SOARCA Peer Review Comments

(Feb. 25, 2010)
DSE

2/12

SOARCA Peer Review Comments

(Feb. 25, 2010)

Peer Review Comment Mtg. (3/25/2020)

- Kathy's ? on EPRI Study similar to SOARCA

- Peer Review Mtg. #4.

- Should a better term than "mitigation" be used.

- When ~~can we~~ is a good block of time to develop a sense of what is being discussed by the Peer Review (ACTION)

- We need to determine how to communicate SOARCA to other Federal agencies.

- Red-Line strickout version

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PREDECISIONAL Revision 1 - 2/14/2010 1:20:00 AM since there are no current full scope level 3 PRAs generally available, considering both internal and external events, to draw upon. However, the preponderance of level 1 PRA information, combined with our insights on severe accident behavior, is available on dominant core damage sequences, especially internal event sequences. This information, combined with our understanding of containment loadings and failure mechanisms together with radionuclide release, transport and deposition, allow us to utilize core damage frequency (CDF) as a surrogate criterion for risk. Thus, for SOARCA we elected to analyze sequences with a CDF greater than 10^{-6} per reactor-year. In addition, we included sequences that have an inherent potential for higher consequences (and risk), with a lower CDF - those with a frequency greater than 10^{-7} per reactor-year. Such sequences would be associated with events involving containment bypass or leading to an early failure of the containment. By the adoption of these criteria, we are reasonably assured that the more probable and important core melt sequences will be captured.	
The application of the screening criteria to the available level 1 PRA information for the pilot plants resulted in the identification of two basic types of scenarios – station blackouts and bypass scenarios. This result presents certain advantages with respect to consideration of the inherent adequacy of our criteria and the adequacy of the scope of scenarios analyzed. First, station blackout scenarios are representative of a broad class of events in PRA – loss of heat removal events. Selection of SBO events in SOARCA insures that we have covered that broader class of transients involving a loss of heat removal, and further, by including a short term blackout we have reasonably bounded that class of accidents (which could include other events such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the station blackout also includes, in part, the effect of a small loss of coolant by considering reactor coolant pump seal leakage. Additionally, by the selection of station blackout sequences for analysis we also include the effects of loss of containment heat removal (fan coolers) and loss of containment spray systems (which are all electrically powered) to remove airborne radionuclides. Thus, our non-bypass sequences also result in containment failure which would not be the case for all other such loss of heat removal transients in a typical PRA. Therefore, while we have used CDF for screening, in effect the CDF in these cases also represents the radionuclide release frequency.	ensures
While we have not included medium or large loss of inventory accidents – because of their very low frequency – it should be noted that such internal events were well below our selection criteria for the BWR and comfortably below our screening criterion of 10^{-6} for the PWR plant. For Peach Bottom the medium and large LOCAs had frequencies of $2x10^{-9}$ and $1x10^{-9}$/ry. For Surry the medium and large LOCAs had frequencies of $6x10^{-8}$ and $7x10^{-10}$/ry. Only a fraction of these sequences would have resulted in containment failure because there may not have been a loss of containment heat removal. Since for Surry we have included an ISLOCA sequence it can also be argued that we have also reasonably bounded events involving a LOCA inside containment for that plant.	Page xv: Both of the reference to "LOCA frequencies" should be changed to "LOCA core damage sequence frequencies."
All the sequences identified in the SOARCA study are significant in an absolute sense. The American Society of Mechanical Engineers' "Standard for Probabilistic Risk Assessment for Nuclear Power Plants," ASME RA-Sb-2005, which was endorsed by the staff in Regulatory Guide 1.200, defines a significant sequence, in part, as one that individually contributes more xv	Page XV: Last paragraph mentions the PRA quality requirements in ASME and RG 1.200 but there is no communication about the level of compliance of the SPAR models used in the analysis. Rev 2 of RG 1.200 includes explicit requirements related to external events. The SOARCA project analyses include significant seismic events. Therefore, a comparison with the Rev 2 of RG 1.200 requirements might be included in the final report. Page xv: The reference to ASME RA-Sb-2005 should be updated to the most recent version of the PRA Standard (ASME/ANS RA-SA-2009).

PAGE 19	COMMENTS
PREDECISIONAL Revision 1 - 2/14/2010 1:20:00 AM **2.4.1 Peach Bottom Internal Event Scenarios** No internal event scenarios for Peach Bottom met the criteria for further evaluation. **2.4.2 Peach Bottom External Event Scenarios** 1. *Initiating Event:* Seismic-Initiated Long-Term Station Blackout *Representative Frequency:* 1×10^{-6} to 5×10^{-6} per reactor-year <u>Scenario Summary:</u> This scenario is initiated by a moderately large earthquake (0.3–0.5 pga). The seismic event results in a LOOP, failure of onsite emergency AC power and failure of the Conowingo Dam power line resulting in a SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray). The turbine-driven injection systems, high pressure coolant injection (HPCI) and/or reactor core isolation cooling (RCIC), are available initially. Loss of room cooling and/or battery depletion results in eventual failure of these systems leading to core damage.	
2. *Initiating Event:* Seismic-Initiated Short-Term Station Blackout *Representative Frequency:* 1×10^{-7} to 5×10^{-7} per reactor-year <u>Scenario Summary:</u> This scenario is initiated by a large earthquake (0.5–1.0 pga). The seismic event results in a LOOP, failure of onsite emergency AC power and failure of the Conowingo Dam power line resulting in a SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray). In addition, HPCI and RCIC are unavailable due to loss of DC power. <u>Notes:</u> This following scenario does not meet the SOARCA screening criterion of 1×10^{-6} per reactor-year; however, the scenario was retained for analysis in order to assess the risk importance of a lower frequency, higher consequence scenario. This type of scenario has been a risk-important severe accident scenario in past PRA studies and, at a frequency of 5×10^{-7} per reactor-year; it is only a factor of two below the screening criterion. **2.5 Generic Factors** The results of existing PRAs indicate that the likelihood of a nuclear power plant accident sequence that releases a significant amount of radioactivity is very small due to the diverse and redundant barriers and numerous safety systems in the plant; the training and skills of the reactor operators; testing and maintenance activities; and the regulatory requirements and oversight of the NRC. In addition, it is important to recognize that risk estimates of nuclear power plants have decreased over the years. There are several reasons for these decreases: • Utilities have completed plant modifications intended to remedy concerns raised in earlier PRAs. 19	Section 2.4.2 (1 Scenario Summary): In the last sentence, the reference to loss of room cooling should be deleted since battery depletion alone is sufficient to lead to loss of the HPCI and RCIC systems. In addition, room coolers are not required to support HPCI/RCIC operability. All design bases scenarios assume loss of room coolers.

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initiators were grouped together. For the externally initiated events, the timeline of operator actions was developed assuming the initiator was a seismic event because the seismic initiator was judged to be the most severe initiator in terms of timing and with respect to how much equipment would be available to mitigate. Thus, there is some conservatism in attributing all of the event likelihood to a seismic initiator.

3.1.1 Sequence Groups Initiated by External Events

The PRA screening identified the following sequence groups that were initiated by external events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events:

- Peach Bottom long-term station blackout – $1x10^{-6}$ to $5x10^{-6}$/reactor-year
- Surry long-term station blackout – $1x10^{-5}$ to $2x10^{-5}$/reactor-year
- Surry short-term station blackout – $1x10^{-6}$ to $2x10^{-6}$/reactor-year
- Surry short-term station blackout with thermally induced steam generator tube rupture – $1x10^{-7}$ to $8x10^{-7}$/reactor year

These sequence groups were initiated by a seismic, fire, or flooding event. The mitigation measures assessment for each of these sequence groups was performed assuming the initiator was a seismic event, because it was judged to be limiting in terms of how much equipment would be available to mitigate. Fewer mitigation measures are expected to be available for a seismic event than for an internal fire or flooding event. For these sequence groups, the seismic PRAs provided information on the initial availability of installed systems. Based on the estimated level of plant damage, the availability of 10 CFR 50.54(hh) mitigation measures, their implementation time, and the timing and effectiveness of the emergency response organization support (e.g., in the Technical Support Center and Emergency Operating Facility) was evaluated.

It is important to note that, although it is not included in the above list, the seismically induced Peach Bottom short-term station blackout was also retained for analysis. With a frequency of $1x10^{-7}$ to $5x10^{-7}$/reactor year this scenario does not explicitly meet the SOARCA screening criterion, it was retained in order to assess the risk importance of a lower frequency, higher consequence scenario.

Seismic events considered in SOARCA result in loss of offsite and onsite AC power, and, for the more severe seismic events, loss of DC power. Under these conditions, the turbine-driven systems RCIC and TD-AFW are important mitigation measures. BWR SAMGs include starting RCIC without electricity to cope with station blackout conditions. This is known as RCIC black start. 10 CFR 50.54(hh) mitigation measures have taken this a step further and also include long-term operation of RCIC without electricity (RCIC black run), using a portable generator to supply indications such as reactor pressure vessel level indication to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine. Similar procedures have been developed for PWRs for TD-AFW. For the Peach Bottom and Surry long-term station blackout sequence groups, RCIC and TD-AFW can be used to cool the core until battery exhaustion. After battery exhaustion, black run of RCIC and TD-AFW can be used

COMMENTS: The approved BWROG EPG/SAC (Rev 2) for SAMGs does not include starting RCIC without electricity (RCIC blackstart).

PAGE 24	COMMENTS
PREDECISIONAL Revision 1 - 2/14/2010 1:20:00 AM Time estimates to implement individual mitigation measures were provided by licensee staff for each sequence group based on scenario descriptions provided by the NRC. The time estimates take into account the plant conditions following the seismic event. Also, for portable equipment at Surry, the time estimates reflect exercises run by licensee staff that provided actual times to move the equipment into place. The time estimates for manning the Technical Support Centers and the Emergency Operating Facilities also were provided by licensee staff and reflect the possible effect of the seismic event on roads and bridges. The mitigation measures assessment noted the possibility of bringing in equipment from offsite (e.g., fire trucks, pumps and power supplies from sister plants or from contractors, external spray systems), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented. Additional information on equipment available offsite and time estimates for transporting this equipment is available in Section 3.2. Evaluating the effectiveness of external water spray using conventional firefighting equipment to scrub an ongoing fission product release was not evaluated in SOARCA. This evaluation is being performed in a separate study. No multi-unit accident sequences were selected for the SOARCA project. Therefore, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor, even though Peach Bottom and Surry are two-unit sites. Also, at the time that the MELCOR models were developed for SOARCA, Surry Unit 1 had an opening in the reactor cavity wall and Surry Unit 2 did not. The MELCOR model for the Surry reactor includes an opening in the reactor cavity wall.	Recommend the SOARCA project document that the analysis and mitigative measures were based on Operator resources for a single unit. Make sure operators from the unaffected unit are not available to support mitigative measures for the affected unit.
3.1.2 Sequence Groups Initiated by Internal Events The PRA screening identified the following sequence groups that were initiated by internal events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events: • Surry interfacing systems LOCA – $7x10^{-7}$/reactor-year (licensee PRA), $3x10^{-8}$/reactor-year (SPAR) • Surry spontaneous steam generator tube rupture – $5x10^{-7}$/reactor-year These sequence groups result in core damage as a result of assumed operator errors. For the interfacing systems LOCA, the operators fail to refill the RWST or cross-connect to the unaffected unit's RWST. For the spontaneous SGTR, the operators fail to 1) isolate the faulted SG, 2) depressurize and cooldown the RCS, and 3) refill the RWST or cross-connect to the unaffected unit's RWST. The SPAR model and the licensee's PRA concluded that these two events proceed to core damage as a result of the above postulated operator errors. However, these PRA models do not appear to have credited the significant time available for the operators to correctly respond to events. They also do not appear to credit technical assistance from the TSC and the EOF. For the ISLOCA, the realistic analysis of thermal hydraulics presented in Volume IV subsequently estimated 3 hours until the RWST is empty and 10 hours until fission product release begins, 24	

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ORIGEN output files are assigned to the specified input power. Second, for three different cycles of fuel, BLEND3 interpolates a radio-isotopic inventory from the relevant ORIGEN output files. Finally, using the input volume fractions for the three different cycles of fuel, it creates a new, volumetrically averaged ORIGEN output file for the node for the specified input conditions.

The PRISM module from SCALE 5.1 was then used to drive ORIGEN decay calculations using the newly created averaged ORIGEN output files as input. PRISM is a SCALE utility module which allows the user to automate the execution of a series of SCALE calculations.

4.4.4.3 Peach Bottom Model

The Peach Bottom model is based on the Global Nuclear Fuel (GNF) 10x10 (GE-14C) fuel assembly. The GNF 10x10 is representative of a limiting fuel type actually being used in commercial BWRs. The GEH 10x10 model is illustrated in Figure 12. The model is very detailed for this application. The only significant assumption was that the part length rod portion of the reactor was modeled as a full assembly.

Twenty-seven different TRITON runs were performed to model three different cycles of fuel at nine different specific power histories. The specific power histories ranged from 2 MW/MTU to 45 MW/MTU to cover all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MW/MTU was used. For example, for second cycle fuel, the fuel was burned for its first cycle using 25.5 MW/MTU, allowed to decay for an assumed 30 day refueling outage and then 9 different TRITON calculations were performed with specific powers ranging from 2 to 45 MW/MTU. The BLEND3 code was then applied to each of the 50 nodes in the MELCOR model using the average specific powers and volume fractions. Once new libraries for each of the 50 nodes in the model were generated, the final step in the procedure was to deplete each node for 48 hours. The decay heats, masses, and specific activities as a function of time were processed and applied as input data to MELCOR to define decay heat and the radionuclide inventory.

Comment: Are the units specified here really MWD/MTU? It is confusing between power histories and decay heat.

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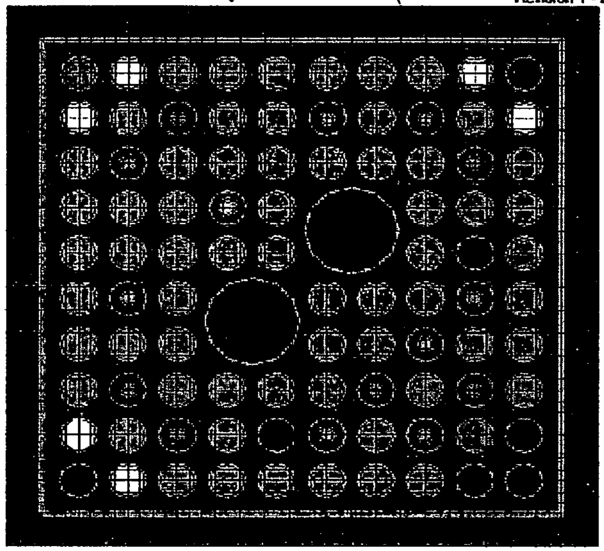

Figure 12 Schematic of Modeling Detail for BWR GNF 10x10 Assembly.

4.4.4.4 Surry Model

Previously, detailed input was developed for Surry in a separate NRC program on the source term from high-burnup uranium (HBU) fuel at the end of the fuel cycle. It used the same methodology as Peach Bottom (Section 4.4.4.3). The actual mid-cycle decay power is lower. However, the SOARCA schedule did not allow for a current operation, decay heat evaluation as was done for Peach Bottom.

4.4.4.5 Evaluation of the Results

There are very few measurements of decay heat in existence and those that do exist are not directly relevant to this study. Therefore, the discussion of the decay heat predictions will be limited to a comparison to previously published work. The best known source of decay heat predictions is summarized in Regulatory Guide 3.54 and results from the guide will be used to assess the predictions in the current study [37]. Decay heat for two decay times will be used as a check on the consistency of the results presented in this study. By interpolation of tables in RG 3.54 for a specific power of 27 MW/MTU, decay powers at 1 and 2 years following shutdown of 9.3 W/kgU and 5.1 W/kgU, respectively, are calculated. Using the results from the Peach Bottom calculations, the corresponding decay powers are 8.92 W/kgU and 4.734 W/kgU. The maximum difference between results is approximately 8 percent which is considered acceptable given the best estimate nature of the SOARCA study compared to the methods used to generate the tables in RG 3.54.

MWD/MTU?

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NRC staff performed quality assurance evaluations of all meteorological data presented using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" [42]. Further review was performed using computer spreadsheets. NRC staff ensured there was joint data recovery rate in the 90th percentile, which is in accordance with Regulatory Guide 1.23 [43] for the wind speed, wind direction, and atmospheric stability parameters. Additionally, atmospheric stability was evaluated to determine if the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). The mixing height data were retrieved from the EPA SCRAM database[5] (using years 1984-1992). Data needed for MACCS2 includes 10-meter wind speed, 10-meter wind direction in 64 compass directions, stability class (via Pasquill-Gifford scale and using representative values of 1-6 for stability classes A-F/G), hourly precipitation, and diurnal (morning and afternoon) seasonal mixing heights.

5.2.1 Summary of Weather Data

A summary of the meteorological statistical data is presented in Table 12, which shows that the predominant ground-level wind directions were generally *blowing to* the same direction during each annual period for each nuclear site. It also shows that the annual average wind speeds were generally low, ranging from 2.02 to 2.63 m/s at ground-level. The atmospheric stability frequencies were found to be consistent with expected meteorological conditions. The neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day. The wind direction and atmospheric stability (unstable, neutral, and stable) data are shown in Figure 13 through Figure 14 for the years that were actually used in the consequence analyses, which were 2005 for Peach Bottom and 2004 for Surry.

Table 12 Statistical Summary of Raw Meteorological Data for SOARCA Nuclear Sites

Parameter		Peach Bottom		Surry	
		Year 2005	Year 2006	Year 2001	Year 2004[1]
Avg. Wind Speed (m/s)		2.25	2.63	2.02	2.28
Yearly Precipitation (hr)		588 (6.7%)	593 (6.8%)	388 (4.4%)	521 (5.9%)
Atmospheric Stability (%)	Unstable	21.43	20.56	7.09	3.94
	Neutral	63.97	62.34	69.67	77.59
	Stable	14.60	17.10	23.24	18.47
Joint Data Recovery (%)		97.53	99.25	99.58	99.24

[1] Year 2004, as used in the Surry meteorological analysis, is a leap year (8784 total hourly data points versus 8760 hourly data points for a regular annual period).

[5] EPA SCRAM website: http://www.epa.gov/scram001/mixingheightdata.htm

COMMENTS:

Table 12 does not have any wind direction data.

Figure 13 was not provided for comment.

The average annual wind speed, that was calculated to be 2.02 to 2.63 m/s at ground level, does not match the 2-year data shot sent to NRC. Our data indicates the range is from 2.17 to 3.05 m/s.

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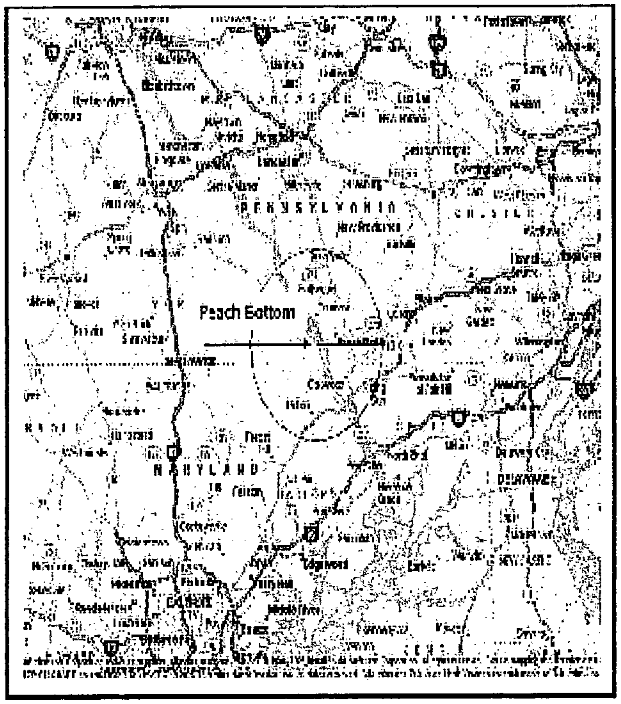

Figure 1 Site Location

Within a 1 mile radius of the plant, and on both sides of Conowingo Pond, steep sloping hills rise to about 300 ft above plant grade, with outcroppings of rock apparent at many locations. Because of the relatively rough terrain, much of this area is desolate, and wooded areas scattered throughout, although the more gentle sloping areas are cleared and cultivated. The site is located in a well-defined river valley, which in turn lies in rolling but not exceptionally rugged country. Maximum elevations in the immediate vicinity of the facility seldom exceed 300 ft above river level, although there are several plateaus and hilltops reaching 500 to 800 ft above the river within 10 mi to the southwest, west, northwest, and north of the site [1] (see Figure 2).

Figure 2 (site photo) does not show what it says it shows.

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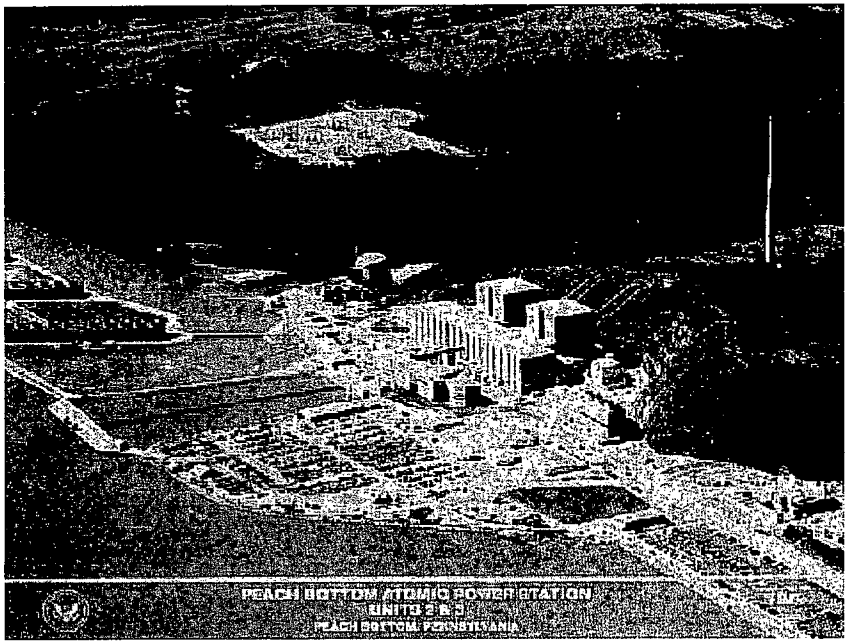

Figure 2 Site Photograph.

Need a more up-to-date photo. This picture is circa mid-1980's.

1.2 Outline of Report

Section 2 of this report briefly summarizes the method used to select the specific accident scenarios subjected to detailed computational analysis. Additional details of this method can be found in Volume 1 of this series of reports. Section 3 then describes the results of the accident scenario selection process when it was applied to Peach Bottom. Section 4 describes the key features of the MELCOR model of the Peach Bottom Atomic Power Station. Section 5 describes the results of MELCOR calculations of severe accident progression and radionuclide release to the environment for each accident scenario. Section 6 describes the way in which plant-specific emergency response actions were represented in the calculations of offsite consequences, and Section 7 describes the calculations of offsite consequences for each accident scenario. References cited in this report are listed in Section 8.

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3.2.3 Mitigative Actions	
No mitigative actions beyond those described in Section 3.2.4 were credited in this scenario.	
3.2.4 Scenario Boundary Conditions	
Two variations of the short-term station blackout scenario were considered. The first case assumes manual actions to manually actuate (*black-start*) the steam-driven RCIC system are either not successful. This action involves local, manual opening of normally closed valves to admit steam from the main steam lines into the RCIC turbine and pump discharge valves to direct water into the reactor vessel. Black-start of RCIC was assumed to occur at 10 minutes, thereby preventing the reactor water level from decreasing below the top of active fuel. While it is possible to start RCIC at a later time and still avoid core damage the latest possible start time was not examined. Results of the variation without RCIC black-start are described in Section 6.3.3.	"Either" should be deleted from the sentence.
In the second variation, operators successfully black-start the RCIC system and establish coolant flow to the reactor vessel. However, manual actions necessary to regulate steam flow into the RCIC turbine are not credited in this scenario because electric power to instrumentation needed to monitor reactor coolant level would not be available. As a result, the system effectively operates at a constant flow rate equivalent to the rated capacity of the system. This flow rate is greater than the rate required to make up for evaporative losses, and after an initial decrease, reactor water level gradually rises above nominal and eventually overfills the reactor vessel[5]. In this context, overfill means that the reactor water level rises to the elevation of the main steam line nozzles, allowing water to spill into the steam lines and causing them to flood with water. The steam extraction line for the RCIC turbine connects to the main steam line at a low elevation [adjacent to the inboard main steam isolation valves (MSIVs).] Therefore, water spilling over into the main steam lines blocks or flows toward the RCIC turbine, causing the system to cease functioning. Results of the short-term scenario with RCIC black-start are described in Section 5.4.	
Section 3.2.4.1 lists the sequence of events to be prescribed for two the unmitigated short-term station blackout calculations.	
[5] If electric (control) power was available, the RCIC system would cycle on/off to maintain reactor level between a minimum and maximum setpoint. Without these control signals, or an independent means of monitoring reactor water level and manually controlling coolant flow rate (i.e., turbine speed), the system is assumed to run at full capacity after it is started.	
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Time estimates to implement individual mitigation measures were provided by licensee staff for each sequence group based on scenario descriptions provided by the NRC. The time estimates take into account the plant conditions following the seismic event. Also, for portable equipment at Surry, the time estimates reflect exercises run by licensee staff that provided actual times to move the equipment into place. The time estimates for manning the Technical Support Centers and the Emergency Operating Facilities also were provided by licensee staff and reflect the possible effect of the seismic event on roads and bridges.	
The mitigation measures assessment noted the possibility of bringing in equipment from offsite (e.g., fire trucks, pumps and power supplies from sister plants or from contractors, external spray systems), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented. Additional information on equipment available offsite and time estimates for transporting this equipment is available in Section 3.2.	
Evaluating the effectiveness of external water spray using conventional firefighting equipment to scrub an ongoing fission product release was not evaluated in SOARCA. This evaluation is being performed in a separate study.	
No multi-unit accident sequences were selected for the SOARCA project. Therefore, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor, even though Peach Bottom and Surry are two-unit sites. Also, at the time that the MELCOR models were developed for SOARCA, Surry Unit 1 had an opening in the reactor cavity wall and Surry Unit 2 did not. The MELCOR model for the Surry reactor includes an opening in the reactor cavity wall.	Recommend the SOARCA project document that the analysis and mitigative measures were based on Operator resources for a single unit. Make sure operators from the unaffected unit are not available to support mitigative measures for the affected unit.
3.1.2 Sequence Groups Initiated by Internal Events	
The PRA screening identified the following sequence groups that were initiated by internal events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events: • Surry interfacing systems LOCA – $7x10^{-7}$/reactor-year (licensee PRA), $3x10^{-8}$/reactor-year (SPAR) • Surry spontaneous steam generator tube rupture – $5x10^{-7}$/reactor-year	
These sequence groups result in core damage as a result of assumed operator errors. For the interfacing systems LOCA, the operators fail to refill the RWST or cross-connect to the unaffected unit's RWST. For the spontaneous SGTR, the operators fail to 1) isolate the faulted SG, 2) depressurize and cooldown the RCS, and 3) refill the RWST or cross-connect to the unaffected unit's RWST.	
The SPAR model and the licensee's PRA concluded that these two events proceed to core damage as a result of the above postulated operator errors. However, these PRA models do not appear to have credited the significant time available for the operators to correctly respond to events. They also do not appear to credit technical assistance from the TSC and the EOF. For the ISLOCA, the realistic analysis of thermal hydraulics presented in Volume IV subsequently estimated 3 hours until the RWST is empty and 10 hours until fission product release begins,	
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and used for a realistic evaluation. Twenty-seven different TRITON runs were performed to model three different cycles of fuel at nine specific power histories. The specific power histories ranged from 2 MW/MTU to 45 MW/MTU, which bounded all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MW/MTU was used. For example, for the second cycle fuel, the fuel was burned for its first cycle using 25.5 MW/MTU, allowed to decay for an assumed 30 day refueling outage, and then nine different TRITON calculations were performed with specific powers ranging from 2 to 45 MW/MTU. The BLEND3 code was applied to each of the fifty core nodes[8] in the MELCOR model using average specific powers derived from data for three consecutive operating cycles and appropriate nodal volume fractions. Once new libraries for each of the fifty nodes in the model were generated, the final step in the procedure was to deplete each node for 48 hours. The decay heats, masses, and specific activities as a function of time were processed and applied as input data to MELCOR to define decay heat and the radionuclide inventory.

MWD/MTU?

4.7 Modeling Uncertainties

The primary objective of the SOARCA project is to provide a best-estimate prediction of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective, the SOARCA project utilizes integrated modeling of the accident progression and offsite consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective body of knowledge on severe accident behavior generated over the past 25 years of research.

The MELCOR 1.8.6 computer code [7] embodies much of this knowledge and was used for the accident and source-term analysis. MELCOR includes capabilities to model the two-phase thermal-hydraulics, core degradation, fission product release, transport, deposition, and the containment response. The SOARCA analyses include operator actions and equipment performance issues as prescribed by the sequence definition and mitigative actions. The MELCOR models are constructed using plant data, and the operator actions were developed based on discussions with operators during site visits. The code models and user-specified modeling practices represent the current best practices.

Uncertainties remain in our understanding of the phenomena that govern severe accident progression and radionuclide transport. Consistent with the best-estimate approach in SOARCA, all phenomena were modeled using best-estimate characterization of uncertain phenomena and events. Important severe accident phenomena and the proposed approach to modeling them in the SOARCA calculations were presented to an external expert panel during a public meeting sponsored by the NRC on August 21 and 22, 2006 in Albuquerque, New Mexico. A summary of this approach is described in Section 4.7.1. These phenomena are singled out because they are important contributors to calculated results and have uncertainty.

The two other topics, steam explosions, and drywell liner melt-through on a *wet* drywell floor have been previously included in lists of highly uncertain phenomena. Section 4.7.1 briefly

[8] Five radial rings by ten axial levels

Summary of NRC staff comments implemented into Revision 1 of SOARCA project documentation

Items listed in the order in which they were incorporated.

Item	Reviewer	Description
1	Schaperow	Schaperow_Summary_PeachBottom.pdf Comments on Summary document and Peach Bottom document as well as Peach Bottom peer review comment resolution list. Comments primarily on PB Section 5.6.1 required additional input from M. Leonard (see JS_PBComments4Mark.pdf). The resulting changes agreed to by Schaperow and Leonard were incorporated seperately and are described below.
2	Leonard	Leonard_NewPB_Ch3.doc Revisions to scenario descriptions in Chapter 3 of the Peach Bottom document. These changes were reviewed and acepted by C. Tinkler.
3	Leonard	Leonard_NewPB_SS5.6.1.doc Revisions to the description of environmental releases of iodine and cesium resulting from changes to the tech base leakage rate for the Peach Bottom site. These changes are in response to some of the comments made by J. Schaperow in item 1.
4	Leonard	Leonard_PB_Changes.txt These changes incorporate the remainder of the comments made by J. Schaperow to the Peach Bottom document listed in item 1 including the discussion of valve sticking.
5	Schaperow	Schaperow_PB_Text.txt Revision to the introduction to section 5.6 of the Peach Bottom document to acknowledge the role of the external peer review panel discussions in determing what sensitivity cases should be run.
6	Leonard	Phone conversation Minor changes to Chapter 3 were included at the request of C. Tinkler (relayed by Leonard). These changes included clarification of a scenario description as well as the rational used for determining the RCIC start time.
7	Sullivan	Sullivan_PeachBottomCh6.doc Gramatical and editorial changes to cohort movement descriptions. Corrected citation numbering and cross referencing.
8	Tinkler	Tinkler_PBComments.docx Additional refinements to chapter 3 and section 5.6 of the Peach Bottom document.
9	Dube	Dan_Dube_Surry.pdf Detailed editorial corrections and grammatical changes mostly to chapters 2 and 3. Inclusion of radionuclide inventory table and scrubbing of citation cross references will be deferred until other reviewer commentsare included.

10	Schaperow	Schaperow_Surry_Ch4.pdf Editorial and technical comments on Chapter 4 of the Surry document. Several comments required additional input from KC Wagner and were incorporated seperately as described below.
11	Sullivan	Sullivan_SurryCh6.doc Minor editorial comments and suggestions to Chapter 6 of the Surry document. These comments were largely overcome be events or already addressed. The comment regarding the definition of Cohort 6 was not included snce this definition would be the only place where dose to the non-evacuating cohort was characterized as voluntary. No formal definition of a voluntary dose has been provided. The SECPOP value was derived from total U.S. population and therefore the identical value is applied to both Surry and Peach Bottom
12	Wagner	Wagner_SurryCorrections.docx Changes to address the comments identified by Schaperow in item 10.
13	Schaperow	Shaperow_PB_Ch6pp1-5.pdf Comments on the first five pages of chapter 6 in the Peach Bottom document.
14	Schaperow	Schaperow_SummaryCommentList.pdf Enhancements to the resolution of peer review comments on the summary document.
15	Sullivan	Sullivan_Summary.doc Jones4Sullivan_Summary.docx Significant modifications to emergency response modeling sections of the Summary document (primarly chpter 5 but also more limited changes in the Executive summary, Ch 1 and Ch 3. J. Jones mapped the original comments onto the current version of the Summary document.
16	Schaperow	Scahperow_SurryCommentList.pdf Revisions to the peer review comment resolution list for the Surry site.
17	Burns	Verified citation cross references and listing for the Peach Bottom and Surry documents. Revised as necessary.

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Summary of NRC staff comments requiring additional time or consideration to implement
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1) Schaperow should draft the discussion of the advantages and disadvantages of conducting site specific analysis that he recommended in his review of item 77 in the Summary document comment list.

2) Jocelyn Mitchell's recommendations on are extensive and require technical input from other team members that is difficult to obtain on the required time scale. There may also be a number of philisophical issues that may need further vetting within the NRC staff.

3) Verify that Tinkler's description of safety valve leakage to obtain high pressure differential-low SG water level conditions is included in the final documentation.

4) Stutzky's recommendations for the Summary document relative to references to the NRC's safety objectives, the use of the term "absolute risk" in the summary results tables, and the potential for including population doses in the SOARCA results.

5) Characterization of the mitigated scenerio results as the "best estimate, base case" scenarios in the executive summary as requested by Schaperow.

6) Schapero's comments on pages 102-109 of Chapter 6 of the Peach Bottom report. These revisions were received at 1:55 MST on Friday, February 12 due to weather related shutdown at the NRC. Many of these comments required input from other team members which was not possible to obtain prior to the Monday, February 15 release to the peer review committee. Changes to this chapter from Sullivan have already been incorporated.

7) Nosek's comments constitute a substantial rewrite of the Executive Summary. There are also a number of open ended and more philisophical comments that are included which are difficult to address in this time frame.

[redacted]

State-of-the-Art Reactor Consequence Analysis (SOARCA) Program
MELCOR Best Modeling Practices Peer Review Comments

Revision	Date	Description
0	1-Jul-09	Review version issued to peer review panel for July 28-29, 2009 kick-off meeting
1	15-Feb-10	Review version incorporating peer review panel comments from first two review meetings

#	Rev.	Reviewer	Comment	Resolution
1	0	Clement	It is not said in § 3.1.1.2 whether ex-vessel steam explosion is considered or not.	There are no deep pools of subcooled water in the Surry simulation and it is not clear that a coherent poor occurs. So an ex-vessel steam explosion is not considered to be credible.
2	0	Clement	Introducing cesium molybdate allows to better reproduce Phebus results. In reality, it is more likely that Mo is released from fuel as an oxide and then reacts with cesium to be transported as molybdate.	This point is conceded with the observation that this reaction is happening too close to the release point for the difference to impact the environmental release or ultimate consequences.
3	0	Clement	Does the fact that RN class 4 is completely transferred to class 16 mean that no iodine is transported as I2 (this is contrary to NUREG 1465)?	The state of knowledge regarding iodine releases has been evolving rapidly since the start of the SOARCA project. For reference NUREG-1465 suggests a value of 5% of the iodine release as gaseous iodine. The Phebus results suggest $1/10^{th}$ of that release with the highest concentration occurring in the presence of boron carbide control rods. In any event, NUREG-1465 is a licensing document and is not necessarily relevant for a best estimate calculation. Whatever arguments are made will need to be sequence specific given differences in the timing and nature of the release.
4	0	Clement	Concerning CCI modelling, is there a criterion to say if and when concrete basement will fail?	The sequences considered in the SOARCA project involve earlier and more severe releases than those that would result from basement failure, e.g., liner melt through. As a result, basement failure was not explicitly considered.
5	0	Clement	For ex-vessel phenomena, is H2 burn triggered by melt ejection considered?	In the current simulations the default HECTR burn criterion is being used. Auto-ignition often occurs at 1300-1500 K. Sensitivity studies relating to hydrogen combustion for were conducted for the Surry analysis. The results of these studies are documented in the Surry analysis documentation for the SOARCA project.
6	0	Clement	There is a specific treatment for zircaloy oxidation	Air ingress into the reactor vessel only occurs in

#	Rev.	Reviewer	Comment	Resolution
			in presence of air. Is there one for ruthenium release?	late stages when the vessel and HL have failed thus creating a chimney. At this point less than 5% of the fuel is left in the vessel. The flow into the vessel will also be heavily steam dominated at this time.
7	0	Clement	There is no description of iodine chemistry modelling in the containment. In particular, was there an attempt to compensate somehow the lack of models for gas phase chemistry?	Since the area of iodine chemistry modeling is evolving rapidly it is difficult to determine what constitutes best estimate . The issue of iodine speciation will be considered as aprt of the uncertainty quantification effort.
8	0	Vierow	Introduction, page 1, second paragraph: How are values for MELCOR sensitivity coefficients verified as being "realistic"? Are there calculation notebooks documenting all of these?	Justifications for some of the input parameters are discussed in detail within the SOARCA documentation. The issue of best modeling practices was also discussed in some detail during the first peer review effort at the start of the SOARCA project. A living document of best practice input values is also maintained by the Sandia MELCOR analysis team. Those parameters assessed as being the most sensitive and uncertain will be subject to further analysis in the uncertainty and sensitivity quantification effort.
9	0	Vierow	Section 2.1, Page 3, 1st paragraph and Volume 1: The version of MELCOR used for these calculations is noted at 1.8.6. Page 40 of Volume 1 can imply that MELCOR 2.0 was used. Will the calculations be re-run with MELCOR 2.0? If not, will the input be modified to MELCOR 2.x format for future calculations, especially if additional plants are evaluated for the SOARCA?	MELCOR version 1.8.6 is the production version of MELCOR for all planned SOARCA calculations. Since the physical modeling capabilities of MELCOR 1.8.6 are equivalent to those of version 2.x, modifying the input for version 2.x format or rerunning the calculations in version 2.x is considered to be beyond the scope of the current project.
10	0	Vierow	Page 58, last paragraph: The thermally-induced SGTR's are assumed to occur prior to other RCS natural circulation failures. The model used to calculation tube wall failure is important when making this assumption. Previous MELCOR analyses by other researchers (for example, Liao and Vierow, *Nuclear Technology*, 2005) have	The two sequences in which SG tube rupture did and did not fail are largely a reflection of the uncertainty regarding this issue. See the resolution to items 76, 78, 81, 86, and 99 in volume IV. Steam generator tube failure without hot leg failure is not considered credible however.

#	Rev.	Reviewer	Comment	Resolution
			shown that the uncertainty in various models is large enough to prevent a clear determination of the first failure location.	
11	0	Leaver	Vol. II, page 70, last sentence of first paragraph, and a number of other places, use the term "physically unreasonable" to describe why early containment failure phenomena are no longer considered. This term does not connote the situation very well to me. I would suggest alternative wording, for example: "While the phenomena are conceivable, the conditions necessary for them to occur in an LWR severe accident environment are so remote that the phenomena are now considered essentially impossible in this environment."	The phrase "physically unreasonable" is chosen to be consistent with prior NRC severe accident research activities. This phrase has become a term of art to refer to an event that, practically speaking, the conditions necessary to produce the phenomena are so remote that the event is probabilistically uninteresting and need not be quantified.

Notes

State-of-the-Art Reactor Consequence Analysis (SOARCA) Program
Surry Integrated Analysis Peer Review Comments

Revision	Date	Description
0	1-Jul-09	Review version issued to peer review panel for July 28-29, 2009 kick-off meeting
1	15-Feb-10	Review version incorporating peer review panel comments from first two review meetings

#	Rev.	Reviewer	Comment	Resolution
1	0	Clement	In the different SBO sequences, there is most generally creep failure of the hot leg nozzle before vessel lower head creep failure. The subsequent RCS depressurization allows discharge of the accumulators that delays the progression of the accident. There is certainly a quite large uncertainty in the timing of creep failure both for hot leg nozzle and vessel lower head. To get an idea of the impact of this uncertainty, it would be interesting to know what would have been the time of lower head failure in the absence of hot leg failure. The scenario could be quite different. In IRSN PRA level 2, we use distributions to calculate induced breaks in RCS various locations for SBO sequences.	The parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.2.3 was added to report to examine the sensitivity of the timing of hot leg failure to the TI-SGTR. Addditional claulctions were performed with MELCOR and SCDAP/RELAP5 to examine the issue. The base case response is shown to be reasonable.
2	0	Clement	It is said that upon hot leg creep failure a large hole opened (i.e. like a large break LOCA). What is the basis for this statement?	Due to the softening of the piping at high temperatures, a large hole was expected to open in the HL.
3	0	Clement	For SBO with thermally induced SGTR, there is about 15 minutes between SGTR and hot leg nozzle failure. Before the latter event, there are two release paths, one to the environment through the failed SG and one to the containment through the pressurizer safety relief valve. After, there is an additional pathway to the containment through the large hole in the hot leg, so less direct release to the environment. As said before, there are probably large uncertainties in timing of failures that also depend on the state of the SG tubes. A full probabilistic treatment would be the best thing to do. If not possible, sensitivity studies would be helpful.	The parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.2.3 was added to report to examine the sensitivity of the timing of hot leg failure to the TI-SGTR. Additional calculations were performed with MELCOR and SCDAP/RELAP5 to examine the issue. The base case response is shown to be reasonable.
4	0	Clement	Using decontamination factors from ARTIST for the retention in the SG secondary side is probably the best thing to do in the absence of a validated	Release of gaseous iodine was not considered in the current MELCOR results. Using the noble gas release through the TI-SGTR and short-term

#	Rev.	Reviewer	Comment	Resolution
			model. It should however be kept in mind that these results are valid for aerosol particles and not for gaseous iodine that may escape from the RCS.	release rates of gaseous iodine from the Phebus program, the contribution of gaseous iodine to the source term was quantified. This analysis has been added as section 5.6.1.
5	0	Clement	Concerning spontaneous SGTR, a release mechanism exists even without core degradation. Part of the iodine dissolved in the RCS water (augmented by the iodine spike induced by the transient) can, upon flashing conditions at the break in the faulted SG, partition to the gas phase and be released. Droplets containing dissolved iodine can also be entrained with a significant retention in the SG (see ARTIST).	The additional source term from iodine spiking was quantified. While an important operational or DBA concern, it is trivial compared to the other iodine source terms and not expected to impact offsite consequences. This analysis has been added as section 5.6.2.
6	0	Clement	Unmitigated STSBO with TI-SGTR: The hot leg failure occurs 15 minutes after SGTR, therefore most FP's go into containment. An uncertainty study can be done on preventing hot leg failure and waiting for a pressure vessel failure. (Some reviewers agree, however SNL noted that the analysis does not approach a high pressure vessel failure.)	The parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.2.3 was added to the report to examine the sensitivity of the timing of hot leg failure to the TI-SGTR. Additional calculations were performed with MELCOR and SCDAP/RELAP5 to examine the issue. The base case response is shown to be reasonable.
7	0	Mrowca	Unmitigated short term SBO: There is the concern that if these procedures are published in a NUREG, the licensees may want to take credit for them.	The SOARCA study represents a best estimate calculation of a limited set of events that dominate the core damage frequency space. As such it is unlikely that the SOARCA results or modeling assumptions will unduly influence NRC regulations.
8	0	Mrowca	Mitigated short term SBO: the water supply needs to be confirmed. Procedures must exist for injecting water.	The mitigation procedures were explicitly confirmed by the NRC staff during a second visit to the Surry site. The 1M gallon injection volume is consistent with the presence of a nearby river source.
9	0	Gabor, Henry	Mitigated short term SBO: why are there H2 burns? Is there a criterion for ignition when there is no power? Is nodalization controlling? What would be the impact of delaying the burns due to	Section 5.1.3 was added to examine uncertainties in the time of combustion and the impact of hydrogen detonation.

#	Rev.	Reviewer	Comment	Resolution
			inadequate ignition?	
10	0	Stevenson	Hydrogen burn (deflagration) was discussed, but there was no discussion of hydrogen detonation. Has this been evaluated to be below the CDF defined? In this reviewer's experience, hydrogen detonation, depending on their size and location, can cause large leakage or breach of containment.	Section 5.1.3 was added to examine uncertainties in the time of combustion and the impact of hydrogen detonation.
11	0	Committee	Consider the state of the steam generator tubes in the Surry analysis.	The parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.2.3 was added to report to examine the sensitivity of the timing of hot leg failure to the TI-SGTR. Additional calculations were performed with MELCOR and SCDAP/RELAP5 to examine the issue. The base case response is shown to be reasonable. We were unable to get information on the current SG tube flaw distribution. However, Section 5.2.3 included consideration of high stress multipliers (i.e., 2 and 3), which relate to severe flaws.
12	0	Clement	The dose limit for radiation workers endorsed by the Health Physics Society that was 5 rem/yr is now 2 rem/yr. (cf. Bixler slide 7 from peer review kickoff meeting)	This reflects a position of the Health Physics society and does not necessarily reflect NRC regulations. In any event, this value has no direct impact on any of the dose truncation criteria used in SOARCA. The value was only mentioned for comparison purposes.
13	0	O'Kula	Ensure text is consistent with meteorological data provided. Discuss how a "representative year" is chosen from data that varies widely, or how a sensitivity study will be performed to confirm year in question is appropriate. For example, p. 58 of Vol. I shows different predominant wind direction for Peach Bottom (2005 and 2006) and large precipitation difference for Surry (2001 and 2004).	For Peach Bottom, the wind direction issue was resolved by plotting wind roses for the two years, 2005 and 2006. The wind roses were very similar even though the peak dominant wind direction for the two years is different by almost 180 degrees. The "Predominant Wind" data given in the table is correct but misleading and has been removed from the table.

#	Rev.	Reviewer	Comment	Resolution
				For Surry, the issue is the number of hours of precipitation. The data indicate that there are 34% more hours of precipitation in 2004 than in 2001. Even so, precipitation only occurs during 6% of the hours of 2004, so precipitation is not a factor the large majority of the time. The resulting difference in the predictions is not expected to be large.
14	0	O'Kula	Consider dose conversion factors for children and adolescents for those cohorts that are largely composed largely of those population groups, e.g. "schools".	This is beyond the scope of what we can do within the SOARCA project. DCF files for children and adolescents that can be used with MACCS2 would need to be created. MACCS2 currently only allows a single DCF file for a run, so separate runs would be needed for each of these groups. Finally, since risk of health effects is the primary metric being reported, we would need to have risk factors (factors that convert dose to likelihood of a health effect) for children and adolescents. To our knowledge these data do not exist In addition, PRA risk studies have not done this historically.
15	0	O'Kula	Three different references are cited for deposition velocity, are they one and the same? Ref. 48 in Vol. I, Fred Harper et al., NUREG/CR-6244, and USNRC/CEC expert elicitation	The CEC expert solicitation study is the source used to determine deposition velocities. This has been clarified in the text. Clarifying text on deposition velocities has also been added to section 5.4 of the SOARCA Methods document.
16	0	O'Kula	Please provide the draft report of the NRC's interpretation of CEC study, "Expert data report for deposition and relocation", or other bases for deposition velocity.	This report remains in draft form and is not yet available for distribution. A table providing specific deposition velocities drawn from this draft report and used in the SOARCA analyses has been included in Section 5.4 of the SOARCA methods document.
17	0	O'Kula	The report should indicate what is included and excluded in population dose. For example, food ingestion, decontamination workers, people returning to their homes. Explain from MACCS2 inputs/assumptions, and results, the key	This information is summarized in the introduction to the Off-Site Consequences chapter of the Integrated Surry analysis report.

#	Rev.	Reviewer	Comment	Resolution
			parameters affecting population dose.	
18	0	Mrowca	Discuss in the report the basis for SOARCA values and mention values used by others, esp. NUREG-1150, for hot spot relocation, normal relocation and habitability criterion.	The NUREG-1150 values for hotspot, relocation, and habitability were 0.5 Sv (50 rem), 0.25 Sv (25 rem), and 40 mSv (4 rem) over 5 years. Additional text was added to section 6.2.1.
19	0	Leaver, Gabor	The ISLOCA sequence does not need to be reported. The sequence is not possible because B.5.b equipment would be used. The best estimate is that this sequence won't happen. Gabor: May be true for PB and Surry, but B.5.b is not completely implemented in other plants.	Although the frequency for the ISLOCA is low, this event is unique in that it has a higher potential risk. The ISLOCA has also been of historical interest and is included in the licensee's PRA. For these reasons as well, this sequence has been included in the SOARCA study.
20	0	Clement	Mechanical resuspension needs to be addressed if turbulent deposition is to be taken into account.	Currently MELCOR does not have models for either turbulent deposition or resuspension. Side calculations are reported in Section 5.5.4 that show turbulent deposition is negligible. There was insufficient geometric information to estimate impaction. In summary, turbulent deposition, impaction, and resuspension were all neglected. Since the calculated retention from other mechanisms was small, the results are conservative (i.e., no impact if resuspension was included because nothing was deposited).
21	0	Leaver	ISLOCA: Once the flow is going, Reynolds numbers will be very large. Turbulent deposition is significant. DF's must be looked at.	Currently MELCOR does not have models for either turbulent deposition or resuspension. Side calculations are reported in Section 5.5.4 that show turbulent deposition is negligible. There was insufficient geometric information to estimate impaction. In summary, turbulent deposition, impaction, and resuspension were all neglected. Since the calculated retention from other mechanisms was small, the results are conservative (i.e., no impact if resuspension was

#	Rev.	Reviewer	Comment	Resolution
				included because nothing was deposited).
22	0	Leaver	ISLOCA: Do we want to show calculations out to 100 miles? Will this result in undue concern?	Results in older studies went out to much longer distances: 500 mi in the siting study and 1000 mi in NUREG-1150. SOARCA takes a departure from these earlier works by limiting consequence analysis results to shorter distances. The final determination by the NRC staff is to limit the consequence predictions to a 50 mile radius which is reflected in revision 1 and subsequent revisions of the documentation.
23	0	Leaver	It is a good idea to do a sensitivity study on later HL creep rupture, but note the point that induced SGTR will hasten the time of HL creep rupture so as to at least qualitatively make the case that significant delay in HL creep rupture after SGTR is very unlikely.	The parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.2.3 was added to the report to examine the sensitivity of the timing of hot leg failure to the TI-SGTR. Additional calculations were performed with MELCOR and SCDAP/RELAP5 to examine the issue. The base case response is shown to be reasonable.
24	0	Leaver	Why not include SG injection as a mitigation action for STSBO? Doing this will cut the induced SGTR contribution to I release (currently 0.5%) in half, and will be even more important if HL creep rupture is delayed	In general SG injection was judged not likely prior to TI-SGTR in a severe seismic event. It is acknowledged, though, that the diesel pump is available post core damage but alignment with sprays would be optimal. The alternative course of action would be to inject into SG which would increase DF in the SG and reduce sensitivity to HL creep rupture. But proximate HL failure would reduce the impact of this measure.
25	0	Leaver	Turbulent deposition should be considered for the ISLOCA. ... this is a typical long pipe problem with a large length to diameter ratio, which tends to produce high decontamination factor for aerosols. (see detailed post kick-off comments from Leaver)	Currently MELCOR does not have models for either turbulent deposition or resuspension. Side calculations are reported in Section 5.5.4 that show turbulent deposition is negligible. There was insufficient geometric information to estimate impaction. In summary, turbulent deposition, impaction, and resuspension were all neglected.

#	Rev.	Reviewer	Comment	Resolution
				Since the calculated retention from other mechanisms was small, the results are conservative (i.e., no impact if resuspension was included because nothing was deposited).
26	0	Leaver	The non-fission product to fission product (inert) aerosol mass ratios used for SOARCA modeling seem low based on our work, particularly for BWRs. For PWR-type fuel bundles measurements from the SFD 1-4 experiment indicate inert aerosol mass (Cs, Sn, Cd, Ag, U, others) in the range of 1 to 3 x the fission product aerosol mass. There is also information available from Phebus FP tests which suggests even larger ratios. BWR cores of the same power level as a PWR core have 2 to 4 x the mass of materials that form inert aerosols in a severe accident, and only about 25% more fission product mass. We typically use 1:1 for PWRs and 2:1 for BWRs in our design basis calculations.	PWRs have a Ag-In-Cd release model. Mass associated with inerts in compound form is included, (e.g., in CsOH, the OH is inert). There is 2005 kg of control material, much of which was vaporized and became aerosols.
27	0	Leaver	In Figure 20, the containment airborne aerosol reduction at the time of HL creep rupture is very fast. It looks like reduction of a factor of 3 in minutes. We have not seen deposition rates from natural processes (sedimentation, diffusiophoresis, and thermophoresis) this high.	Addressed in Section 5.6.4 of revision 1.
28	0	Leaver	The matter of potential radiation exposure to the operator for each of the mitigation actions should be addressed.	With the exception of the Surry STSBO and TISGTR, the mitigation actions prevent core damage, so there would be no radiation exposure. For the Surry STSBO, the containment is intact, so the radiation exposures is expected to be within DBA limits (<5 rem). For the Surry TISGTR, the release to the environment is also naturally mitigated by deposition in the steam generator and subsequent rupture of the hot leg regardless of any operator mitigation actions.

#	Rev.	Reviewer	Comment	Resolution
				A credible analysis of operator exposure would require a detailed human reliability evaluation of plant procedures and detailed scenario specific information. Such a study is currently beyond the scope of the SOARCA project.
29	0	Leaver	Vol IV, page 105, second paragraph, 6th line: Should it be "from the vessel"?	No the release is from the fuel but the release is relatively small.
30	0	Leaver	A basis should be provided for assuming safety systems and structures (including containment leak rate) function as designed after an earthquake which is 3 or 4 x the SSE. This is also an appropriate matter for a sensitivity study (i.e., increased containment leakage early).	The general topic of containment structural response to seismic events beyond SSE is an area for further NRC research and is beyond the scope of SOARCA. However, the sensitivity of calculated source terms to the possibility of enhanced containment leakage caused by a large seismic initiating event was examined for the BWR LTSBO scenario. Results of these calculations suggest release of important fission products is insensitive to increases in containment leakage up to 10 times the Tech Spec limit (the largest leak rate examined in the sensitivity analysis.)
31	0	Leaver	The notion of emergency response out to 20 miles was very prominent in Section 6 and as presented conveys the wrong idea. I suggest toning down the amount of information on 20 mile effort (other than consideration of shadow evacuation which is a realistic consideration of the 10 mile evacuation) and when it is discussed make clear that it is just a sensitivity study.	Agreed. The discussion on areas beyond the EPZ on page 176 were moved to the Sensitivity Study section in Section 6.4. Additionally, to better account for cohort movements, the cohorts have been redefined eliminating the 10 to 20 public as a cohort group. The text was updated accordingly.
32	0	Leaver	The references apparently are misnumbered. Also two different ways are used in referring to references (see for example the first paragraph on page 176 ("[10]" and the last paragraph on page 177 ("(NRC, 2005)").	The references have been changed as follows: NRC, 2005 on pages 177, 178 and 199 is [43]. NRC, 2007 on page 183 is [44]. TRB, 2000 on pages 198 and 206 is [45]. NRC, 2008 on page 199 is [46]. The additional references identified in this response will be added to the reference list.

#	Rev.	Reviewer	Comment	Resolution
33	0	Leaver	First paragraph on page 179: "WINMACCS allocates 0.061 percent…" should be 6.1 percent.	Agreed. However, the revised approach to cohorts eliminates this paragraph in both Volume III and Volume IV.
34	0	Leaver	Really hard to read or figure out Figure 130.	Agreed. Figure 130 was intended to help describe the user interface for the WinMACCS model; however, it is not necessary to use the picture. Figure 130 has been deleted.
35	0	Leaver	Hard for me to discern Table 18 though if I spent more time maybe I'd get it.	Table 18 and similar tables consist of WinMACCS parameters primarily of interest to the consequence modelers. Additional discussion has been added including: "The columns identify input parameters of interest to the MACCS2 and WinMACCS user and are provided to support detailed use of this study [26]. A brief description of the parameters is provided below. • Delay to Shelter (DLTSHL) represents a delay from the time of the start of the accident until cohorts shelter. • Delay to Evacuation (DLTEVA) represents the length of the sheltering period from the time a cohort enters the shelter until the point at which they begin to evacuate. The (ESPEED) was assigned for each of the three phases of the evacuation used in WinMACCS including Early, Middle, and Late. ESPEED Early is typically a faster speed for a very short duration until the point at which congestion overcomes the network. ESPEED Middle is the average evacuation speed, derived from the Surry 2000 ETE report, and reflects congested travel. Speed adjustment factors were utilized in the WinMACCS application to better account for free flow in rural areas and congested flow in urban areas. ESPEED Late begins at the point evacuees have exited the affected area where additional roadways are available and congestion

#	Rev.	Reviewer	Comment	Resolution
				dissipates.
36	0	Leaver	First full paragraph on page 185: "EAL SS1.1 specifies that if all offsite AC power is lost for greater than 15 minutes an SAE is declared" should be all onsite and offsite AC power. This phrase occurs in many other places.	Agreed. Verbiage has been corrected on page 185 to state loss of all offsite power and all onsite AC power.
37	0	Leaver	"Cohort 4: 10 to 20 Public" paragraph on page 186: "This was established at 3 hours after gap release." I think this should be at 6 hours after gap release.	Agreed. However, the cohorts have been revised and this cohort has been eliminated, therefore this text has been deleted.
38	0	Leaver	Similar comment as item [37] applies to Section 6.4.1.2 on page 187, i.e., gap release for unmitigated STSBO occurs at 3 hours, not 9 hours.	See Item 37.
39	0	O'Kula	Figures 145, 147, 149, 151, 153, and 154 show EARLY, CHRONC, and total results for the unmitigated STSBO sequence, unmitigated STSBO sequence with TISTGR sequence, mitigated STSBO sequence with TISTGR sequence, LTSBO sequence, unmitigated ISLOCA and SST1 source term, respectively. To properly review the offsite consequences of these sequences, tables of the key input parameter values for the EARLY and CHRONC modules are needed. We are interested in site-to-site differences as well as changes in assumptions/inputs from the NUREG-1150 era analysis to the SOARCA analysis.	A description of MACCS 2 input and best practices is under development separate from the SOARCA project. When completed this document will provide a companion piece to the MELCOR best practices document prepared within the SOARCA project. The MACCS2 best practices document is not yet ready for release however.
40	0	Vierow	The probability of a thermally induced SGTR was noted to be just above the screening criteria. The assumption of a stuck-open SG safety valve at 3 hours may reduce the sequence probability below the screening criteria. This is a good example of an event retained for completeness. Include Tinkler's explanation in the final documentation that other analyses consider safety valve leakage	Agreed, the SOARCA treatment of the SGTR event is slightly conservative.

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			to obtain the high pressure differential-low SG water level conditions.	
41	0	Gabor	Is a Decontamination Factor of 7 still valid late in time when flow rates are reduced?	Flow rates remain high (i.e., choked) until the primary system fails. At that point the releases are so small that the decontamination factor has a small impact on the environmental release.
42	0	Henry	The assumption of "no UO_2 present after vessel failure" needs to be justified. There may be some reactor designs in which not all of the debris exits the core region. Some Westinghouse designs have upflow and downflow (Vierow - in the downcomer?) which allows a fraction of the debris to remain. (Wagner said that they may need to consider Ru release. He noted that a ring of fuel may remain in the lower plenum.) (Wagner slide 19)	Section 5.6.3 was added to address the Surry plant design and the sequences analyzed. Other designs and sequences must be examined on a case by case basis.
43	0	O'Kula	Provide citation for data used to infer radionuclide pipe deposition rate. Verbal discussions during second peer review meeting referenced a draft NUREG with Dana Powers as the lead.	A memo describing the results of the ARTIST program was transmitted to the peer review panel. The draft NUREG has not been completed.
44	0	Stevenson	Detonation needs to be examined, not just deflagration. There is a factor of 3 difference in pressure (Wagner slide 26).	Section 5.1.3 was added to examine uncertainties in the time of combustion and the impact of hydrogen detonation.
45	0	Canavan	Canavan will provide data to Schaperow on spray patterns at low flow rates (less than 2/3 rated flow) for containment sprays. This data should be reflected in analysis (Wagner slide 26).	No data was available on this point, but it is important to note that heat removal from the containment is insensitive to spray pattern.
46	0	Leaver	Consider whether it is possible to have a single burn that could lead to detonation (Wagner slide 28)	Section 5.1.3 was added to examine uncertainties in the time of combustion and the impact of hydrogen detonation.
47	0	Gabor	LERF represents about 10% of the core damage frequency (CDF) by industry data for PWRs. This is inconsistent with SOARCA and will need to be explained.	While an examination of the implications of the SOARCA results relative to current PRA practice should be considered, undertaking such a study is beyond the scope of the SOARCA project. Any equivalency between CDF and release timing implies assumptions regarding accident

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				progression. The SOARCA project was undertaken in large part to reexamine traditional accident progression assumptions.
48	0	Stevenson, Leaver	The concern remains about increased leakage due to seismic events. The concern is particularly for PWRs. An expert is needed to help define the fragility of leakage. A possible reference is the SQUG (Seismic Quality Uncertainty ???) data on fragility.	Fragilities of key components are being examined by NRC staff but are not available for inclusion in the SOARCA documentation. In general, the importance of future research into seismically initiated events has been identified by the SOARCA project but is considered beyond the scope of the current SOARCA project. (See also item 30)
49	0	O'Kula	The MELMACCS treatment of source terms needs to be better explained. As discussed in the draft Vol. I and plant-specific Vols. III and IV, there is a wide gap in the discussion from once the source term is determined to the point where the evacuation, sheltering, and normal activities are modeled. There needs to be more discussion on how the MELMACCS model transitions the MELCOR output to forming WinMACCS input, the assumptions applied, etc.	More detail has been added to the methods document on some MELMACCS-related information including deposition rates. Also, a MACCS2 best practices document is being prepared by the NRC external to the SOARCA project but is not yet available.
50	0	Kowieski	The evacuation time of the Special Facilities is late and will not go over well with the public. (Bixler 1st pres. Slide 20)	The relevant text has been updated to clarify that these groups shelter earlier in the event and then evacuate the time specified.
51	0	Kowieski	Too much time is spent on the non-evacuating public.	Consequence results for the non-evacuating cohort will continue to be included in the overall consequence calculations but a short paragraph has been inserted to describe the fraction of the emergency phase risk within 10 miles of the plant that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.
52	0	Leaver	The evaluations can be done on the basis of 100% evacuation, therefore the early fatality risk is zero. (Bixler 1st pres. Slide 16)	Consequence results for the non-evacuating cohort will continue to be included in the overall consequence calculations but a short paragraph

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				has been inserted to describe the fraction of the emergency phase risk within 10 miles of the plant that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.
53	0	Leaver, Kowieski	There is a strong precedent for presenting only out to 50 miles of data. Consider not showing the 100-mile data. (Bixler 1st pres. Slide 18)	Results in older studies went out to much longer distances: 500 mi in the siting study and 1000 mi in NUREG-1150. SOARCA takes a dramatic departure from these earlier works by limiting consequence analysis results to much shorter distances. The final determination by the NRC staff is to limit the consequence predictions to a 50 mile radius which is reflected in revision 1 and subsequent revisions of the documentation.
54	0	Canavan	Make comparisons to voluntary or involuntary exposure to assist the public with understanding the doses. (Bixler 1st pres)	A short paragraph has been inserted to describe the fraction of the emergency phase risk within 10 miles of the plant that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees
55	0	Gabor	Eliminate the original results in the report and show only the latest cases with the new cohorts. (Bixler 1st. pres slide 20)	Agreed. It was never the intention to show results from both cohort designs.
56	0	Gabor	Is a loss of ac power a unique event? It may lead down a path that is different than for a non-blackout event. Blackout may not be conservative. Consider when EAL is triggered.	Other scenarios were eliminated by the SOARCA screening criteria. Nevertheless, the SBO remains one of the fastest scenarios in terms of reducing water inventory.
57	0	Leaver	The effect on risk of the declaration of EAL (Emergency Action Level) needs to be captured.	This comment has already been covered by the response to other comments (see items 56, 59) regarding the timing of the declaration of general emergency. This will be considered as part of the uncertainty analysis effort.
58	0	Leaver	Applying the LNT seems inconsistent with the habitability criterion.	The return criteria represents a best estimate of existing emergency response procedures and policies. The different dose response models are

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				provided to aid in the interpretation and comparison of the predicted off-site consequences.
59	0	Kowieski	One of the accident progression time lines suggests that after declaration of GE by the plant, sirens and EAS message could be activated within 45 minutes. Based on the actual field experience, it could take up to 60 minutes to complete the A/N sequence (Sirens/EAS message).	The timelines used in the analyses are very near the times experienced in exercises. To address any difference in timing, Sensitivity #3 was performed increasing the initial delay in the notification of the public by 30 minutes.
60	0	Kowieski	It appears that the existing documents do not address the notification of public in case of siren(s) failure. Should a siren fail, it may take additional 45 minutes to notify the affected public by Route Alerting procedures.	The siren operating rates were reviewed under the reactor operations program (ROP) and found to be 99.9% at Surry which would correspond to the loss of about 1 siren. Route alerting for this one area would not affect the total evacuation time of the public. Text has been added to Section 6.2.5 to reflect the performance of the sirens.
61	0	O'Kula	How would different values for the surface roughness length change the risk results at the mean (average) level? Could a short paragraph or limited sensitivity analysis be used to address whether this is important within the 10-mile EPZ, and within the 20-mile region?	The surface roughness length will be considered as part of the SOARCA uncertainty quantification effort.
62	0	Stevenson	While subsurface fault movement is not a credible event at the 10^{-6}/RY frequency level of the SOARCA project, it is not clear that liquefaction of cohesionless soil, including engineered backfill, or failure of buried piping will not impact containment integrity at this frequency level. The typical slope of seismic hazard curves suggest that peak ground accelerations of 1-2 g could persist for more than a minute at the 10-6/RY frequency level. Beyond ground acceleration, the potential for soil liquefaction has not been sufficiently evaluated to date.	While it is acknowledged that more work must be done in the area of seismic impacts on containment structures, the treatment of seismic impacts on reactor containments used in the SOARCA project remains state-of-the-art within the nuclear safety community. The effort to advance this state-of-the-art is justified but far beyond the scope of the SOARCA project.

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63	0	Stevenson	The potential for hydrogen deflagration within containment as a result of a LOCA appears to have been carefully studied particularly with respect to steam inerting which precludes hydrogen reaction with oxygen. However, there *does not appear to have been a distinction made* between hydrogen deflagration (burning) which may occur several times without steam inerting during the course of LOCA with hydrogen volume percentages below 10 percent and detonation (explosion) of hydrogen concentrations above 10%. Existing containment design can be expected to accommodate hydrogen deflagration without failure, but the potential for a hydrogen detonation with a resultant pressure load at or near the containment failure load should be evaluated explicitly.	The only scenario with conditions suitable for burns was the mitigated STSBO. Section 5.1.3 was added to examine uncertainties in the time of combustion and the impact of hydrogen detonation.
64	0	Canavan	Safety valves and pilot operated relief valves play a significant role in the accident sequences analyzed in SOARCA. Both the successful operation as well as the failure modes under beyond design basis conditions are clearly significant in the analysis. While the failure modes considered in the SOARCA analysis are, in the opinion of this reviewer likely, others with more expertise in the area of safety valves should be consulted. (cf. detailed comments submitted by Canavan 10/14/09 for examples)	We used the median value of the normal valve failure that was supplied from the plant PRA staff, which was in line with NUREG/CR-6928. In addition, the STSBO + TI-SGTR considered failure of the secondary safety valve well below its expected failure duty to conservatively examine containment bypass.
65	0	Leaver	Regarding the matter of the 0.5% who choose not to evacuate, it is suggested that results be reported for non-voluntary risk (i.e., 100% evacuation) and that the voluntary risk (for those who choose not to evacuate) be reported as part	Consequence results for the non-evacuating cohort will continue to be included in the overall consequence calculations, but a short paragraph has been inserted to describe the fraction of the emergency phase risk within 10 miles of the plant

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			of the sensitivity study.	that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.
66	0	Leaver	A summary of fragilities for key components (e.g., Surry low pressure injection and containment spray; PB torus integrity, RCIC) for the 0.3 to 1 pga earthquakes would be useful, or at least the basis for assuming that they can perform their function after the earthquake. Both Surry and Peach Bottom are members of the Seismic Qualification Users Group (SQUG) which was developed by industry for older plants and may have some useful data. Dr. Robert Kassawara (650 855 2775) is the EPRI Program Manager for SQUG. NRC is aware of the SQUG database, having considered it in conjunction with resolution of USI A-46. NRC's Goutam Bagchi was involved in this. The EPRI seismic margins report (NP 6041, Rev. 1 – a licensable document) may also be useful.	Fragilities of key components are being examined by NRC staff but are not available for inclusion in the SOARCA documentation. In general, the importance of future research into seismically initiated events has been identified by the SOARCA project but is considered beyond the scope of the current SOARCA project.
67	0	Leaver	The LCF consequence curves (such as Volume III, Figure 64 and Volume IV, Figure 145) might be more meaningful if the risk was presented for a given radius (or ring of some average radius) as opposed to plotting the risk to all residents inside a given radius.	The analysis team felt that the current format provided the easiest interpretation. The format has been changed from curve to bar chart format to further improve interpretation.
68	0	Gabor	H_2 burning sensitivity – a delay in hydrogen burn should be analyzed (at higher H_2 concentration)	An extensive sensitivity analysis of hydrogen combustion has been added to the Surry documentation in section 5.1.3.

Notes:

State-of-the-Art Reactor Consequence Analysis (SOARCA) Program
Peach Bottom Integrated Analysis Peer Review Comments

Revision	Date	Description
0	1-Jul-09	Review version issued to peer review panel for July 28-29, 2009 kick-off meeting
1	15-Feb-10	Review version incorporating peer review panel comments from first two review meetings

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1	0	Gabor	Penetration failures should be considered. Without RPV depressurization, instrument tube and CRD tube ejection may dominate and could occur early.	There are multiple mechanisms for RPV depressurization so there is high confidence that the RPV will be depressurized. Several sensitivity calculations were performed to examine the effects of uncertainty in criteria used to evaluate mechanisms of depressurization. In all cases the RPV was fully depressurized before significant quantities of molten debris entered the RPV lower plenum.
2	0	Henry	If $CsMoO_4$ is modeled, then methyl iodide is also needed. The document reads that $CsMoO_4$ is modeled because it was seen in Phebus. If this is true, then methyl-iodide should also be tracked.	Sensitivity analyses documented in the Surry integrated analysis report demonstrated that iodine vapor had a minor effect on the environmental release. Based on this result it was determined that additional analysis of the Peach Bottom plant was not necessary.
3	0	Mrowca	The assumption that the diesel generators "fail to start" is questionable. PRA uses "fail to run", therefore the analysis is conservative.	Agreed. It should be noted that the effects of delays on loss of power between the "fail to start" and "fail to run" cut sets may not be significant relative to the STSBO and LTSBO scenarios already considered.
4	0	Leaver	Battery life may be another item for a sensitivity study.	The STSBO, STSBO with RCIC black start, and LTSBO effectively represent battery life times of 0, 1.7 and 4 hours. We have another undocumented case with 6 hrs. A sensitivity study was also conducted for the loss of vital AC Bus E-12, which has been added to the documentation of the BWR calculations (App A, section 5.5.3).
5	0	Henry, Mrowca	Look at the SRV fully open and partially open in the Peach Bottom analysis of long term SBO, i.e. make sure that failure to a fully open state is not used as a significant benefit.	A sensitivity calculation was performed (for the LTSBO scenario) to examine the effects of valve seizure in a partially-open position. The effects of this uncertainty are very small in comparison to uncertainties in the criteria for valve failure (see Section 5.6 of Appendix A).
6	0	Gabor	SRV NOT sticking open should also be considered in sensitivity analysis with impact on	Several new sensitivity calculations were performed and results added to documentation of

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			potential for penetration ejection as vessel failure mode.	the PB MELCOR analysis (see Section 5.6 of App A). The sensitivities examined alternative assumptions regarding SRV failure as well as the possibility of main steam line creep rupture, if SRV cycling persists beyond the time calculated in the LTSBO baseline analysis.
7	0	Henry	Confirm whether separators and dryers remain supported in the Peach Bottom long term SBO.	Calculated temperatures of the separators and dryers in the unmitigated LTSBO remain below 1500K. Therefore material melting is not likely. However, portions of the core shroud and other structures that support the separators/dryers reach temperatures that cannot support the weight of the separators/dryers. It is reasonable to expect the separators/dryers would move from the original position to some other position within the RPV, but the structure temperatures are <u>not</u> sufficiently high to result in substantial material melting and incorporation of additional metal mass to debris in the RPV lower plenum.
8	0	Henry	Consider Te reaction with unoxidized zircaloy (and therefore Te reaction with Sn)	The treatment of Tellurium release in severe accident modeling has varied over the years. Based on chemical thermodynamics, Te is suspected to form the inter-metallic compound SnTe, binding with the alloying agent Sn found in many forms of Zircaloy cladding. Some modeling treatments have attempted to capture this effect by binding the released Te with remaining unoxidized metallic cladding as it is being thermally driven out of the fuel. These treatments would subsequently release the trapped Te as the Zr became fully oxidized. It might be argued that some Te might remain with unoxidized Zr that has become molten and begun to relocate. This relocated material will subsequently refreeze at a lower cooler location and be subject to a second heatup and oxidation phase as the oxidation front

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				migrates downward during melt progression. While the formation of the inter-metallic compound certainly occurs, it is believed that due to the *general spatial incoherency of core heatup,* oxidation and melt progression (i.e. all states of damage potentially co-exist at the same time throughout the core region during core damage) that the effects of such potential sequestering of Te cannot be detected in a practical way and will not significantly affect the overall core-wide Te release signature. For this reason, this proposed release phenomenon is not treated explicitly in MELCOR. Instead, the overall net release signature of Te in MELCOR is based on an overall calibration of Te release predicted by the Booth formula and adjusted to match the integral release signatures determined from the Phebus *experiments* (FPT-1).
9	0	Mrowca	For Loss of Class IV bus, the SPAR has a stuck open SRV, not battery failure. Boundary conditions for this analysis need to be checked.	Stuck-open SRV is not an initiator for this sequence. The initiator is "loss of Div I Vital ac bus E12."
10	0	O'Kula	Ensure text is consistent with meteorological data provided. Discuss how a "representative year" is chosen from data that varies widely, or how a sensitivity study will be performed to confirm year in question is appropriate. For example, p. 58 of Vol. I *shows different predominant* wind direction for Peach Bottom (2005 and 2006) and large precipitation difference for Surry (2001 and 2004).	For Peach Bottom, the wind direction issue was resolved by plotting wind roses for the two years, 2005 and 2006. The wind roses were very similar even though the peak dominant wind direction for the two years is different by almost 180 degrees. The "Predominant Wind" data given in the table cited is correct but misleading and has been removed from the table. For Surry, the issue is the number of hours of precipitation. The data indicate that there are 34% more hours of precipitation in 2004 than in 2001. Even so, precipitation only occurs during 5.9% of the hours of 2004 and 4.4% in 2001, so precipitation does not play a large role in the

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				predicted mean offsite consequences. The remaining weather metrics between the years examined are very similar.
11	0	O'Kula	Consider dose conversion factors for children and adolescents for those cohorts that are largely composed largely of those population groups, e.g. "schools".	This is beyond the scope of what can be accomplished within the SOARCA project. DCF files for children and adolescents that can be used with MACCS2 would need to be created. MACCS2 currently only allows a single DCF file for a run, so separate runs would be needed for each of these groups. Finally, since risk of health effects is the primary metric being reported, we would need to have risk factors (factors that convert dose to likelihood of a health effect) for children and adolescents. To our knowledge these data do not exist. In addition, PRA risk studies have not done this historically.
12	0	O'Kula	Three different references are cited for deposition velocity, are they one and the same? Ref. 48 in Vol. I, Fred Harper et al., NUREG/CR-6244, and USNRC/CEC expert elicitation	The CEC expert solicitation study is the source used to determine deposition velocities. This has been clarified in the text. Clarifying text on deposition velocities has also been added to section 5.4 of the SOARCA Methods document.
13	0	O'Kula	Please provide the draft report of the NRC's interpretation of CEC study, "Expert data report for deposition and relocation", or other bases for deposition velocity.	This report remains in draft form and is not yet available for distribution. A table providing specific deposition velocities drawn from this draft report and used in the SOARCA analyses has been included in Section 5.4 of the SOARCA methods document.
14	0	O'Kula	The report should indicate what is included and excluded in population dose. For example, food ingestion, decontamination workers, people returning to their homes. Explain from MACCS2 inputs/assumptions, and results, the key parameters affecting population dose.	This information is summarized in the introduction to the Off-Site Consequences chapter of the Integrated Peach Bottom analysis report.
15	0	Mrowca	Discuss in the report the basis for SOARCA values and mention values used by others, esp. NUREG-1150, for hot spot relocation, normal	The NUREG-1150 values for hotspot, relocation, and habitability were 0.5 Sv (50 rem), 0.25 Sv (25 rem), and 40 mSv (4 rem) over 5 years.

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			relocation and habitability criterion.	*Additional text was added to section* 6.2.1.
16	0	O'Kula	Show how health risk impacts can be reduced to various countermeasure criteria (long-term dose) for a given sequence. Possibly tie operating procedures and accident mitigation procedures with early phase risk metrics.	The intent of this comment is not entirely clear. Clarifying text has been added to section 6.2.1 regarding the hotspot and relocation values used in SOARCA relative to NUREG-1150. It is not clear whether "countermeasures" refers to reactor operators, emergency responders, or both. The second sentence seems to focus on the reactor operators and other plant personnel. A number of evacuation sensitivity calculations have been conducted and are included in the documentation which explore impacts on off-site consequences.
17	0	Gabor	For the SST1 sensitivity study, highlight qualitatively the differences between SOARCA and SST1 results and the general reasons for the differences.	In general, the differences can be characterized by a massive change in the source term coupled with modest changes to evacuation planning models..
18	0	Leaver	The timings listing in the slides [for evacuation planning vs. consequence analysis] should be consistent.	Agreed. The correct timing was presented on the Jones slide 24 (from peer review kick-off meeting) and reflects the timing that was used in the model runs.
19	0	Leaver	The non-fission product to fission product (inert) aerosol mass ratios used for SOARCA modeling seem low based on our work, particularly for BWRs. For PWR-type fuel bundles measurements from the SFD 1-4 experiment indicate inert aerosol mass (Cs, Sn, Cd, Ag, U, others) in the range of 1 to 3 x the fission product aerosol mass. *There is also information available from* Phebus FP tests which suggests even larger ratios. BWR cores of the same power level as a PWR core have 2 to 4 x the mass of materials that form inert aerosols in a severe accident, and only about 25% more fission product mass. We typically use 1:1 for PWRs and 2:1 for BWRs in our design basis calculations.	The MELCOR PB (BWR) model accounts for a release of inert Sn alloy from Zircaloy clad. The release rate from Zr clad is assumed to parallel the release rate for fission product (radioactive) Sn from fuel. Typically 600 to 700 kg of non-radioactive Sn are released. This represents approx. 70 to 80% of the total mass of Sn alloy in *the core and is roughly twice the total core* inventory of Cesium (the most massive of the volatile FPs) and nearly four times more than the radioactive portion of the Cesium inventory.
20	0	Leaver	The matter of potential radiation exposure to the	With the exception of the Surry STSBO and

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			operator for each of the mitigation actions should be addressed.	TISGTR, the mitigation actions prevent core damage, so there would be no radiation exposure. For the Surry STSBO, the containment is intact, so the radiation exposures is expected to be within DBA limits (<5 rem). For the Surry TISGTR, the release to the environment is also naturally mitigated by deposition in the steam generator and subsequent rupture of the hot leg regardless of any operator mitigation actions. A credible analysis of operator exposure would require a detailed human reliability evaluation of plant procedures and detailed scenario specific information. Such a study is currently beyond the scope of the SOARCA project.
21	0	Leaver	A basis should be provided for assuming safety systems and structures (including containment leak rate) function as designed after an earthquake which is 3 or 4 x the SSE. This is also an appropriate matter for a sensitivity study	The general topic of containment structural response to seismic events beyond SSE is an area for further NRC research and is beyond the scope of SOARCA. However, the sensitivity of calculated source terms to the possibility of enhanced containment leakage caused by a large seismic initiating event was examined for the BWR LTSBO scenario. Results of these calculations suggest release of important fission products is insensitive to increases in containment leakage up to 10 times the Tech Spec limit (the largest leak rate examined in the sensitivity analysis.)
22	0	O'Kula	Figures 63, 65, 67 and 69 show EARLY, CHRONC, and total results for the unmitigated LTSBO sequence, STSBO sequence with RCIC blackstart, unmitigated STSBO sequence, and SST1 source term, respectively. To properly review the offsite consequences of these sequences, tables of the key input parameter values for the EARLY and CHRONC modules are	A description of MACCS 2 input and best practices is under development separate from the SOARCA project. When completed this document will provide a companion piece to the MELCOR best practices document prepared within the SOARCA project. The MACCS2 best practices document is not yet ready for release however.

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			needed. We are interested in site-to-site differences as well as changes in assumptions/inputs from the NUREG-1150 era analysis to the SOARCA analysis.	
23	0	Mrowca	Provide SPAR models for Peach Bottom and Surry, if possible	The SPAR models are not available for public release. However, it is important to note that the SOARCA scenario selection process identified scenarios that have historically been important contributors to overall severe reactor accident risk.
24	0	Henry	Add implications of steel failure, both static and dynamic	Movement of steam separators/dryers due to loss of structural support could conceivably dislodge fission product aerosols deposited on their surfaces, but the details of structural relocation cannot be calculated with confidence. However, the effects of sudden structural movement on aerosol retention were examined by reviewing the measured resuspension efficiency of aerosols deposited on structures subjected to sudden mechanical forces. DOE Handbook 3010-04 (Section 5.3.3.2) describes the potential for aerosol resuspension from the surfaces of solids subjected to severe vibration or shock (impaction) stresses. The bounding (maximum) fractional release under these circumstances is 0.1% (i.e., fractional release of 0.001). This value is sufficiently low to be neglected.
25	0	Leaver	How do we know that the valves will function after sitting open and exposed to hot fluid?	Failure of an SRV to continue cycling is examined explicitly in the MELCOR calculations for all sequences. Heating of valve internal components due to high gas temperatures (after the onset of core damage) is assumed to lead to component expansion, closure of necessary clearances and eventual valve seizure. The precise time at which this would occur is uncertain. However, several sensitivity calculations were performed to examine

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				the effects of alternative assumptions regarding the criteria for valve failure. Results of these sensitivity calculations have been added to documentation of the PB MELCOR calculations.
26	0	Henry	An approach to quantify or bound movement of structures in the BWR is needed.	Bounding the physical motion of the steam separators/dryers (or other internal structures) is beyond the scope of the SOARCA analysis. However, the extent to which structural relocation might cause resuspension of aerosols deposited on these structures was described in the response to item 24..
27	0	Henry	Buoyancy flows in the containment are not part of the calculations. They need to be discussed, along with the concern that any cases that are more important are not being neglected.	A sensitivity calculation was performed to examine potential effects of natural circulation flow within the drywell. A summary of results have been added to documentation of the MELCOR calculations. Mixing of the drywell atmosphere by circulation flow was found to not significantly affect results.
28	0	Kowieski	Why is siren used as particular points? It gives the impression that people move at this time. Suggest changing to "siren + ES message".	The figures and associated text describing evacuation timing have been updated to clarify population motion.
29	0	Kowieski	Reconsider the 1 hour allowed to evacuate after second siren. (SOARCA team requested feedback from the committee on this 1-hour time.) Peach Bottom long term station blackout.	The data available to the SOARCA analysis team is consistent with the time lines provided in the documentation to within 15 minutes. 1 hour is also standard in evacuation time estimates. Sensitivity study #3 was performed which includes a delay of an additional 30 minutes in the response of the public. This delay did not result in any changes in the off-site consequences relative to the baseline case.
30	0	Vierow	Sensitivity studies could be done here. Some parameters are plant specific, e.g. bus availability, while others are random, e.g., weather, time of day. These should be distinguished in the report. Peach Bottom long term station blackout.	The availability of buses is captured in the "tail" cohort. Although evacuation time estimates could be shortened to account for the potential of night evacuations, examining shorter evacuation times would not be relevant as even the current

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				evacuation times allow for populations to be evacuated prior to radionuclide exposure. It should also be noted that daytime evacuation is assumed to represent the most demanding public evacuation scenario while nighttime staffing of emergency response organizations is also assumed to provide additional conservatism. Sensitivity studies have been conducted and documented to explore other aspects of the evacuation planning. Further exploration of these issues will be included as part of the SOARCA uncertainty quantification effort.
31	0	Kowieski	The evacuation time of the Special Facilities is late and will not go over well with the public.	The relevant text has been updated to clarify that these groups shelter earlier in the event and then evacuate the time specified.
32	0	Canavan	Specify when each group is notified in order to show that none of them are being neglected.	The text and figures have been updated to clarify this point.
33	0	Leaver	Discuss the best way to present the data. Consider showing a histogram to see the differentials.	The off-site consequence graphs have been changed to bar chart format for clarity.
34	0	O'Kula	The y-axis for the unmitigated STSBO off-site consequence graph will be confusing to the public. It is a conditional risk, or risk given that the accident (STSBO) has occurred. So risk here is not per year, but per the accident occurring. If we say "risk" alone, it should factor in the mean estimate of the frequency (3E-07) and show units on the order of 10^{-11}. We will need to have these plots be standardized one way if "conditional risk" results are portrayed, and another way if absolute risk is being shown. As it stands now someone will see the y-axis numbers and misinterpret the result, e.g. try to relate it to meeting the safety goals.	Figure and table captions have been modified to clarify that conditional risk values are presented.
35	0	Stevenson	Note that "mean" is conservative with respect to the "median"	Agreed, however the use of the mean (expected) value is consistent with the best estimate

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				objective of the SOARCA project.
36	0	Leaver	The data is extremely important but may lead to a negative perspective. Consider deleting this data in the NUREG.	This comment refers to a peculiarity in the PB off-site consequences relating to the small population close to the plant and the relative effectiveness of evacuation procedures within 10 miles of the plant. This leads to low conditional risks in this region relative to the 20 mile region. The text and graphics have been updated to aid proper interpretation.
37	0	Gabor	Is a loss of ac power a unique event? It may lead down a path that is different than for a non-blackout event. Blackout may not be conservative. Consider when EAL is triggered.	In the case of the BWR, the top of active fuel would be reached in 15 minutes in the blackout event. It is unlikely that the loss of ac power would be more severe.
38	0	Leaver	The effect on risk of the declaration of EAL (Emergency Action Level) needs to be captured.	This comment has already been covered by the response to items 37 and 40 regarding the timing of the declaration of general emergency. This will be considered as part of the uncertainty analysis effort.
39	0	Leaver	Applying the LNT seems inconsistent with the habitability criterion.	The return criteria represents a best estimate of existing emergency response procedures and policies. The different dose response models are provided to aid in the interpretation and comparison of the predicted off-site consequences.
40	0	Kowieski	The seismic analysis time line suggests that after declaration of GE by the plant, sirens and EAS message could be activated within 45 minutes. Based on the actual field experience, it takes approximately 15 minutes for the nuclear power plant to notify the state authorities, and may take additional 38-40 minutes, before the sirens activation and EAS message are completed. Therefore, total time required to complete the A/N sequence may vary between 53-55 minutes.	The timelines used in the analyses are very near the times experienced in exercises. To address any difference in timing, Sensitivity #3 was performed increasing the initial delay in the notification of the public by 30 minutes.
41	0	Kowieski	It appears that the existing documents do not address the notification of public in case of	Data has been added to section 6.2.5 justifying the assumption that sirens operate correctly.

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			siren(s) failure. Should a siren fail, it may take additional 45 minutes to notify the affected public by Route Alerting procedures.	
42	0	O'Kula	How would different values for the surface roughness length change the risk results at the mean (average) level? Could a short paragraph or limited sensitivity analysis be used to address whether this is important within the 10-mile EPZ, and within the 20-mile region?	The surface roughness length will be considered as part of the SOARCA uncertainty quantification effort.
43	0	Canavan	Safety valves and pilot operated relief valves play a significant role in the accident sequences analyzed in SOARCA. Both the successful operation as well as the failure modes under beyond design basis conditions are clearly significant in the analysis. While the failure modes considered in the SOARCA analysis are, in the opinion of this reviewer likely, others with more expertise in the area of safety valves should be consulted. (cf. detailed comments submitted by Canavan 10/14/09 for examples)	The effects of reasonable variations in SRV failure criteria were examined in sensitivity calculations. Results of these calculations have been added to SOARCA documentation (see App A, Section 5.6.2)
44	0	Leaver	Volume III, Section 3.1.4.1 is confusing. It states that, "One unmitigated case was considered." But then it goes on to discuss two unmitigated cases: a first case with RCIC black run and use of portable power supply credited, and a second case with RCIC black run and portable power supply not credited.	In the SOARCA documentation for the SBO scenarios, the term "mitigated" refers to the use of additional safety equipment required under 10CFR50.54(hh). In this case, two variations of the unmitigated case are described. The text has been modified to provide clarity.
45	0	Leaver	Regarding the matter of the 0.5% who choose not to evacuate, it is suggested that results be reported for non-voluntary risk (i.e., 100% evacuation) and that the voluntary risk (for those who choose not to evacuate) be reported as part of the sensitivity study.	Consequence results for the non-evacuating cohort will continue to be included in the overall consequence calculations but a short paragraph has been inserted to describe the fraction of the emergency phase risk within 10 miles of the plant that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.

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46	0	Leaver	A summary of fragilities for key components (e.g., Surry low pressure injection and containment spray; PB torus integrity, RCIC) for the 0.3 to 1 pga earthquakes would be useful, or at least the basis for assuming that they can perform their function after the earthquake. Both Surry and Peach Bottom are members of the Seismic Qualification Users Group (SQUG) which was developed by industry for older plants and may have some useful data. Dr. Robert Kassawara (650 855 2775) is the EPRI Program Manager for SQUG. NRC is aware of the SQUG database, having considered it in conjunction with resolution of USI A-46. NRC's Goutam Bagchi was involved in this. The EPRI seismic margins report (NP 6041, Rev. 1 – a licensable document) may also be useful.	Fragilities of key components are being examined by NRC staff but are not available for inclusion in the SOARCA documentation. In general, the importance of future research into seismically initiated events has been identified by the SOARCA project but is considered beyond the scope of the current SOARCA project.
47	0	Leaver	The LCF consequence curves (such as Volume III, Figure 64 and Volume IV, Figure 145) might be more meaningful if the risk was presented for a given radius (or ring of some average radius) as opposed to plotting the risk to all residents inside a given radius.	The analysis team felt that the current format provided the easiest interpretation. The format has been changed from curve to bar chart format to further improve interpretation.
48	0	Leaver	SOARCA indicated that it is pursuing this, but just for the record, the Ba release for Peach Bottom STSBO both without (Figure 38) and with (Figure 45) RCIC Blackstart looks very suspicious. It is 4 x the iodine release early, and ends up nearly the same as iodine in the longer term, in the range of 6% to 8%.	The sharp Ba release post vessel breach is a result of a chemical reaction with unoxidized Zirconium in the melt. These releases are entirely ex-vessel (MCCI) and are not subject to the same deposition mechanisms that the volatiles experience.

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49	0	Leaver	Land contamination results probably do not belong in the SOARCA reports, but was there any condemned land in any of the sequences?	Condemned land approximations require the use of economic models which where explicitly excluded from the scope of the SOARCA analyses. A dose level was specified as a return criterion, but the extent of land that might exceed this criterion for a given scenario and time period was not calculated.
50	0	Leaver	Volume III, page 8 – Second full paragraph: "The process identified two sequence groups which met the screening criteria of $1x10^{-6}$ per reactor-year for containment failure events…" looks wrong. Should it not be "…$1x10^{-6}$ per reactor-year for core damage frequency"?	Agreed. The text has been modified appropriately.
51	0	Leaver	Suggested parameters for uncertainty and sensitivity analyses: 1. Higher confidence weather. The risk from this (i.e., the higher LCF consequences together with the lower frequency of the higher confidence weather) can then be compared with the risk from the mean weather. 2. Habitability criterion (e.g., cut by a factor of 5, and/or vary the costs used in the decision as to whether contaminated areas can be restored to habitability). See Volume I, page 65 and 67. 3. Relocation criteria (e.g., what is additional LCF risk for 5 rem for normal relocation?) See Volume I, page 66. 4. How about a no ad-hoc evacuation sensitivity case? 5. Time for mitigation measures (e.g., 8 hours for transporting and connecting the Surry diesel-driven injection pump could be increased to 12 hours). See Volume I, page 23.	These items will be considered for the SOARCA uncertainty quantification effort. In particular however: Item – 10: Sensitivity calculations were performed for the BWR LTSBO scenario to examine the effects of enhanced containment leakage on source term. Results are summarized in App. A, Section5.6.1. Item – 11: If an SRV sticks closed, the next SRV would pick up the load and begin cycling. This possibility was taken into account in selecting the confidence level for stochastic failure of the SRV. Rather than using the "median" probability of 0.5, a value of 0.9 was used to represent the possibility that a second SRV might be demanded, increasing the effective number of cycles to failure. In addition, the design of the Peach Bottom SRVs make it unlikely that they will seize in the closed or partially open position.

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			6. Aerosol deposition velocity in consequence calculations. See Volume I, page 64. 7. Shielding factors. See Volume I, page 65. 8. Time of Declaration of GE. See, for example, Volume IV, Figures 131 and 132, which have GE at 2 hours. The paragraph above Figure 131 says, "It is assumed under this scenario, that plant operators would recognize rather soon that restoration of power within 4 hours is unlikely. A 2 hour period from loss of power was selected as a reasonable time for declaration of a GE…" This certainly is reasonable, but the plant operators could also think that power might be restored and thus delay the declaration of GE a bit longer, say until 3 hours. 9. Delay times for shelter and evacuation By inspection, modest differences in the delay times won't matter much, but it is good to demonstrate it. 10. What is the effect of degradation of containment leaktightness due to an earthquake in the 0.3 to 0.5 pga range, and in the 0.5 to 1.0 pga range? For example, consider DBA leakage x3 for 0.3 to 0.5 pga, and x10 for 0.5 to 1.0 pga. 11. This matter was brought up in one of the first two meetings by Jeff Gabor. What about a sensitivity on the radionuclide release assuming that the SRV sticks closed after excessive cycles (see Volume III, Figure 31)?	

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52	0	Gabor	Calculate the BWR Main Steam Line heatup without assuming a stuck open SRV. In addition, run a case without the SRV failing open, but with a Main Steam Line failure.	Several new sensitivity calculations were performed and results added to the BWR MELCOR documentation (App A, Section 5.6.2). A specific case was run assuming SRV seizure was delayed, allowing more time for main steam heat-up and failure by creep rupture.

Notes

State-of-the-Art Reactor Consequence Analysis (SOARCA) Program
Summary Document Peer Review Comments

Revision	Date	Description
0	2-Jul-09	Review version issued to peer review panel for July 28-29, 2009 kick-off meeting
1	15-Feb-10	Review version incorporating peer review panel comments from first two review meetings

#	Rev.	Reviewer	Comment	Resolution
1	0	Clement	In tables 1 and 2 of the executive summary, there is roughly an order of magnitude difference in CDF for SBO between Peach Bottom and Surry. What is the reason for such a difference?	It must first be recognized that one plant (PB) is a BWR; the other (Surry) is a PWR. The inherent differences in reactor coolant system and safety system configurations between these two designs greatly affect the way in which they react to a loss of offsite electric power. Both plants have similar onsite, back-up ac power capabilities (diesel generators), which respond similarly to an earthquake (i.e., similar fragilities). However, the BWR also has an onsite back-up dc power system that supports operation of two, independent steam-driven coolant injection systems, while the PWR has a single steam-driven pump to provide auxiliary feedwater to the steam generators. These differences in reactor design, and others, collectively lead to the differences in station blackout frequency.
2	0	Clement	For the selection of sequences results of PRA level 1 are used with screening criteria on CDF. It is also stated (p.8) that full scope level 3 PRAs are not generally available. What about level 2 PRAs?	Licensees generally maintain a limited scope Level 2 PRA for the purpose of estimating large early release frequency (LERF). Licensees who have been granted license renewals (specifically Peach Bottom and Surry) or who have applied for license renewal have limited scope Level 3 PRAs for the purpose of evaluating severe accident mitigation alternatives (SAMAs). When the SOARCA sequence selection was being performed, the staff was in the process of developing a small number of Level 2 SPAR models and, thus, decided to rely on Level 2 PRA information and insights provided by licensees during the SOARCA sequence selection process.
3	0	Clement	It is stated in § 1.5 that future work will deal with uncertainty analysis. It is not clear from the brief description how epistemic uncertainties, inherent to the complex severe accident processes will be	Text has been added to Section 1.8 to provide additional detail. While the technical approach to the SOARCA uncertainty analysis will be discussed as part of the peer review process, the

#	Rev.	Reviewer	Comment	Resolution
			taken into account.	final results will not be available before the conclusion of that process. The final uncertainty results will be reviewed by the NRC staff and a review by the Advisory Committee on Reactor Safeguards is also anticipated.
4	0	Clement	The approach for accident scenario selection described in § 2.1 uses a CDF screening value rather than a radionuclide frequency release value. Could the limitations of the methodology be discussed in more details?	Although the SOARCA scenario screening criteria uses CDF as a screening metric (because it is available), it is important to note that all of the SOARCA scenarios (unmitigated sensitivities) result in containment failure, very large leakage or bypass. While a large fraction of the scenarios considered in a full scope PRA effort do not proceed to containment failure the SOARCA scenario selection has resulted in and focused on sequence groups which in fact reflect radionuclide release frequencies. Further, by using an even more inclusive criterion for bypass events we are reasonably assured of capturing events which dominate release (and risk). In the case of the SBO with tube failure, tube failure probabilities from an independent study were employed.
5	0	Clement	The SPAR results were compared to the utility PRAs. Some examples of comparison are given (e.g. SGTR for Surry) but a synthetic comparison would be useful.	The SPAR models, which address internal initiating events, have been benchmarked against licensee PRAs in conjunction with the implementation of the Mitigating Systems Performance Index (MSPI). This benchmarking activity included a cutset-level review to check the structure of the SPAR logic model. In addition, licensee PRAs have been peer reviewed to either Nuclear Energy Institute guidance or the combined ASME/ANS PRA standard. The staff has used licensee external event PRAs to develop a limited number of SPAR external event models. As a result, the SPAR external event models yield results and risk insights that are similar to licensee external event PRAs.

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6	0	Clement	It is said that no internal event meets the criteria for Peach Bottom (§2.4.1). Are they far from it?	Internal event station blackout may be the only one that comes close. In any event, the seismically initiated STSBO is a rapid loss of heat removal that bounds other events so it represents a convenient surrogate for other sequences.
7	0	Clement	Only qualitative arguments are given for justifying the 48h truncation of releases (§3.2). Maybe a sensitivity study on a selected scenario would be useful.	It is the NRC position that the assumption that ad hoc measures would not be taken within 48 hours to change the course of a severe accident progression is not credible. Since the nature and efficacy of these ad hoc actions cannot be predicted a priori, extending the release beyond this point unperturbed is inconsistent with the best estimate objectives of the SOARCA project. At a minimum it is reasonable to assume that the release rate predicted at 48 hours would represent an upper bound on the releases beyond that point. Given that the SOARCA analyses suggest that accident progression extends for a much longer period of time than earlier studies suggested (cf. conclusions section of the executive summary), it is reasonable to consider studies to examine what actions might be taken to mitigate long term releases. It is not feasible to conduct such a study within the scope of the SOARCA project however. For these reasons the releases for these scenarios were truncated at 48 hours (72 hours for the Surry LTSBO)
8	0	Clement	It is not clear (§4.4.1) whether MELCOR 1.8.6 or 2.0 version was used.	Only version 1.8.6 was used for the SOARCA calculations.
9	0	Clement	In equation 1 of § 4.4.1, the creation of radio-nuclides by neutron absorption does not appear explicitly.	Although generation by neutron absorption does not appear explicitly in the equation, generation by neutron absorption is related to the loss term and is included in the overall radionuclide inventory methodology.

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10	0	Clement	It is said in §4.4.5 that TRITON prediction is at a level of accuracy consistent with other methods. More information would be useful.	This sentence has been removed. The relevant evaluation is included in the reports cited and a longer description here would not substantially benefit the SOARCA documentation.
11	0	Clement	The meaning of last sentence of § 5.4 is unclear.	This sentence indicates that plume segments that were trivial were broken into longer time intervals. The text has been reworked.
12	0	Stevenson	Other initiating events might be considered including: • Narrow or wide body jet aircraft crash • Ash fall from volcanic eruption loads on safety related structures other than containment and its effects on diesel generator intake filters • Seismic induced liquefaction or differential ground displacement • Certain flooding phenomenon caused by landslides, upstream dam failure and tsunamis • Internal flooding due to large flat bottom tank rupture. • Heavy load drop	Liquefaction, dam failure, and landslide effects were implicitly included in the seismic scenarios already selected for the SOARCA project. In the case of the SBO event the diesel generators were assumed not to operate which is similar to the volcanic ash scenario. In general, the other sequences mentioned did not arise from the SPAR analysis of the plants. It can be argued that the release characteristics of these other events are already adequately represented by the current SOARCA scenarios. On a practical note, there is very limited risk information about external events other than seismic events and fires. Most of these types of events were screened out on the basis of initiator frequency in the IPEEEs and, accordingly, neither the staff or licensees have current estimates of the CDF due to these types of events.
13	0	Vierow	Executive Summary, page x, first paragraph: Will other representative plants be analyzed, as was done for NUREG-1150? A statement to this effect appears somewhere well into the document, but the question arises in the reader's mind much earlier.	A statement regarding the potential for future analyses following review of the Peach Bottom and Surry results has been added to the objectives section of the executive summary.
14	0	Vierow	Executive Summary, page xix, Table 3: add a statement in the text as to why the time to lower head failure for Peach Bottom and the time to start of release to the environment are the same.	A clarifying statement has been added to the accident progression and radionuclide release section regarding the proximity of drywell shell melt through and lower head failure (15 minutes)
15	0	Vierow	Page 7, First paragraph and Table 9: consider	The purpose of the table is to provide historical

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			adding data from NUREG-1150 for the other 3 plants. Is the LOCA category for all LOCA's? If the "Internal Initiators, Fire and Seismic" is changed to "Internal Initiators: reactor coolant pump seal LOCAs", then the text and table would appear consistent with each other.	context for the PB and Surry analysis in SOARCA. Additional plants may be included when we get to the point of discussing other design classes The LOCA category includes LOCAs that are initiated by pipe break events. Transient-induced LOCAs are captured under the other categories (e.g., the SBO and TRANS categories include induced RCP seal LOCAs and stuck-open SRV LOCAs). An annotation has been added to the table. The column labeled "Fire and Seismic" under the "Internal Events" heading is a duplication of the right column, and has been removed.
16	0	Vierow	Misc typos throughout	Editorial and typographical errors will be addressed by a technical editor once substantive changes to the NUREG documentation has been completed.
17	0	O'Kula	Page xi (editorial) - 2nd paragraph, 2nd line: American Society of Mechanical Engineers'	Error has been fixed.
18	0	O'Kula	Page 3 (editorial) - A introductory, transition sentence or two is needed ahead of the first paragraph on page 3. The paragraph reads as though it is the present tense, e.g. " Yet the possibility remains . . .". Suggest a statement to note that it is in reference to the state of knowledge during or after WASH-1250.	The paragraph has been deleted. While it was generally accurate (but could be improved) it did not substantially clarify the WASH-1250 discussion.
19	0	O'Kula	Page 15 (minor importance) - Suggest that first use of SPAR models be noted with a citation/reference.	The SPAR models are not publically available and there is no NUREG or NUREG/CR that provides a broad overview of their scope, development, and results.
20	0	O'Kula	Page 22 (minor importance) - Was short-term Station Blackout from a seismic event for Peach Bottom included or dropped?	The PB STSBO was retained even though it fell below the screening criteria because it is an important loss of heat removal event in terms of timing. This is described in the Peach Bottom

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				results (cf. Section 3.2). The description on this page provides just those scenarios that exceeded the screening criteria. A clarifying paragraph has also been added to the text on this page.
21	0	O'Kula	Page 57 (medium importance) - Is the selection of METCOD still based on machine time considerations? Would runs using METCOD=5 be too machine-intensive to run? Is there a technical basis for LHS more so than Stratified Random Sampling (METCOD=5; with NSMPLS=24; so that every hour of the 8760 hour data set is sampled)?	Current runs use about 1000 weather trials and required about 2 hr CPU time for LNT and about 20 hr CPU time for dose truncation. Increasing the number of runs to 8760 would increase the CPU time by almost a factor of 10. Although this could be pursued to demonstrate convergence for LNT case it might be prohibitive for the dose truncation runs. It is important to recognize that this effort is unlikely to change the mean consequence results cited by the SOARCA documentation.
22	0	O'Kula	Page 58 (medium) - Table 12 shows characteristics of the two years of meteorology considered for each plant. For Peach Bottom, the predominant wind changed by nearly 180 degrees (SSE to N). For Surry, the number of hours with precipitation went from 388 to 521. Was any work done to determine why one year was more representative over another year in each case?	For Peach Bottom, the wind direction issue was resolved by plotting wind roses for the two years, 2005 and 2006. The wind roses were very similar even though the peak dominant wind direction for the two years is different by almost 180 degrees. The "Predominant Wind" data given in the table is correct but misleading and has been removed from the table. For Surry, the issue is the number of hours of precipitation. The data indicate that there are 34% more hours of precipitation in 2004 than in 2001. Even so, precipitation only occurs during 6% of the hours of 2004, so precipitation is not a factor the large majority of the time. The resulting difference in the predictions is not expected to be large.
23	0	O'Kula	Page 64 (medium importance) - Deposition velocity is an area where the uncertainty analysis capability in WinMACCS could offer a big improvement over the point value selection process that was applied in previous studies.	Older calculations used a single deposition velocity to represent all aerosol particles, regardless of size. The baseline SOARCA calculations treat deposition velocity as a function of aerosol diameter. The uncertainty study will

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			Could this be entered as a distribution rather than a single point estimate? Ref. 48 is described as an expert elicitation for deposition velocity. Could this report be made available to know the values used?	also treat deposition velocity for each aerosol size as a distribution rather than as a point estimate. Ref 48 is still in draft form and is not yet available for release.
24	0	O'Kula	Throughout (major importance) - What kind of uncertainty analysis for the overall SOARCA project is envisioned? Will there be any attempt to examine aleatory and epistemic classes of uncertainties?	Text has been added to Section 1.8 to provide additional detail regarding the uncertainty analysis. While the technical approach to the SOARCA uncertainty analysis will be discussed as part of the peer review process, the final results will not be available before the conclusion of that process. The final uncertainty results will be reviewed by the NRC staff and a review by the Advisory Committee on Reactor Safeguards is also anticipated
25	0	Gabor	Given that there has been some criticism of the CDF screening process and its ability to capture the significant risk contributors, could there be any value in comparing the consequence results from the published Peach Bottom and Surry Level 3 PRAs from License Renewal with the current SOARCA results?	While CDF was used as a screening criterion, other criteria were also used to identify specific sequences leading to radionuclide release. As a result, the CDFs associated with the SOARCA sequences effectively represent release frequencies. This is not the case for a typical PRA analysis in which many of the sequences do not lead to release. In addition, since licensee PRAs are not as detailed as the SOARCA studies and do not explicitly include external events it is not clear how a comparison between the two would be conducted. For example, the Peach Bottom and Surry license renewal applications and the staff's corresponding EIS do not provide specific information on release frequencies and offsite consequences of SOARCA-like sequences. The staff had the benefit of this information and the underlying Level 3 PRA, however, during the SOARCA sequence selection process.

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				The SOARCA CDF screening process has identified sequences which characterize broad classes of loss of heat removal and small loss of coolant events including bypass loss of coolant accidents.
26	0	Henry	Add common-mode failure to list of items not included in scope. Shutdown and low power also need to be considered to some level of detail since those states have an unknown configuration until the reactor is at full power.	The SPAR models contain a comprehensive treatment of support system dependencies, phenomenological dependencies (e.g., loss of NPSH to ECCS pumps), and component-level common-cause failures. One of the principle purposes for conducting the SOARCA project is to update the quantification of offsite consequences found in earlier NRC studies (e.g., the Sandia Siting Study), which have focused on accidents initiated during power operations. Text has been included in Section 1.4 to clarify the basis for the scope of SOARCA.
27	0	Peer review committee	Provide technical justification for each item in the report.	This comment is too broad to be addressed effectively. Clarification is needed. Is it possible that information was lost in transcribing this comment?
28	0	Gabor	Defend not including dual plant failures in the report.	Multiple unit failures are discussed in section 1.4 of volume I. Additional text has also been included on this and other specific classes of events in the executive summary and Section 1.4.
29	0	Leaver	Discuss in the document whether "screening" of events is acceptable.	This discussion is provided in section 1.5 and the executive summary..
30	0	Stevenson	Explain in the document why general aviation small aircraft impact is not considered.	These sequences did not arise from the SOARCA sequence analysis. It is also possible that the existing SOARCA sequences bound the consequences of small aviation impacts.
31	0	Leaver, Henry	Consider increased leakage and varying the amount of leakage at different times in the event sequence. Increased leakage early in the accident may lead to higher release. Current PRA	Containment leakage rates are based on available technical specifications and PRA data. While the adequacy of these data may be an important area of investigation, such an investigation cannot be

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			may not be adequate. If release into the containment is seen within the first 7-8 hours, SOARCA must be able to field questions about early environmental release. TMI-2 also gives us the perspective that a closed system can release fission products to the containment within a few hours, i.e. when the reactor vessel is intact.	undertaken within the scope of the SOARCA project. It is important to note that with the SOARCA containment performance treatment, releases to the environment prior to containment failure do occur. The PB analysis, for example, accounts for leakage from containment prior to failure. The leak rate is defined by a fixed area, calibrated to the Tech Spec leak rate at the design basis internal pressure. Therefore, leak rate increases as internal pressure increases and releases to the environment begin many hours before containment structural failure occurs
32	0	Mrowca	In the final report, provide probabilities, or HRA numbers, used for mitigation. (cf. J. Schaperow slide 28 in peer review kick-off meeting)	A full scope HRA analysis of the mitigative actions is beyond the scope of the SOARCA project. However, screening estimates of the human error probabilities (HEPs) are being developed in conjunction with a study to assess the 10CFR50.54(hh) mitigative actions. Specifically, these HEPs are being used to modify the staff's SPAR models to assess the CDF impact of these strategies. This work has not been completed and cannot be made publically available due to its security implications.
33	0	Stevenson	Consider the use of the term "mitigation". Mitigation implies a reduction of the consequences of an accident or an initiating event. It is also possible that operator or other actions could aggravate accident consequences. The term mitigation appears to bias any action.	This question relates to the treatment of so-called "operator mistakes," i.e., having a wrong impression of what to do coupled with an improper action or decision. As discussed in Section 2.3 of NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method," the SPAR-H method uses a set of performance shaping factors (PSFs) to distinguish among operator slips, lapses, and mistakes. That is, the human error probabilities are adjusted through use of the PSFs to account for the specific type of error that is relevant to the

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				operator action being assessed. The term "mitigation" is used intentionally to indicate successful operator actions as opposed to adverse operator actions. A full scope HRA analysis would be required to assess the probability of adverse operator actions. Such an effort is beyond the scope of the SOARCA project.
34	0	Mrowca	*Add to the report a description of "what is State-of-the-Art about SOARCA?"*	*The claim to state-of-the-art is established in* section 1.1 as well as in the executive summary. This claim is based on three characteristics of the SOARCA analyses Detail – in terms of the fidelity of facility representation, including auxiliary buildings and spatial resolution, as well as the representation of emergency response and evacuations Realism – In terms of the use of modern phenomenological models developed over the past 20-30 years as well as representation of current plant and emergency response procedures and public behavior Consistent – In terms of the tight coupling between traditional Level II and Level III analyses using scenario specific source terms and event progressions *rather than characteristic source* terms as in NUREG-1150 style analyses. Clarifying text has been added to Section 1.1
35	0	Henry	The current description of NRC sponsored studies includes the major improvements in understanding and analyzing the responses of representative BWR and PWR designs. These include the Reactor Safety Study (WASH-1400),	Agreed, this text has been added to Section 1.2.

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			NUREG-1150 and now SOARCA. In addition to the improvements in understanding and calculational capabilities, there have been numerous influential changes in the training of operating personnel and the increased utilization of plant specific capabilities. For example: • The transition from event based to symptom based Emergency Operating Procedures (EOPs) for the BWR and PWR designs. • The performance and maintenance of plant specific PRAs that cover the spectrum of accident scenarios. • The implementation of plant specific, full scope control room simulators to train operators. • An industry wide technical basis, owners group specific guidance and plant specific implementation of the Severe Accident Management Guidelines (SAMGs). • Improved phenomenological understanding of influential processes such as (a) in-vessel steam explosions, (b) Mark I liner attack, (c) dominant chemical forms for fission products, (d) Direct Containment Heating, (e) hot leg creep rupture, (f) Reactor Pressure Vessel (RPV) failure and (g) Molten Core Concrete Interactions (MCCI). • Proceduralized use of plant	

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			specific B.5.b systems. All of these have contributed to reductions in the likelihood of a severe accident as well as a reduced potential for radioactive releases to the environment. As such, they should also be identified in the historical background for SOARCA.	
36	0	Leaver	In the Executive Summary, emphasize mitigation effects. Consider deleting unmitigated results since these are not best estimate. Emphasize what was learned from mitigation analysis.	Additional text has been added to the mitigation measures section and elsewhere in the executive summary. The inclusion of both mitigated and unmitigated results is an important feature for the SOARCA study. Although additional mitigation measures were established under 10CFR50.54(hh) with the intention of providing defense in depth for security related events, the unmitigated results were also included to provide a basis of comparison to earlier studies as well as to assess the benefits of these additional measures. It is also important to note that the precise impact of the additional 10CFR50.54(hh) procedures on the underlying frequency used to identify the SOARCA scenarios would require a more rigorous risk and human reliability analysis than was feasible within the scope of the SOARCA project.
37	0	Gabor	Industry heavily focused on PRA quality and methods. Relate SOARCA to existing risk informed regulation.	SOARCA has demonstrated areas for potential improvement in PRA methods, particularly characterization of plant response, that may ultimately find its way into the development of PRA methodology. The SOARCA project is not intended to modify existing NRC rulemaking or supplant existing PRA standards however.
38	0	Leaver, Clement	Add a faster LOCA for completeness. (note from Vierow - There was discussion that such events are of too low a frequency.) In France, faster	The medium and large LOCA frequencies for both Peach Bottom and Surry are 2 to 3 orders of magnitude below the SOARCA screening criteria.

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			sequences are used to study the consequences even though they are of lower frequency and not best estimate.	Including sequences with frequencies in this range would represent a fundamental shift in the SOARCA objectives and methodology.
39	0	Gabor	SOARCA needs to have the claim that it has captured all of the risk. Therefore, completeness is needed.	The objective of the SOARCA project is to characterize the off-site consequences and risk of event sequences which reflect the important characteristic severe accident sequences for common power reactor types.
40	0	Stevenson	A Station Blackout may not be the worst consequence of a seismic event. A seismic event in the 10^{-6} to 10^{-7}/yr probability of event range may be sufficient to cause by fault displacement, liquification, or subsidence a movement that could rupture the containment and cause structural collapse or rupture of RCS piping or components. This potential needs to be addressed to show hopefully such events are below the 10^{-7}/yr threshold for consideration.	Consideration of a large seismic event that fails containment and ruptures RCS is already addressed in executive summary.
41	0	Clement	The dose limit for radiation workers endorsed by the Health Physics Society that was 5 rem/yr is now 2 rem/yr. (cf. Bixler slide 7 from peer review kickoff meeting)	The dose limit for radiation workers was only mentioned as a point of comparison. It was not used as the basis for choosing any of the dose truncation criteria used in the study.
42	0	Leaver	Between the slides and the report it appears that there are five event types which SOARCA does not address: multi-unit events, spent fuel pool accidents, low power or shutdown events, security-related events, and the very large seismic event causing simultaneous breach of containment and a LOCA with ECCS failure. Discussion of the reasons for not addressing these event types is spread out in the report and is somewhat uneven. It is suggested that the reasons for not addressing these five event types be discussed in a more even-handed,	Text has been added to Section 1.4 describing the basis for not including these events in the SOARCA analysis.

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			consolidated manner, probably in Volume I. The reasons for not addressing a given event type might include, for example: plans exist to address it in the future, it is judged to be low priority, or it is already adequately addressed somewhere else. This discussion is part of the matter of completeness which, along with the screening approach and sensitivities, is very important to the credibility of the SOARCA effort. It is certainly acceptable to carry out the project without claiming to be complete, but the SOARCA effort should be as complete as practical and should deliberately defend its degree of completeness.	
43	0	Leaver	It would seem appropriate and desirable to benchmark MELCOR fission product releases against the TMI-2 accident and SFD.	The MELCOR code has already been extensively benchmarked. Adding to this benchmarking data base is not within the scope of the SOARCA project. Validation against the TMI-2 event which had a very limited release would also be of limited benefit considering the accident sequences of interest to the SOARCA project.
44	0	Leaver	Some of the support points for screening are marginal. For example, the first full paragraph on Vol. I, page xi, justifies 1E-6 as 1% of CDF and uses the 1E-4 QHO as the CDF. But these days, CDFs for U.S. plants are more like 1E-5 to 1E-6, and 1% of this is a factor of 10 or more less than 1E-6. Another example [of marginal support points for screening] is in the next paragraph [second paragraph on page xi] where it is stated, "Another	The scenario selection process employed in the SOARCA project is based on available level 1 PRA data. This resulted in selection of scenarios which are also representative of broad classes of transients. This scenario set has also been enhanced by including events with assessed frequencies below the screening criteria that are of historical interest. Text has been added to the scenario selection section of the executive summary to emphasize these points.

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			way to judge the impact of low-frequency events is to consider the increase in the latent cancer consequences that would be necessary to offset the lower frequency." This is a good argument and should be used. But what about early fatality consequences which are more visible and will start to show up as frequencies get lower? It might be wise to cite screening precedents. See, for example, NUREG-1420 which indicates that consequences with frequencies lower than about 10^{-7} per year "...are not meaningful for decision making," and Regulatory Guide 1.174 and the U.S. Reactor Oversight Program significance determination process, among others, which use a frequency threshold for non-risk-significant changes. The best screen is one where you defend its reasonableness and its application, but then show you don't really need to lean on it too much.	Existing guidance that is based on changes in CDF (e.g., RG 1.174 and the SDP) are not directly applicable to SOARCA because this regulatory guide was developed for different purposes. Specifically, the concept behind such guidance is to provide an aid to regulatory decision-making (e.g., does a proposed license amendment cause an unacceptably high change in risk?). This is a fundamentally different concept than identifying the most likely sources of risk.
45	0	Leaver	For all of the sequence types, the mitigated sequences appear to be the only ones that survive the screen. (see detailed post kick-off review comments by Leaver). It may make sense to lump the unmitigated sequences, along with uncertainty and sensitivity results, into something called sensitivity studies rather than call them out separately.	A full scope HRA/PRA analysis would be required to provide an assessment of the frequency of the mitigated and unmitigated accident sequences. The SOARCA project addresses the uncertainty in the frequency and efficacy mitigation by running both mitigated and unmitigated simulations. In evaluating the event frequencies assigned to the unmitigated cases, it is important to remember that these frequencies do not account for the B.5.b procedures. For example, 2E-5/yr is the original assigned frequency for the LTSBO from existing external event PRA. Table top exercise showed it could reasonably be mitigated. However, we performed an analysis of the event

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				assuming it was unmitigated but the portrayal of absolute risk did not credit any reduction in frequency due to the 10CFR50.54(hh) mitigation. Text has been added to the executive summary to clarify this.
46	0	Leaver	The bottom paragraph on page 7, Vol. I is not very clear. An example would help.	Agreed. The purchase and development of procedures for diesel-driven pumps required by 10CFR50.54(hh) has been added to the text as an example.
47	0	Leaver	It is very reasonable to limit dose results to 10 miles as was done in the Executive Summary, based on the NRC safety goal policy. The dose results elsewhere in the report should be limited to 50 miles. (see justification given in detailed post kick-off review comments by Leaver)	Results in older studies went out to much longer distances: 500 mi in the siting study and 1000 mi in NUREG-1150. SOARCA is a departure from these earlier works by limiting consequence analysis results to shorter distances. The final determination by the NRC staff is to limit the consequence predictions to a 50 mile radius which is reflected in revision 1 and subsequent revisions of the documentation.
48	0	Leaver	References should be available and traceable (e.g., "Keith Eckerman [51]" should be a memorandum or some such document so the public can access it).	The reference to the Eckerman memo has been revised. The specific modifications to the dose conversion factors based on the Eckerman recommendations are described explicitly in the text of the report.
49	0	Leaver	Regarding the matter of the 0.5% who choose not to evacuate, it is suggested that results be reported for non-voluntary risk (i.e., 100% evacuation) and that the voluntary risk (for those who choose not to evacuate) be reported as part of the sensitivity study.	Consequence results for the non-evacuating cohort will continue to be included in the overall consequence calculations but a short paragraph has been inserted into the executive summary to describe the fraction of the emergency phase risk within 10 miles of the plant that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.
50	0	Leaver	The ES should be changed to make more visible the main objectives and conclusions from SOARCA. The objectives are clear and are summarized on slide 4 of the presentation,	The executive summary has been revised to provide clarity. Specifically a detailed bulleted objectives section has been added as well as a detailed bulleted conclusions section to be more

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			"SOARCA – Scenario Selection and Mitigation Measures". A text version of these objectives appears in the ES (page ix), but the objectives are somewhat run together and not very visible. Conclusions are given on slide 9 of the same presentation and appear in text form to some degree in the ES but are not succinct and visible.	succinct and visible.
51	0	Leaver	There should be further discussion on what the important results and conclusions are involving the full peer review group and after sensitivity and uncertainty results are available. It is suggested that the results and conclusions be divided into main, high-level conclusions, and supporting results. (see strawman outline provided in detailed post kick-off comments by Leaver) The main conclusions should be followed by a set of more specific results which support and amplify the conclusions (e.g., accident scenarios progress more slowly with smaller releases; accident mitigation is likely (due to time and redundancy) and would be effective when implemented; emergency response is likely to be effective in significantly reducing health risk)	Text has been added to Section 1.8 to provide additional detail. While the technical approach to the SOARCA uncertainty analysis will be discussed as part of the peer review process, the final results will not be available before the conclusion of that process. The final uncertainty results will be reviewed by the NRC staff and a review by the Advisory Committee on Reactor Safeguards is also anticipated.
52	0	Leaver	An important result is that the long-term portion of the LCF risk (which is ~90% of the total risk) is controllable. This should be stated in Volumes III and IV and reflected in the ES.	The executive summary has been modified including text emphasizing this point in the offsite radiological consequences section.
53	0	Leaver	The executive summary should be written around and emphasize the realistic, best-estimate consequence results (i.e., the mitigated sequences). The sensitivity results can then be presented and discussed (including unmitigated sequences, uncertainty results, and other sensitivities). An important point here is that the main conclusions from SOARCA (whatever those	Additional text has been added to the mitigation measures section of the executive summary. The inclusion of both mitigated and unmitigated results is an important feature for the SOARCA study. Although additional mitigation measures were established under 10CFR50.54(hh) with the intention of providing defense in depth for security related events, the unmitigated results were also

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			end up being – see comment 20 b) apply even when sensitivity results are taken into account.	included to provide a basis of comparison to earlier studies as well as to assess the benefits of these additional measures. It is also important to note that the precise impact of the additional 10CFR50.54(hh) procedures on the underlying frequency used to identify the SOARCA scenarios would require a more rigorous risk and human reliability analysis than was feasible within the scope of the SOARCA project.
54	0	Canavan	As an EPRI project, Surry is updating their seismic PRA. The complete PRA is expected to be completed in early 2010. Canavan will inquire as to whether he can share preliminary results. (Sch. Presentation)	Updated seismic PRA information was not available as of mid January, 2010.
55	0	Henry	Consider whether catastrophic containment failure should be addressed. (Schaperow noted that the probability is about 10^{-7}, which is below the criteria of 10^{-6} unless it is a bypass. This was left out since evaluation capabilities are not currently sufficient.) (Sch. Presentation)	While it is acknowledged that more work must be done in the area of seismic impacts on containment structures, the treatment of seismic impacts on reactor containments used in the SOARCA project remains state-of-the-art within the nuclear safety community. The effort to advance this state-of-the-art is justified but far beyond the scope of the SOARCA project.
56	0	Canavan	NUREG-1855 (EPRI 101 6737) reports on treatment of uncertainties in risk-informed applications. The SOARCA team should refer to this report. (Leonard noted that epistemic portions will apply.) (Burns pres.)	The portions of NUREG-1855 relating to treatment of Level II PRA uncertainty will be relevant. This report will be considered in the development and execution of the SOARCA uncertainty quantification effort.
57	0	Henry	The definitions of "sensitivity" and "uncertainty" are needed. These will promote the decisions as to which sequences and cases need to be analyzed. For example, with the thermally-induced SGTR, does the base case quantify risk?	In general "uncertainty analysis" relates to the impact on the output from a model due to uncertainties in the model input parameters. "Sensitivity analysis" is an evaluation of how sensitive the model outputs are to the uncertainty in a specific input. In the context of the SOARCA project, these evaluations were made both by explicitly exploring different accident progression paths, without regard to the resulting sequence

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				frequency, as well as formal uncertainty analysis. In the case of the Surry thermally induced SGTR, the base case does a reasonable job of quantifying the risk of that specific scenario.
58	0	Leaver	The matter of completeness may be the most critical issue we have. How can the story on completeness be made? The Executive Summary was uneven-handed regarding completeness. (Schaperow noted that SOARCA is a truncated risk study.) • How does the NRC make the case for completeness? • For events just below the cutoff frequency, how can their deletion be justified?	The SOARCA screening procedure is intended to revise the consequence estimate of severe reactor accident sequences that are important contributors to overall reactor risk. The SOARCA results show that the consequences of these historically important sequences are significantly lower that previously estimated. More severe sequences may not be equivalent in terms of consequences but are small in terms of overall risk.
59	0	Gabor	We have a base method for performing consequence analysis, as has been presented to us. How do we incorporate results of sensitivity calculations into the consequence analysis?	The sensitivity of results to input assumptions is being explored in two ways. Alternative accident progression paths of interest to the peer review committee and the SOARCA analysis team have been explicitly explored. The results of these cases have been included along with the base case results in the SOARCA documentation. In addition, a more systematic input uncertainty analysis and sensitivity quantification evaluation for a specific accident sequence will also be performed for the SOARCA project. This systematic analysis will implicitly explore other accident progression paths in addition to those already examined by the SOARCA team. The outcome of the systematic uncertainty analysis will also be included in the overall SOARCA documentation in terms of the uncertainty in the primary consequence results.
60	0	Yanch	There may be more completeness than is stated in Volume 1 of the draft NUREG. The case needs to be made better.	Text has been added to Section 1.4 addressing completeness of the scenario selection process.

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			Add more references and point to more data. There is too much assuming what the reader already knows.	
61	0	Gabor	For the completeness story, focus should be on the Level I selection and screening process.	Text has been added to the executive summary addressing completeness of the scenario selection process.
62	0	Leaver	The completeness argument is fundamental. • Address the fact that there are no cliffs lurking below the screening cutoff • If security arguments are not to be addressed, state that security events are not expected to have an effect on SOARCA results. • With respect the Human Reliability (HRA), mitigation actions are considered in the SOARCA and they could drive the sequence below the screening cutoff.	The completeness issue has been addressed in previous comments (cf. item 58 for example). In general, the SOARCA project was intended to reevaluate the consequences of specific risk significant events. Specifically various sensitivity calculations have been conducted to explore accident progression sequences of interest to the peer review committee for potential "cliffs" in the data. A comprehensive consideration of security related events and HRA issues associated with 10CFR50.54(hh) mitigation procedures is also beyond the scope of the SOARCA project.
63	0	O'Kula	In Volume I, add lessons learned since NUREG-1150, and what is leading to the reduction in risk for these selected sequences. Are we smarter with our methods and tools? Have experiments given us insights that we didn't have before? Have any of the post-TMI requirements improved the outcome? Is it better operating training that eliminates sequences? What is driving the reduction acute and latent risk? If Volume I is the most read of the SOARCA NUREGs, then let's be clear on the sources of reduction in risk. {If the final report from NUREG-1150 is read, you get an appreciation on the changes between WASH-1400 (1975) and NUREG-1150 (1990)}.	Additional background on the development of the current state of the art has been included in Section 1.2.
64	0	Mrowca	Consider relooking Level I. State-of-the-Art was not done for seismic or fire PRA. It was used at the end of the analyses.	Although the SOARCA project has underscored the need for better data and analysis of seismic events, the SOARCA team believes that the SOARCA analysis does represent the current

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				state of the art in this area. For example, the American Nuclear Society has drafted a new standard for seismic PRA but this standard has yet to be exercised. Extending the state of the art in this area is beyond the scope of the SOARCA project.
65	0	Leaver	A systematic discussion that screened sequences are not fundamentally different from the ones looked at is needed.	Clarifying text has been added to the Scenario Selection section of the Executive Summary.
66	0	Yanch	Some data is referred to as coming from the utilities. Consider adding an independent source so that there is not an appearance of having flavored data.	Data is drawn primarily from plant-specific PRA and IPE analyses for which there are no independent sources of information. In the specific case of evacuation modeling, plant-specific data was supplemented by data from drill times. In addition, sensitivity calculations were performed independent of plant-specific data as in the case of battery life.
67	0	Leaver	Land contamination and security events are missing from this report. The security events, in particular, may likely draw claims of missing events.	While economic impacts associated with land contamination do influence the modeling of cleanup personnel, and the resulting exposure of this cohort to radionuclides, economic impacts associated with land contamination were explicitly excluded from the SOARCA results due to the complex nature of these calculations. Security events were also explicitly excluded from the SOARCA analysis to prevent materially aiding terrorists.
68	0	Leaver	Elaborate more on the screening process in the document.	Clarifying text has been added to the Executive Summary and Section 1.4.
69	0	Yanch	The public session should be opened with a statement on where SOARCA is conservative. This will give the public a better understanding of the thought processes and methodologies behind the analyses.	A guiding principal of the SOARCA analysis has been to avoid undue conservatisms and make every attempt to provide best estimate results. Nevertheless, there are a number of conservatisms that are still reflected in the SOARCA results. For example, the assumption of

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				mid-day population motion during a weekday to present the most challenging evacuation scenario. At the same time emergency response organizations are assumed to be staffed at nighttime levels.
70	0	Leaver	Assess the sensitivity on the time to declare a General Emergency (GE). Even if the sensitivity is low, that is valuable information. The sensitivity of health effects on the speed of declaring GE should also be measured. For example, a LOCA does not survive the screening process but could it have health effects?	Although there is high confidence in the current timing of the declaration of general emergency this parameter will be considered for assessment in the uncertainty quantification effort. In any event, only consequences associated with scenarios that pass the SOARCA screening procedure will be evaluated.
71	0	Canavan	The conclusions need to be documented better throughout the NUREG. Too much is left for the reader to interpret.	Previous comments from other reviewers have led to a number of changes to the SOARCA documentation. Many of these changes are intended to clarify the primary observations of the SOARCA analyses and make the discussion more coherent.
72	0	Gabor	With the Station Blackout conditions for the long term (transient), use different EALs and see effects. Try normal EALs, not the SBO EALs.	Although there is high confidence in the current timing of the declaration of general emergency this parameter will be considered for assessment in the uncertainty quantification effort. In any event, only consequences associated with scenarios that pass the SOARCA screening procedure will be evaluated.
73	0	Yanch	Calculate for different weather conditions as a sensitivity study. It is important to report the consequences of bounding weather conditions, along with the consequences of mean weather conditions. (Bixler 2nd pres slide 4)	The SOARCA project has explicitly focused on the mean consequence results associated with weather variability. Quoting consequence results associated with specific weather conditions would significantly complicate the communication of the SOARCA results and would be prone to misinterpretation.
74	0	Canavan	Pick a specific rainy day and a specific sunny day, since these days really happened, and analyze under these conditions. This can be used to justify the mean. (Bixler 2nd pres slide 4)	The SOARCA project has explicitly focused on the mean consequence results associated with weather variability. Quoting consequence results associated with specific weather conditions would

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				significantly complicate the communication of the SOARCA results and would be prone to misinterpretation. Furthermore, these specific weather conditions would not necessarily bracket the mean result.
75	0	Mrowca	The connectivity between thermal hydraulic consequences and risk is weak.	This comment refers to the description of the link between the thermal-hydraulic accident progression model and the off-site consequence analysis. Given the nature of the off-site consequence calculation it is not possible to describe this calculation in a manner equivalent to the scenario specific thermal-hydraulic calculation. To do so would require the selection of a specific weather scenario which would allow a more detailed description of plume motion, radionuclide deposition, etc. both spatially and temporally. Since the weather conditions at the time of a specific event cannot be assumed, the approach taken in the SOARCA analyses is to conduct many hundreds of calculations with different weather conditions. Reporting the mean, e.g., expected, result of this large set of trials is consistent with the "best estimate" objectives of the SOARCA project. Reporting the details of each of these weather trials in a way that is comparable to the thermal-hydraulic analysis however would be impractical.
76	0	O'Kula	MELMACCS is being relied upon to perform post-processing of MELCOR results to provide a set of deposition velocities for MACCS2 (page 64, paragraph 4). To understand this set of inputs, and the basis for their preparation, we would need to see a discussion/document on MELMACCS to describe its technical basis, and the inputs used to generate the sets of deposition velocities. In addition, a table is needed, if not in Volume I, then	The equations used in MELMACCS are also documented in Reference [48]. A table providing the specific deposition velocities used in the SOARCA analyses has been included in Section 5.4.

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			in Volume III (Peach Bottom) and Volume IV (Surry), on the input deposition velocities used for the MACCS2 analysis.	
77	0	Canavan	The SOARCA analysis and report is developed by applying a method to two specific plants Surry and Peach Bottom. The use of two specific plants has both positive and negative aspects. The positive aspects are that with plant specific information, plant specific conclusions can be drawn and can be based on the specific design features, maintenance and operation practices at that particular site. The downside to this approach is that not all the plant specific features, both those features that reduce consequences as well as those that might increase consequences, are represented in the two plants chosen. As such, some conclusions are likely applicable to that site only and the results may not be typical. While an alternative to the current approach or analysis is not recommended or sought by this comment a short discussion of the necessity of the approach as well as the benefits and potentials issues maybe warranted. In addition, sensitivity cases of known issues such as the Surry specific interfacing systems LOCAs may be warranted (cf. detailed comments submitted by Canavan 10/14/09 for examples)	Agreed, the choice of plant specific analysis for the SOARCA project has advantages and disadvantages. The choice of a plant specific approach is however consistent with the intent of the SOARCA results to reflect risk significant and historically important scenarios rather than a comprehensive evaluation of severe reactor accident risks. A number of sensitivity studies have already been performed and have been described in the revised documentation.
78	0	Canavan	In many locations in the report, the facts are provided in the appropriate level of detail. Often these facts represent specifically what was done in the analysis. What is not always presented is the conclusions that can be drawn from the facts provided or any alternative information that supports the conclusions that are drawn but not stated. The use of affirmative statement and/or any additional evidence that supports the	The executive summary has been enhanced including clarifying text in the offsite radiological consequences section as well as a detailed conclusions section.

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			conclusion could be helpful in some instances. (cf. detailed comments submitted by Canavan 10/14/09 for examples)	
79	0	Canavan	An important aspect of this type of analysis is to ensure that it is complete and all aspects and range of variables that can impact the consequences have been considered. During the detailed discussions and question and answer period with the authors, it was clear that analysis beyond what was documented in the current 4 volumes had been performed. These discussions and additional analysis, evidence or information should be documented in the reports. So as not to detract from some of the more important points of the analysis, appendices can be used. There are several specific areas which are noteworthy of further consideration, analysis or documentation. These are all in the larger category of completeness and are the treatment of security related events, the treatment of the accident sequence selection and application of the screening criteria and the external event scenarios.	A number of comments have already been made and addressed on the subject of the completeness of the scenarios considered in the SOARCA project. In some cases, these comments have resulted in additional sensitivity calculations that have since been documented along with the base case results. The treatment of security related events and beyond state-of-the-art treatment of seismic events are beyond the scope of the SOARCA project however.
80	0	Canavan	The impact of the sequence frequency truncations is significant on the outcome of the study. As the study is a consequence study, the specific frequency of occurrence of the scenario is not relevant except to choose the most frequent scenario groups to analyze. This is also not well described in Volume 1. At this time this reviewer is not suggesting that the truncation process is flawed, only that the text has begged a significant question that remained unanswered. As part of this reviewers tasks will be the attempt to provide any specific scenario groups that maybe missing from the scope of the SOARCA review. (cf.	Previous comments from other reviewers have led to a number of changes to the SOARCA documentation. Many of these changes are intended to clarify the primary observations of the SOARCA analyses and make the discussion more coherent.

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			detailed comments submitted by Canavan 10/14/09 for examples)	
81	0	Canavan	Based on this reviewers experience, there are a number of BWR related scenarios that, if they don't exceed the SOARCA screening criteria, may significantly overlap the criteria. They are presented here for consideration (cf. detailed comments submitted by Canavan 10/14/09 for greater detail) • LOCAs outside containment (estimated frequency 10^{-9}/RY to $5x10^{-7}$/RY) • Subsequent failure of RPS following a transient event (estimated frequency $1x10^{-7}$/RY to $3x10^{-7}$/RY) • Other containment bypass events (estimated frequency <10^{-6}/RY) • LOCAs with vapor suppression failure (estimated frequency (estimated frequency 10^{-8}/RY	These sequences are included here for reference purposes but no specific action has been taken as all of these sequences fall below the current SOARCA screening criteria. Additional text has been added to the executive summary to address the "completeness" of the SOARCA accident scenarios and the associated screening process.
82	0	Leaver	So as to make the frequency cutoff more robust and less of a black and white process, it would be prudent to examine an order of magnitude or so below the frequency cutoff to confirm that there are no sequences with consequences that might significantly exceed those already being considered in SOARCA or that might impact overall conclusions which are derived from the best-estimate, baseline sequences. To an extent, SOARCA has already done this by virtue of including Surry interfacing LOCA which came in at less than 10^{-7}, including Peach Bottom unmitigated STSBO which is less than 10^{-6}, including Peach Bottom Loss of Vital AC Bus E-12 which was less than 10^{-6}, and including the unmitigated sequences which when quantified even in a conservative manner should drop below	As the reviewer states, scenarios have already been included in the SOARCA analysis that fall below the formal screening criteria but have the potential for yielding larger or earlier environmental releases. In addition, a number of sensitivity calculations have been conducted that effectively constitute scenarios that may also have lower frequencies than the screening criteria. As stated elsewhere, the objective of the SOARCA project is to evaluate the impact of modern analysis methods, phenomelogical understanding, and plant procedures on the analysis of accident sequences that represent a significant fraction of overall reactor accident risk. Additional text has been included to address the "completeness" of the SOARCA scenarios as well as to discuss special classes of scenarios not considered by the

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			the cutoff. But it needs to be documented and presented in the report as part of, or a backup to, the screening process.	SOARCA project.
83	0	Leaver	Of the event types that were not addressed in the draft report, the most important is security events, particularly airplane crash. A study such as SOARCA will lose credibility and impact if it is silent on this. It is recognized that for confidentiality reasons, there is limited information that can be presented on security events; plus it may only be possible to characterize probability in a qualitative manner. But there is much that could be said about what the Commission has done to address these events, and the limited consequences which are expected (e.g., no more significant than the sequences that are analyzed explicitly in SOARCA). (cf. item 42)	The SOARCA project originated in insights obtained through recent NRC analyses of security related events. The "mitigated" SOARCA analyses also credit security related measures recently implemented under 10CFR50.54(hh). Nevertheless, security related scenarios are explicitly excluded from the SOARCA analysis specifically for classification reasons.
84	0	Leaver	There are no mitigated STSBO sequences (i.e., no STSBO sequences with 10CFR50.54(hh) measures considered). What is the reason for this? Apparently Peach Bottom had not yet procured the required portable equipment as of the time of the site visit, yet the 10CFR50.54(hh) portable pump is credited in the Peach Bottom mitigated LTSBO (see Volume III, Table 4). For STSBO without RCIC blackstart, RPV pressure is less than 100 psi after about 4 hours, and lower head failure does not occur until about 8 hours. For STSBO with RCIC blackstart, these times are even longer. It would appear that there is time to put the portable pump in place to achieve a benefit, possibly preventing lower head failure, or at least delaying lower head failure, and also reducing radionuclide release. (cf. detailed	A separate MELCOR calculation for the mitigated STSBO was not performed because mitigation would have had the same result as the LTSBO scenario, i.e, no core damage.

#	Rev.	Reviewer	Comment	Resolution
			comments by Leaver 10/5/09 for frequency estimates)	
85	0	Leaver	For the same reasons as described in my August 5, 2009 Comment 5, some reasonable probability should be assigned to operator failure to implement the 50.54(hh) mitigative measures. If a factor of 10 is assumed as was done in the August 5, 2009 Comment 5, the unmitigated STSBO sequences (two of them) probabilities would decrease to 1E-8 – 5E-8, and the mitigated STSBO sequences (if they were added to the analysis) would be 1E-7 – 5E-7. (cf. detailed comments by Leaver 10/5/09 for frequency estimates)	The inclusion of both mitigated and unmitigated results is an important feature of the SOARCA results. Excluding the mitigated results would be to err on the side of conservatism while excluding the unmitigated results would be to err on the side of non-conservatism. While this observation has merit, an assessment of the impact of the 10CFR50.54(hh) measures on the scenario frequency would require a risk and human reliability study beyond the scope of the SOARCA project. The executive summary has been enhanced to emphasize that the probability of 10CFR50.54(hh) mitigation is assumed to be zero for the purposes of the SOARCA analysis of the unmitigated cases.
86	0	Leaver	If the Peach Bottom mitigated STSBO sequences are considered, the unmitigated STSBO sequences would then become sensitivities, ... (cf. detailed comments by Leaver 10/5/09 for frequency estimates)	See resolution to items 84 and 85.
87	0	Leaver	The Loss of Vital AC Bus E-12 sensitivity for operator failure to manually depressurize and failure to open CRDHS throttle valve has core damage, but there is no radioactive release analysis. (cf. detailed comments by Leaver 10/5/09 for frequency estimates)	Although one sensitivity considered for this scenario led to core damage, there was no resulting vessel failure or release. Beyond this sensitivity calculation, the best estimate for this scenario was that core damage would be averted without the use of 10CFR50.54(hh) related equipment so no off-site consequence assessment was performed.
88	0	Leaver	If the sensitivity for Loss of Vital AC Bus E-12 with operator failure to manually depressurize and failure to open CRDHS throttle valve is included,	Closer examination of the frequency of this event determined that it fell below the screening criteria. It was included in the SOARCA documentation

#	Rev.	Reviewer	Comment	Resolution
			a probability should be estimated. The frequency would likely be an order of magnitude or more below the <1E-6 number that is given in the report for the base case. (cf. detailed comments by Leaver 10/5/09 for frequency estimates)	due to the useful insights it provided regarding the performance of low capacity safety equipment in mitigating events.
89	0	Leaver	In Volumes III and IV, Sections 6 (EP) and 7 (Consequences), it appears that the unmitigated sequences are given undue emphasis. For Volume III (Peach Bottom), per Table 9 all 3 of the scenarios assessed for emergency response are unmitigated. For Volume IV (Surry), per Table 15 4 out of the 5 scenarios assessed for emergency response are unmitigated. Emergency response and consequence analysis of *unmitigated sequences is appropriate as a* sensitivity, but why not have a best-estimate, base case which uses sequences that survive the screen? Based on the August 5, 2009 Comment 5 table, there are two such Surry sequences with a non-zero release (mitigated STSBO and mitigated STSBO with induced SGTR). There may not be any non-zero release sequences for Peach Bottom that survive the screen, but the next closest sequence could be considered (either the unmitigated LTSBO or the mitigated STSBO) for the base case so as to have a Peach Bottom release for the best-estimate, base case consequence and emergency response analysis.	It was the determination of the SOARCA analysis team that without a detailed PRA/HRA assessment of the 10CFR50.54(hh) procedures it is not possible to evaluate the influence these procedures may have on the underlying frequency used to identify the sequence. Therefore both the mitigated and unmitigated scenarios were evaluated both to avoid undue conservatism and to allow for more effective comparison to previous studies to evaluate the impact of modern analysis capabilities.
90	0	Yanch	Explain why the RBE for bone marrow is reduced to 1.	The text in revision 0 was incorrect. The reduction in biological effectiveness for both bone marrow and breast tissue was recommended in

#	Rev.	Reviewer	Comment	Resolution
				Federal Guidance Report 13 (cf. pg. 174).

Notes

2006 Peach Bottom

591 hours calm 31 hours bad data

99.9 removed	99.9 removed	99.9 removed
77.7 → 0	77.7 → 0.5	77.7 deleted
sum 41066.8	sum 41362.8	sum 41144.5
count 8729	count 8729	count 8138
avg 4.7046	avg 4.739	avg 5.0558
avg x .447 2.103	avg x .447 2.118	avg x .447 2.260

Nate's values on MACCS2 file avg = 2.12 corrected to 0.5 m/sec min = 2.14

2005 Peach Bottom

82 hours calm 214 hours bad data

99.9 removed	99.9 removed	99.9 removed
77.7 → 0	77.7 → 0.5	77.7 deleted
sum 41692.6	sum 41733.6	sum 41692.6
count 8556	count 8556	count 8474
avg 4.8729	avg 4.8777	avg 4.9201
avg x .447 2.178	avg x .447 2.180	avg x .447 2.199

Peach Bottom comment –

2-year data shot 2.17 to 3.05

Nate's values on MACCS2 file avg = 2.17 corrected to 0.5 m/sec min = 2.17

STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES PEER REVIEW PLAN

August 6, 2008
Draft

1. BACKGROUND

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC). More recently, with Commission guidance and as part of plant security assessments, the staff concentrated on applying the accumulated research to perform considerably more detailed, integrated, and realistic analyses of severe accident progression and consequences. The results of these recent studies confirmed and quantified that some past studies of plant response and offsite consequences could be extremely conservative, to the point that predictions were not useful for characterizing results or guiding public policy. In some cases, overly conservative results were driven by the combination of conservative assumptions or boundary conditions. In other cases, simple bounding analyses were used in the belief that if the result was adequate to meet an overall risk goal, bounding estimates of consequences could be tolerated. The subsequent misuse or misinterpretation of such bounding estimates indicated that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes.

The State-of-the-Art Reactor Consequence Analyses (SOARCA) project is currently being developed by the NRC to create a body of knowledge regarding the likely outcomes of severe reactor accidents, based on the most current emergency preparedness (EP) and plant capabilities. Towards this objective, it is being used to realistically evaluate important accident scenarios that could potentially release radioactive materials to the environment and to provide a more accurate assessment of potential offsite consequences from these scenarios, given the new understanding of plant performance under accident conditions. The project focus is to perform consequence analyses for those internally and externally initiated accident sequences estimated to have a core damage frequency (CDF) approximately equal to or greater than 10^{-6} per reactor-year (greater than 10^{-7} per reactor-year for containment bypass events). The NRC also reviewed scenarios with CDF less than 10^{-6} per reactor-year to show that such scenarios would not result in an accident of significantly higher consequence or an accident that has not previously been analyzed. The NRC is using state-of-the-art information and analysis tools (MELCOR) to develop best estimates of the time to core heat up, degradation, fission product release magnitude and timing, and reactor vessel and containment performance to realistically calculate the radioactive material released into the environment. The NRC is using the MELCOR Accident Consequence Code System, Version 2 (MACCS2) to develop site-specific estimates of the potential consequences to the public that account for site-specific weather conditions, population distribution, and EP assumptions.

The staff expects that the results of the reanalysis of severe accident consequences via SOARCA would provide the foundation for communicating that aspect of nuclear safety to Federal, State and Local authorities, licensees, and the general public. The reanalysis would also update earlier site-specific quantifications of offsite consequences such as the 1982 Sandia Siting Study (NUREG/CR-2239).

Analyses have been planned for one plant of each of the following major plant types: General Electric with a Mark I containment, General Electric with a Mark II containment, General Electric with a Mark III containment, Westinghouse with a large dry containment, Westinghouse with an ice condenser containment, Combustion Engineering, and Babcock and Wilcox. The analyses for a General Electric with a Mark I containment (Peach Bottom) and a Westinghouse with a large dry containment (Surry) are nearing completion. The analyses for other plant types are just beginning or have not begun.

The NRC is initiating an independent peer review of the SOARCA approach and results obtained for the Peach Bottom and Surry plants.

2. OBJECTIVE

The objective of the peer review for the SOARCA project is to have independent scientific and technical experts review the approach and underlying assumptions and results obtained for Peach Bottom and Surry to ensure that they are defensible and represent the state-of-the-art. A peer review is necessary because the SOARCA project is based on state-of-the-art and, in some areas, novel methods; presents complex challenges for interpretation; contains precedent-setting methods and models; and presents conclusions that are likely to change prevailing practices.

3. SCOPE OF WORK

At the start of the peer review, the NRC will provide the peer review panel with the documentation of the Peach Bottom and Surry analysis. In particular, the NRC will provide the peer review panel with two reports, the Integrated Analysis Report for Peach Bottom and the Integrated Analysis Report for Surry. These reports will describe the following:

- sequence grouping and sequence selection, including internal and external events
- mitigation measures assessment
- accident progression and radiological release analysis
- offsite radiological consequence analysis, including analytical treatment of site-specific evacuation and relocation and health effects modeling

(Note: Per the 189, the last two reports are being prepared by Sandia. However, RES staff needs to write and provide to Sandia the sections on sequence grouping and sequence selection and mitigation measures assessment because this work was done in-house at the NRC.)

The peer review will include a series of three meetings of the peer review panel. These meetings are intended to accomplish the following: NRC staff and NRC-contractor staff present to the peer review panel the SOARCA methods and results for Peach Bottom and Surry, peer reviewers discuss issues, NRC and NRC-contractor staff help clarify issues as needed, and peer reviewers develop findings and recommendations. NRC also may request that the peer review panel present findings and recommendations at a meeting in the Rockville area, such as at an ACRS meeting.

The peer review panel will be requested to provide comments on the overall approach to SOARCA, the assumptions made, and the technical basis that supports the overall conclusions. SOARCA is intended to be a state-of-the-art reactor consequence analysis for risk-important sequences. Because SOARCA is not intended to be an overall risk assessment, the peer review panel will be requested to address the following questions:

- Is SOARCA's use of conditional risk adequate?
- Is not reporting consequences from extremely unlikely weather adequate?
- For station blackout scenarios, is it appropriate to do the SOARCA analysis with and without portable equipment, instead of performing a detailed HRA analysis?
- For spontaneous steam generator tube rupture and interfacing system LOCA scenarios which involve operator errors, is it appropriate to do the SOARCA analysis with and without operators eventually correcting their errors, instead of performing a detailed HRA analysis?
- In the SOARCA sequences with portable equipment, has this portable equipment been appropriately credited?
- Is SOARCA's limited treatment of uncertainties adequate?
- SOARCA includes input analysis form each level of a PRA (i.e., Level 1, Level 2, and Level 3). However, it is not a full blown level 3 PRA. What are the pitfalls, if any, of SOARCA's use of such an approach?

The following additional questions will be provided to the peer review panel to help guide the review:

- To what degree does the SOARCA project reflect the current state-of-the-art in PRA including consideration of both internal and external initiators and mitigation measures such as portable power supplies and portable pumps?
- To what degree does the SOARCA project represent the current state-of-the-art in accident progression, radiological release, and offsite consequences? Are the MELCOR and MACCS codes adequate for analyzing the sequences evaluated in SOARCA?
- Has the SOARCA study correctly identified conservatisms and non-conservatisms in the accident progression, radiological release, and offsite consequence analyses? What other conservatisms and non-conservatisms are there?
- Has the SOARCA study correctly identified uncertainties in the accident progression, radiological release, and offsite consequence analyses? What other uncertainties are there?
- Are there data or analyses that can shed light on the significance of some of the identified conservatisms, non-conservatisms, and uncertainties?

- The original objective of SOARCA was to examine significant radiological release scenarios having estimated release frequencies greater than 10^{-6}/year with consideration given to lower frequency events with much higher consequences. The intent was to focus attention on the scenarios of greatest interest and provide insights into the effectiveness of current and postulated mitigation strategies. Are use of CDFs of 10^{-6}/year for containment failure events and 10^{-7}/year for containment bypass events reasonable surrogates for this release frequency?
- In the SOARCA approach, individual sequences from plant-specific Level 1 PRAs are grouped and their CDFs are summed to estimate a sequence group CDF. The sequence group CDF is compared with a CDF of 10^{-6}/year for containment failure events and a CDF of 10^{-7}/year for containment bypass events. Offsite radiological consequences are then estimated for sequence groups with a higher frequency. Has use of these CDFs inadvertently screened out risk-important sequences from being analyzed in SOARCA? If so, how much lower CDFs should be used?
- Have the individual sequences been grouped in a best-estimate fashion, or have significant conservatisms or non-conservatisms been introduced?
- Is the SOARCA approach for reporting latent cancer fatality consequences (individual probability of latent cancer fatality for a person in the 0-10 mile zone, for a person in the 0-50 mile zone, and for a person in the 0-100 mile zone) helpful in explaining severe accident consequences to the range of stakeholders or is another approach recommended?
- Is the SOARCA approach to low dose health effects (LNT and no latent cancer fatalities from doses less than 10 mrem/year) reasonable for the SOARCA project, which is a best-estimate analysis, given uncertainties in low dose health effects modeling?
- Are the SOARCA reports well-written, well-organized, and understandable? Have the goals and objectives of SOARCA been clearly described in the SOARCA reports? Have the range of applicability and limitations of SOARCA been clearly described in the SOARCA reports?

Each panel member shall provide a draft report which includes an evaluation for the topics and focus areas listed above in his area(s) of expertise. Following discussions of findings by the individual panel members, the panel shall assemble a final report that addresses the technical findings of all the panel members.

Because this will be a non-FACA[1] review, no attempt will be made to develop a consensus report. Instead, each committee member will present his own individual viewpoint and

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[1] FACA = Federal Advisory Committee Act. The Federal Advisory Committee Act was enacted in 1972 to ensure that advice by the various advisory committees formed over the years is objective and accessible to the public. The Act formalized a process for establishing, operating, overseeing, and terminating these advisory bodies and created the Committee Management Secretariat to monitor compliance with the Act.

recommendations. The final report shall identify areas where a consensus exists among the panel members, and specify areas where differences of opinion exist among the panel members.

4. PANEL MEMBERSHIP

Areas of expertise that will be in the SOARCA peer review panel are the following: PRA, accident progression and radiological release, offsite radiological consequences, and emergency preparedness. A list of potential peer review panel members is attached.

5. MEETINGS AND TRAVEL REQUIREMENTS

Each panel member shall attend three working meetings which may be held at various locations such as the NRC, Sandia National Laboratories, and near the Peach Bottom and Surry sites. In addition, each panel member shall attend a meeting at the NRC offices in Rockville, MD, to present the panel's findings.

6. SCHEDULE AND DELIVERABLES

Milestone or Activity	Estimated date*
Peach Bottom and Surry analysis complete	September 2008
Peach Bottom and Surry documentation complete	November 2008
Commission makes Peach Bottom and Surry reports publicly available	November 2008
Peer review panel meetings (3 meetings)	December 2008 through May 2009
Draft reports by individual peer reviewers	January 2008
Draft peer review panel report	March 2009
Final peer review panel report	May 2009
Present peer review findings and recommendations	To be determined

*Assumes Commission makes Peach Bottom and Surry reports publicly available in November 2008.

7. COST ESTIMATE

The estimated level of effort for the peer review is 12 staff-months, including both peer review and SNL staff effort.

Potential Peer Review Panel Members

1. Chairman of Peer Review Committee (Reactor Safety)

Brent Boyack (LANL)

2. PRA (Sequence Selection and Mitigative Measures)

Mohammad Modarres (U. of Maryland)

Bruce Morowca (ISL)

3. Accident Progression and Radiological Release

Robert E. Henry (Fauske & Associates)

M. Khatib-Rahbar (ERI)

Neil Todreas (MIT)

4. Offsite Radiological Consequences

Kevin O'Kula, Washington Group International (WSMS, Aiken, SC)

Dave Leaver (Polestar)

5. Emergency Preparedness

Steven Hook, Emergency Preparedness Expert (Contingency Management Consulting, LLC)