($/reactor-year).

## A.2 INTERFACE WITH CONTAINMENT PERFORMANCE ASSESSMENT

The analysis for the US-APWR internal events at full power is executed against 6 release categories which are shown in Table A-1.

- Containment Bypass (RC1)
- Containment Isolation Failure (RC2)
- Containment Failure before Core Damage (RC3)
- Early Containment Failure (RC4)
- Late Containment Failure (RC5)
- Intact Containment (RC6)

The frequency of the release categories is described in Table 7.

## A.3 EVALUATION OF FISSION PRODUCT SOURCE TERMS

The fission product source term assessed by the MAAP 4.0.6 code (Reference A-3) is provided as the accidental release conditions. These conditions include the fission product release fractions, release height, release energy, and release duration. They are listed as follows.

1) Release Fractions

Results of MAAP 4.0.6 analysis for nuclide release fraction are reconstructed to match for MACCS2 nuclide groups, because nuclide groups of MAAP 4.0.6 and MACCS2 are different.

The radionuclide groups are separated by nuclide's chemical specific. Table A-2 shows the radionuclide groups of MACCS2. Table A-3 shows correspondence of MAAP 4.0.6 and MACCS2 radionuclide groups. The fission product group and inventory fraction released, timing and duration for each of 6 release categories are shown in Table 8.

2) Release Height
Release height is established 0 meter from ground conservatively.
3) Release Energy
Release Energy is a parameter which has influence for plume ascent, when plume is released. Release Energy is established 0 watt conservatively.
4) Release Duration
In the MACCS2 code, the state of the nuclide source in plume is imitated by setting the release duration and release fraction of each plume. The number of plume sets 4 groups.

## A.4 DOSE CONSEQUENCE MODELING

The MACCS2 version 1.13.1 developed by NRC is used for the offsite population dose quantification. The MACCS2 code includes three modules: ATMOS, EARLY, and CHRONC modules. The ATMOS module treats the source term and performs the atmospheric dispersion analysis. The EARLY module performs the offsite population dose calculations during the emergency phase. The offsite population dose are calculated during 24 hours following the onset of core damage The CHRONC module has been used for offsite property damage risk evaluation.

The calculation of radiation doses from early exposure considers four pathways: (1) direct external exposure to radioactive material in the plume (cloud-shine), (2) exposure from inhalation of radio nuclides in the cloud (cloud inhalation), (3) exposure to radioactive material deposited on the ground (ground-shine), (4) inhalation of re-suspended material (re-suspension inhalation). Cloud-shine and cloud inhalation exposures are limited to the time of cloud passage. Ground-shine and re-suspension inhalation doses for early exposure are limited to the duration of the emergency phase.

In general, the dose equation for an early exposure pathway in MACCS2 in a given spatial element is the product of the following quantities: radionuclide concentration, dose conversion factor, duration of exposure, and shielding factor. The quantities used in the dose equations depend on the exposure pathway. For example, for the cloud inhalation exposure pathway, these quantities are the ground-level air concentration at a spatial element, inhalation dose conversion factor, duration of exposure, and inhalation shielding factor.

The CHRONC module calculates the property damage costs of all the long-term protective actions as well as the cost of the emergency response actions that are modeled by EARLY module. The property damage cost is consist of population-dependent costs and farm-dependent costs. Population-dependent costs are summed up population-dependent decontamination, interdiction, and condemnation costs. Farm-dependent costs are summed up farm-dependent decontamination, interdiction, and condemnation costs as well as milk and crop disposal costs.

### A.4.1 METEOROLOGICAL DATA

In the analysis, the meteorological data of the Surry site has been used as “typical”, which is accessible in the MACCS2 code sample input file attached to the MACCS2 code. Since the Surry site is not very near the ocean, the sea-breeze effects are not included (which is good) and the terrain is flat which is more typical than sites with complex river valley effects. Average wind speeds are also not excessively high (i.e.: they are not too high which would be non-conservative).

### A.4.2 POPULATION DATA

In the analysis, the population data of the Surry site in the MACCS2 code sample input file has been adjusted to be representative of about 80% of the U.S. nuclear plant sites in NUREG/CR-2239, “Technical Guidance for Siting Criteria Development”. Table A-4 shows the population data using for this evaluation.

### A4.3 CHRONC MODULE DATA

The CHRONC module in MACCS2 performs the intermediate and long-term (for many years) offsite population dose and property damage cost calculations. The CHRONC module accounts for decontamination, relocation, contaminated crops, and food and water ingestion.
The CHRONC module data such as decontamination, relocation, property damage , and agricultural data is left to be the Surry data in the sample problem provided with the MACCS2 code package.

## A.5 QUANTIFICATION AND RESULTS

### A.5.1 DOSE AND COST EVALUATION RESULTS

The results of the MACCS2 analysis are indicated as the offsite population dose (TEDE, person-rem) and property damage cost within 50 miles. Dose and property damage cost are based on the probabilistic atmospheric dispersion analysis coupled with annual meteorological conditions. In this report, the results are expressed as a mean value. The offsite population dose is described in Table 10a and the offsite property damage cost is described in Table 10b. Release category (RC) 3 that is containment failure before core damage due to loss of heat removal is the dominant category for the offsite population dose and property damage cost.

### A.5.2 RISK QUANTIFICATION

The offsite population dose and property damage risk quantification for the US-APWR internal events at full power is executed for the release categories. The population dose risk is calculated by multiplying the release category frequency by the mean value of the consequence result. Therefore, the overall population dose risk is the sum of the six release category risks and is reported in terms of person-sievert/year (person-Sv/y). Similarly, the offsite property damage risk is calculated based on the sum of the six individual mean property damage risks and reported in dollars/per reactor-year, ($/reactor-year) The consequence is used the mean value on the probabilistic atmospheric dispersion analysis. Table 10a shows the results of the offsite population dose risk and Table 10b shows the results of the offsite property damage risk. RC 3 that is containment failure before core damage due to loss of heat removal is the dominant category for the risk.

## A.6 SENSITIVITY ANALYSIS

RC 3 is the dominant category for the offsite population dose and property damage cost, therefore sensitivity analysis is performed for the RC 3. Source term of the RC 3 is based on a LOCA event, but in sensitivity analysis, source term is selected a transient event one. The comparison of the base case and the sensitivity analysis case for the RC 3 source term is shown in Table A-5. The results of sensitivity analysis show that the source term change form a LOCA event to a transient event occurs about one order decrease in the offsite population dose (See Table A-6).

## A.7 CONCLUSION AND INSIGHTS RELATED TO OFFSITE CONSEQUENCES EVALUATION

For the US-APWR internal events at full power, the offsite population dose and property damage risk quantification are analyzed with the MACCS2 version 1.13.1 (Reference A-1). Total offsite dose risk is about 0.27 person-rem per reactor-year and total property damage risk is about $8.9 per reactor-year. This offsite population dose and property damage risk prepares input data for the severe accident mitigation design alternative (SAMDA) analysis.

## A.8 REFERENCES

A-1 NUREG/CR-6613, Vol. 1 (SAND97-0594), Code Manual for MACCS2 User's Guide, U.S. Nuclear Regulatory Commission, Washington, DC, and Sandia National Laboratories, May 1998.

A-2 NUREG/CR-2239, Technical Criteria for Siting Criteria Development, U.S. Nuclear Regulatory Commission, Washington, DC, and Sandia National Laboratories, December 1982.

A-3 FAI/05-47, MAAP4 Modular Accident Analysis Program for LWR Power Plants, Transmittal Document for Maap4 Code revision MAAP 4.0.6, Rev. 0, Electric Power Research Institute, 2005.

**Table A-1 Description of release categories**

| Designator | Description |
|---|---|
| RC1 | Containment bypass which includes both core damage after SGTR and thermal induced SGTR after core damage |
| RC2 | Containment isolation failure |
| RC3 | Containment overpressure failure before core damage due to loss of heat removal |
| RC4 | Early containment failure due to dynamic loads which includes hydrogen combustion before or just after reactor vessel failure, in-vessel or ex-vessel steam explosion, rocket-mode RV failure and containment direct heating |
| RC5 | Late containment failure which includes containment overpressure failure after core damage, hydrogen combustion long after reactor vessel failure and basemat melt through |
| RC6 | Intact containment in which fission products are released at design leak rate |

**Table A-2 Group of Isotope in Each Radionuclide Release Class**

| Group | Release Class (Isotopes Included) |
|---|---|
| Group 1 | Gases (Kr, Xe) |
| Group 2 | Iodine (I) |
| Group 3 | Cesium (Cs + Rb) |
| Group 4 | Tellurium (Te + Sb) |
| Group 5 | Strontium |
| Group 6 | Ruthenium (Co + Mo + Tc + Ru + Rh) |
| Group 7 | Lanthanum (Y + Zr + Nb + La + Pr + Nd + Am + Cm) |
| Group 8 | Cerium (Ce + Np + Pu) |
| Group 9 | Barium (Ba) |

**Table A-3 Correspondence of MAAP and MACCS2 Radionuclide Groups**

| MAAP Analysis | |
|---|---|
| Group 1 | Noble Gases |
| Group 2 | CsI+RbI |
| Group 3 | $TeO_2$ |
| Group 4 | SrO |
| Group 5 | $MoO_2+RuO_2+TcO_2$ |
| Group 6 | CsOH+RbOH |
| Group 7 | BaO |
| Group 8 | $La_2O_3+Pr_2O_3+Nd_2O_3+Sm_2O_3+Y_2O_3+ZrO_2+NbO_2$ |
| Group 9 | $CeO_2+NpO_2+PuO_2$ |
| Group 10 | Sb |
| Group 11 | $Te_2$ |
| Group 12 | $UO_2$ |

| MACCS2 Analysis | |
|---|---|
| Group 1 | Noble Gases |
| Group 2 | Iodine |
| Group 3 | Cesium |
| Group 4 | Tellurium |
| Group 5 | Strontium |
| Group 6 | Ruthenium |
| Group 7 | Lanthanum |
| Group 8 | Cerium |
| Group 9 | Barium |

**Table A-4 Population Input Data for MACCS2 code**

| Direction | Radial Ring Distance, Miles | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | 0-1 | 1-2 | 2-3 | 3-5 | 5-10 | 10-20 | 20-30 | 30-40 | 40-50 |
| N | 0 | 0 | 74 | 246 | 7952 | 6961 | 5837 | 5704 | 17211 |
| NNE | 0 | 111 | 180 | 619 | 3971 | 11054 | 7262 | 18339 | 16937 |
| NE | 0 | 0 | 90 | 309 | 1959 | 15857 | 19292 | 3438 | 0 |
| ENE | 0 | 0 | 17 | 103 | 1294 | 17588 | 0 | 250 | 21276 |
| E | 0 | 0 | 0 | 0 | 11416 | 60897 | 0 | 4488 | 5594 |
| ESE | 0 | 0 | 17 | 64 | 19789 | 171209 | 241590 | 155741 | 183089 |
| SE | 0 | 0 | 0 | 0 | 0 | 87319 | 311911 | 721395 | 217651 |
| SSE | 0 | 0 | 33 | 127 | 304 | 19284 | 44175 | 36836 | 10703 |
| S | 499 | 1162 | 278 | 920 | 724 | 2432 | 12673 | 28403 | 12968 |
| SSW | 0 | 0 | 360 | 1158 | 114 | 2865 | 6403 | 34157 | 15691 |
| SW | 0 | 111 | 278 | 920 | 375 | 2448 | 9144 | 298 | 16870 |
| WSW | 0 | 111 | 278 | 920 | 266 | 2259 | 5451 | 6107 | 13162 |
| W | 0 | 0 | 475 | 1523 | 76 | 2807 | 1641 | 128856 | 130291 |
| WNW | 0 | 0 | 131 | 468 | 76 | 1313 | 5217 | 62866 | 583358 |
| NW | 0 | 0 | 0 | 0 | 532 | 3257 | 12326 | 12769 | 308532 |
| NNW | 0 | 0 | 286 | 912 | 5959 | 14519 | 8845 | 4036 | 19973 |

**Table A-5 Sensitivity Analysis Source Term Data**

| | Plume No. | START TIME (seconds) | END TIME (seconds) | Duration (seconds) | Release Fraction (MACCS2 Group) | | | | | | | | | Plume Position |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | 1 Kr/Xe | 2 I | 3 Cs | 4 Te/Sb | 5 Sr | 6 Ru | 7 La | 8 Ce | 9 Ba | |
| RC3 | 1 | 1.7E+05 | 2.1E+05 | 4.1E+04 | 9.4E-01 | 4.7E-01 | 4.6E-01 | 4.2E-01 | 4.2E-02 | 2.7E-01 | 1.5E-03 | 6.3E-03 | 1.0E-01 | Leading |
| | 2 | 2.1E+05 | 2.6E+05 | 4.4E+04 | 4.7E-02 | 8.4E-03 | 6.5E-03 | 6.4E-03 | 1.8E-03 | 4.9E-03 | 6.6E-05 | 8.7E-05 | 3.5E-03 | Leading |
| | 3 | 2.6E+05 | 3.4E+05 | 8.3E+04 | 1.5E-03 | 1.0E-03 | 1.1E-03 | 2.8E-03 | 4.4E-04 | 1.8E-04 | 6.4E-06 | 6.0E-05 | 2.2E-04 | Midpoint |
| | 4 | 3.4E+05 | 4.2E+05 | 8.6E+04 | 5.5E-04 | 2.5E-04 | 1.8E-05 | 1.5E-03 | 5.4E-05 | 0.0E+00 | 2.3E-07 | 2.8E-06 | 2.4E-05 | Midpoint |
| RC3 Sensitivity Analysis | 1 | 1.7E+05 | 2.2E+05 | 5.3E+04 | 9.9E-01 | 4.8E-02 | 3.9E-02 | 4.8E-02 | 5.3E-03 | 1.6E-02 | 2.1E-04 | 1.6E-04 | 9.5E-03 | Leading |
| | 2 | 2.2E+05 | 2.7E+05 | 4.7E+04 | 9.5E-03 | 1.4E-03 | 2.9E-03 | 3.2E-03 | 5.8E-04 | 2.9E-04 | 8.5E-05 | 5.2E-05 | 6.6E-04 | Leading |
| | 3 | 2.7E+05 | 3.0E+05 | 3.6E+04 | 4.5E-05 | 1.2E-04 | 1.2E-04 | 1.3E-04 | 1.1E-06 | 1.8E-06 | 3.8E-06 | 2.4E-06 | 1.6E-06 | Leading |
| | 4 | 3.0E+05 | 3.9E+05 | 8.6E+04 | 0.0E+00 | 1.3E-03 | 1.2E-03 | 1.1E-02 | 1.5E-04 | 0.0E+00 | 4.7E-07 | 1.5E-05 | 6.6E-05 | Leading |

**Table A-6 Result of Sensitivity Analysis (RC3)**

| Case | Mean Dose (Person- rem) | Property Damage Cost($) |
|---|---|---|
| Base case | 1.2E+07 | 2.6E+08 |
| Sensitivity Analysis | 8.2E+05 | 1.5E+08 |