

## 15.0 ACCIDENT ANALYSIS

The evaluation of the safety of a nuclear power plant includes analyses of the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the combined license (COL) reviews. In Chapter 15 of the Final Safety Analysis Report (FSAR), the COL applicant discussed the applicable transient and accident analyses to justify its conformance to the applicable regulations.

The Nuclear Regulatory Commission (NRC) staff's review of V.C. Summer Nuclear Station (VCSNS) COL FSAR Chapter 15 follows the format in VCSNS Chapter 15.

### **15.0 Accident Analysis (Related to Regulatory Guide (RG) 1.206, Section C.III.1, Chapter 15, C.I.15.1, "Transient and Accident Classification," C.I.15.2, "Frequency of Occurrence," C.I.15.3, "Plant Characteristics Considered in the Safety Evaluation," C.I.15.4, "Assumed Protection System Actions," and C.I.15.5, "Evaluation of Individual Initiating Events")**

#### **15.0.1 Introduction**

Design basis transient and accident analyses are required as a part of an evaluation of the safety of a nuclear power plant by analyzing the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. The safety analyses provide a significant contribution to the determination of limiting conditions for operation, limiting safety system settings, and design specifications for plant components and systems to protect public health and safety.

#### **15.0.2 Summary of Application**

Section 15.0 of the VCSNS COL FSAR, Revision 2, incorporates by reference Section 15.0 of the AP1000 Design Control Document (DCD), Revision 17.

#### **AP1000 COL Information Item**

- STD COL 15.0-1

In letters dated August 25, 2010, and November 8, 2010, the applicant endorsed Vogtle Electric Generating Plant (VEGP) letters dated May 21, 2010, and October 29, 2010, respectively. In these letters, the applicant proposed Standard (STD) COL 15.0-1, adding new text to VCSNS COL FSAR Section 15.0. STD COL 15.0-1 was provided in a response to a request for additional information (RAI) related to the AP1000 design certification (DC) amendment review. Specifically, in its response dated May 6, 2009, to NRC RAI AP1000 DCD RAI-SRP15.0-SRSB-02, Westinghouse proposed COL Information Item 15.0-1 to provide documentation of the plant calorimetric uncertainty methodology. RAI-SRP15.0-SRSB-02 noted that the AP1000 DCD assumes a 2 percent power uncertainty for the initial condition for most

transients and accidents. However, a 1 percent power uncertainty is assumed for the initial reactor power for the large-break loss-of-coolant accident (LOCA) in AP1000 DCD Section 15.6.5.4A, as well as the mass and energy release calculation in AP1000 DCD Sections 6.2.1.3 and 6.2.1.4. In response to this RAI, Westinghouse proposed a new COL information item to be included in a future revision to AP1000 DCD Section 15.0.15. COL Information Item 15.0-1 states:

Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters prior to fuel load, the Combined License holder will calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

#### License Conditions

- License Condition 2, Item 15.0-1

In a letter dated August 25, 2010, the applicant endorsed the VEGP letter dated May 21, 2010, that proposed adding Item 15.0-1 to License Condition 2, which would confirm that the plant-operating instrumentation installed for feedwater flow measurement is a Caldon/Cameron Leading Edge Flow Meter (LEFM) CheckPlus™ system. In its letter dated November 8, 2010, the applicant endorsed the VEGP letter dated October 29, 2010, letter that revised Item 15.0-1 to state that the documentation of plant calorimetric uncertainty methodology would be addressed as a plant-specific inspections, tests, analyses and acceptance criteria (ITAAC) item in lieu of License Condition 2.

- License Condition 6

In its letter dated November 8, 2010, the applicant endorsed the VEGP letter dated October 29, 2010, that proposed adding new line items to proposed License Condition 6, associated with the power calorimetric uncertainty instrumentation.

#### Inspections, Tests, Analyses and Acceptance Criteria

In its letter dated November 8, 2010, the applicant endorsed the VEGP letter dated October 29, 2010, that proposed ITAAC associated with the plant calorimetric uncertainty methodology.

### **15.0.3 Regulatory Basis**

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," and its supplements.

The need to address the calorimetric power uncertainty is found in Section 15.0 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for

Nuclear Power Plants.” Specifically, NUREG-0800 Section 15.0, Section I.3, “Plant Characteristics in the Safety Evaluation,” states in part that “the reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.” For the LOCA analysis, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic licensing of production and utilization facilities,” Appendix K, ECCS [Emergency Core Cooling System] Evaluation Models,” specifies that an assumed power level lower than 1.02 times the licensed power level may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

#### 15.0.4 Technical Evaluation

The NRC staff reviewed Section 15.0 of the VCSNS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.<sup>1</sup> The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to accident analysis. The results of the NRC staff’s evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (VEGP, Units 3 and 4) were equally applicable to the VCSNS Units 2 and 3 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 2, to the VCSNS COL FSAR. In performing this comparison, the staff considered changes made to the VCSNS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the VCSNS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

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<sup>1</sup> See Section 1.2.2 for a discussion of the staff’s review related to verification of the scope of information to be included in a COL application that references a DC.

The following portion of this technical evaluation section is reproduced from Section 15.0.4 of the VEGP SER:

AP1000 COL Information Item

- STD COL 15.0-1

*In a letter dated May 21, 2010, as revised by a letter dated October 29, 2010, the VEGP applicant submitted information to address COL Information Item 15.0-1. In these letters, the applicant stated that the plant operating instrumentation for feedwater flow measurement would be the Caldon/Cameron LEFM CheckPlus™ system and referenced the NRC staff's final safety evaluation that approved the Caldon topical report, ER-157P, Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or Checkplus™ System." The NRC staff has previously approved several plant applications of the Caldon/Cameron CheckPlus™ LEFM system to support a power measurement uncertainty lower than 1 percent. This AP1000 COL information item supports the 1 percent power uncertainty. The NRC staff's review herein focused on ensuring that the generically approved Caldon/Cameron topical reports are properly implemented for the VEGP COL application. The NRC staff verified compliance with the applicable conditions in the NRC staff's safety evaluations approving the topical reports. The NRC staff's review also confirmed that appropriate license conditions and ITAAC were established for those items that cannot be resolved prior to issuance of the COL.*

Compliance with Caldon/Cameron Topical Report ER-80P

*NRC staff approval of the Caldon/Cameron topical report ER-80P (safety evaluation (SE) dated March 8, 1999) established four criteria to be satisfied by each applicant or licensee. The VEGP applicant addressed each criterion as described below.*

Criterion 1

*Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.*

*The VEGP applicant stated that calibration and maintenance programs would be developed in accordance with the Caldon/Cameron LEFM technical manuals and recommendations. Preventative Maintenance (PM) tasks would be periodically performed within the plant control system and support systems to provide continued reliability. Plant instrumentations that affect the power calorimetric, including the Caldon/Cameron LEFM CheckPlus™ inputs, would be monitored by plant system engineering personnel. These instruments would be included in the*

*plant PM program for periodic calibration. The NRC staff finds these measures acceptable.*

*The VEGP applicant stated when the Caldon/Cameron LEFM CheckPlus™ flow meter becomes inoperable beyond the allowed outage time; the plant would be operated at de-rated conditions. De-rated operation is appropriate at power levels consistent with a 2 percent power uncertainty. With the plant operating at 100 percent load with 1 percent uncertainty, a de-rating to 99 percent maintains a 2 percent uncertainty. When the LEFM CheckPlus™ is inoperable, plant calorimetric power would be monitored with the use of feedwater venturi elements. An inoperable LEFM would not leave the plant in a condition where steady-state operation would be immediately compromised since it would not directly impact the calibration of the nuclear instrumentation utilized for power level related trips or safety system actuations. Thus, procedures require confirmation of the availability of alternate instrumentation (i.e., the feedwater venturi instrumentation) and initiation of the above described reduction in power within 48 hours. These measures are consistent with the operating plants. The NRC staff finds that operation with an inoperable Caldon/Cameron CheckPlus™ has been acceptably addressed.*

#### Criterion 2

*For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analyses and assumptions set forth in TR ER-80P.*

*The VEGP applicant stated that, since this application represents construction of a new plant with no previously installed LEFM equipment, this item is not applicable. The NRC staff finds the VEGP applicant's response acceptable.*

#### Criterion 3

*Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.*

*The VEGP applicant stated that the uncertainty of the LEFM would be calculated in accordance with the Westinghouse methodology as applied in the Beaver Valley Power Station Units 1 and 2 License Amendment Request Nos. 289 and 161, which was approved by the NRC staff in a letter dated September 24, 2001, titled, "Beaver Valley Power Station, Unit Nos. 1 and 2*

*(BVPS-1 and 2) – Issuance of Amendment Re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves.” The NRC staff reviewed this SE and found that the calculation methodology complies with the recommendations of American National Standards Institute/Independent Safety Assessment (ANSI/ISA) Standard 67.04-2000, "Setpoints for Nuclear Safety-Related Instrumentation," and Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," Revision 2. In these calculations, uncertainties for the parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined. Then, all independent parameters/groups that contribute to the power measurement uncertainty are combined using a square root of sum of squares (SRSS) approach to determine the overall power measurement uncertainty. This methodology has been reviewed and approved by the NRC staff for Westinghouse pressurized-water reactors (PWRs) (e.g., Beaver Valley), and is also acceptable for AP1000, which is a Westinghouse-designed PWR. The staff finds the AP1000 design sufficiently similar to other Westinghouse PWR designs that have been approved such that the methodology applies to both designs. Therefore, the NRC staff finds that the VEGP applicant's response acceptable.*

#### Criterion 4

*Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.*

*The VEGP applicant stated that its application represents construction of a new plant with no previously installed flow metering equipment. The AP1000 main feedwater flow measurement instrumentation, consistent with the use of normalized flow meters, would be required to be calibrated at a certified test laboratory in hydraulic model geometry consistent with the AP1000 plant design. The LEFM commissioning process (i.e., installation acceptance testing) would confirm that the actual instrument performance is consistent with the assumptions of the uncertainty calculation. The NRC staff finds this response acceptable.*

Compliance with Caldon/Cameron Topical Report ER-157P, Revision 8

The VEGP applicant addressed the five SE conditions found in the NRC SE for ER-157P, Revision 8, dated August 16, 2010, as described below.

Condition 1

*Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.*

*The VEGP applicant stated that a failure of the ultrasonic flow meter (UFM) will result in the use of the feedwater venturi as the input into the calorimetric calculation. Since the contingency is not based on continued reliance on the CheckPlus™ system, the NRC staff finds the VEGP applicant's response acceptable.*

Condition 2

*A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.*

*The VEGP applicant stated that a degraded UFM resulting in an instrument uncertainty greater than the values assumed in the AP1000 calorimetric uncertainty calculation would be considered a failure and subject to compensatory actions as discussed above in response to Caldon/Cameron topical report (ER-80P) Criterion 1. Since the applicant does not intend to operate using a degraded CheckPlus™, the NRC staff finds the VEGP applicant's response acceptable.*

Condition 3

*An applicant with a comparable geometry can reference the above Section 3.2.1 [of the SE for ER-157P] finding to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.*

*The VEGP applicant stated that the AP1000 feedwater flow measurement instrumentation would be located in piping with downstream geometry more favorable than the arrangements referenced in Section 3.2.1 of the SE for ER-157P. Therefore, the effects of downstream piping geometry are not considered to have a significant influence on the accuracy of the UFM. Because the flow measurement instrumentation would be located in piping with favorable downstream geometry, the NRC staff finds the VEGP applicant's response acceptable.*

Condition 4

*An applicant that requests a MUR [measurement uncertainty recapture] with the upstream flow straightener configuration discussed in Section 3.2.2 [of the SE for ER-157P] should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 [Letter from E. Hauser dated March 19, 2010]. Since the Reference 17 evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.*

*The VEGP applicant stated that the AP1000 UFM installation would not utilize an upstream flow straightener. Therefore, this condition is not applicable to the AP1000 design. The NRC staff finds the VEGP applicant's response acceptable.*

Condition 5

*An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18 [Letter from E. Hauser dated March 18, 2010].*

*The VEGP applicant stated that this AP1000 application of the CheckPlus™ LFM is to support a 1 percent overall power uncertainty, as compared to lower than 0.5 percent typically justified for operating plants using CheckPlus™. The result of this application of the LFM at a higher uncertainty (i.e., lower accuracy) is that the assumed steam separator/dryer performance becomes less of a relative contribution to the overall uncertainty. Furthermore, an engineering basis for the AP1000 moisture content assumption is in the calorimetric uncertainty calculation. Because the steam separator/dryer performance uncertainty is a relatively small contribution to the overall uncertainty of 1 percent, the NRC staff finds the VEGP applicant's response acceptable.*

*Based on its review of the VEGP applicant's responses, the NRC staff finds that the licensee has acceptably addressed all applicable conditions specified in the*



*NRC staff's SEs for the Caldon/Cameron topical reports. Hence, the NRC staff finds that the Caldon/Cameron topical reports, ER-80P and ER-157P, are acceptable for referencing in the VEGP COL application and that the applicant has adequately addressed COL Information Item 15.0-1.*

License Conditions

- *License Condition 2, Item 15.0-1*

*In a letter dated May 21, 2010, the applicant proposed adding Item 15.0-1 to License Condition 2 that would confirm that the plant operating instrumentation installed for feedwater flow measurement is a Caldon/Cameron LEFM CheckPlus™ system. In its October 29, 2010, letter, the applicant revised Item 15.0-1 to state that the documentation of plant calorimetric uncertainty methodology would be addressed as a plant-specific ITAAC item in lieu of License Condition 2. The staff finds the use of ITAAC to confirm proper documentation of plant calorimetric uncertainty methodology to be acceptable. The plant-specific ITAAC item proposed by the applicant is evaluated below.*

- *License Condition 6*

*In a letter dated October 29, 2010, the applicant proposed adding new line items to proposed License Condition 6, associated with the power calorimetric uncertainty instrumentation. Specifically, the applicant proposed to add the following two items:*

- *The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (prior to initial fuel load).*
- *The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (prior to initial fuel load).*

*The two items under License Condition 6 are needed because documentation for the actual instrument uncertainties would only be available after the equipment is procured and tested and administrative controls would not be available until after the equipment is procured, which would be after the COL license is issued. The staff finds the first item acceptable because, when combined with the methodology in the proposed ITAAC, it would allow the staff to confirm that the procured equipment results in a power uncertainty of no more than 1 percent prior to the start of plant operation. The staff finds the second item acceptable because it would allow the staff to confirm that the administrative controls are in place to meet ER-80P Criterion 1 prior to the start of plant operation. These items correspond to License Condition 15-1 in the following section.*

Inspections, Tests, Analyses and Acceptance Criteria

*In a letter dated October 29, 2010, the applicant proposed ITAAC associated with the plant calorimetric uncertainty methodology. The proposed ITAAC item is repeated in Table 15.0-1 of this SER. This ITAAC would confirm that: (1) the installed feedwater flow measurement device is the Caldon CheckPlus™ LEFM; (2) the power calorimetric uncertainty calculation for that instrumentation is based on an acceptable Westinghouse methodology as described above in Criterion 3 for ER-80P and the uncertainty values in the calculation for that instrumentation are not lower than those for the actual installed instrumentation; and (3) the calculated calorimetric power uncertainty measurement values are bounded by the 1 percent uncertainty value assumed for the initial reactor power in the safety analysis. The proposed ITAAC will allow the NRC staff to confirm, prior to initial fuel load, that the necessary conditions for STD COL 15.0-1 (COL Information Item 15.0-1) have been satisfied. Therefore, the NRC staff finds the proposed ITAAC acceptable.*

*The incorporation of the planned changes to the VEGP COL FSAR detailed in the applicant's letters dated May 21, 2010, and October 29, 2010 will be tracked as **Confirmatory Item 15.0-1**.*

**15.0.5 Post Combined License Activities**

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following ITAAC:

- The licensee shall perform and satisfy the plant calorimetric uncertainty and plant instrumentation performance analysis ITAAC defined in SER Table 15.0-1, "Power Calorimetric Uncertainty Methodology."

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition:

- License Condition (15-1) - The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections. The schedule shall be updated every six months until 12 months before scheduled fuel loading, and every month thereafter until the license condition has been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall address:
  - The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (prior to initial fuel load).
  - The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (prior to initial fuel load).

### 15.0.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to accident analysis and there is no outstanding information expected to be addressed in the VCSNS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that, pending closure of **Confirmatory Item 15.0-1**, the relevant information presented in the VCSNS COL application is acceptable and meets the NRC regulations. The staff based its conclusion on the following:

- STD COL 15.0-1 is acceptable because the applicant has demonstrated that the conditions identified by the NRC in its generic evaluation have been satisfied for the use of the Caldon/Cameron LEFM CheckPlus™ system for VCSNS Units 2 and 3. In addition, ITAAC and a license condition have been put in place to allow the staff to verify the plant calorimetric uncertainty methodology prior to initial fuel load.

### **15.1 Increase in Heat Removal from the Primary System (Related to RG 1.206, Section C.III.1, Chapter 15, C.I.15.6, "Event Evaluation")**

Analyses focused on the increase in heat removal from the primary system address anticipated operational occurrences (AOOs) and accidents that increase the heat removal by the secondary system, which could result in a decrease in reactor coolant temperature. Increased heat removal can be caused by:

- Feedwater system malfunctions causing a reduction in feedwater temperature
- Feedwater system malfunctions causing an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal heat exchanger

Section 15.1 of the VCSNS COL FSAR, Revision **2**, incorporates by reference, with no departures or supplements, Section 15.1, "Increase in Heat Removal from the Primary System," of Revision **17** of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

### **15.2 Decrease in Heat Removal By the Secondary System**

Analyses focused on the decrease in heat removal by the secondary system address AOOs and accidents that could result in a reduction of the capacity of the secondary system to remove

heat generated in the reactor coolant system (RCS). Decreased heat removal can be caused by:

- Steam pressure regulator malfunction or failure that results in decreasing steam flow
- Loss of external electrical load
- Turbine trip
- Inadvertent closure of main steam isolation valves
- Loss of condenser vacuum and other events resulting in turbine trip
- Loss of alternating current (ac) power to station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break

Section 15.2 of the VCSNS COL FSAR, Revision 2, incorporates by reference, with no departures or supplements, Section 15.2, “Decrease in Heat Removal by the Secondary System,” of Revision 17 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

### **15.3 Decrease in Reactor Coolant System Flow Rate**

Analyses focused on the decrease in RCS flow rate address AOOs and accidents that could result in a decrease in the RCS flow rate. Decreased flow rate can be caused by:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump (RCP) shaft seizure (locked motor)
- RCP shaft break

Section 15.3 of the VCSNS COL FSAR, Revision 2, incorporates by reference, with no departures or supplements, Section 15.3, “Decrease in Reactor Coolant System Flow Rate,” of Revision 17 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

### **15.4 Reactivity and Power Distribution Anomalies**

#### **15.4.1 Introduction**

Analyses focused on reactivity and power distribution anomalies address AOOs and accidents that could result in anomalies in the reactivity or power distribution in the reactor core. Reactivity and power distribution anomalies can be caused by:

- Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition
- Uncontrolled RCCA bank withdrawal at power
- RCCA misalignment
- Startup of an inactive RCP at an incorrect temperature
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant
- Inadvertent loading and operation of a fuel assembly in an improper position
- Spectrum of RCCA ejection accidents

#### **15.4.2 Summary of Application**

Section 15.4 of the VCSNS COL FSAR, Revision 2, incorporates by reference Section 15.4 of the AP1000 DCD, Revision 17.

In addition, in Section 1.9 of the VCSNS COL FSAR, the applicant provided the following:

##### *Generic Letter 85-05*

In its letter dated August 23, 2010, the applicant endorsed a letter dated January 22, 2010, from the VEGP applicant that proposed to include Generic Letter (GL) 85-05, "Inadvertent Boron Dilution Events," in Table 1.9-204 of the FSAR as part of STD COL 1.9-2 to address Bulletins and GLs.

#### **15.4.3 Regulatory Basis**

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

#### **15.4.4 Technical Evaluation**

The NRC staff reviewed Section 15.4 of the VCSNS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.<sup>1</sup> The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to reactivity and power distribution anomalies. The results of the NRC staff's evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in

evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP, Units 3 and 4) were equally applicable to the VCSNS Units 2 and 3 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 2, to the VCSNS COL FSAR. In performing this comparison, the staff considered changes made to the VCSNS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the VCSNS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 15.4.4 of the VEGP SER:

*Generic Letter 85-05*

*GL 85-05, "Inadvertent Boron Dilution Events," informed each PWR licensee of the NRC staff position resulting from the evaluation of Generic Issue 22, "Inadvertent Boron Dilution Events," and urges each licensee to ensure that its plants have adequate protection against boron dilution events. GL 85-05 was evaluated as a part of the AP1000 DCD review, and the evaluation was documented in NUREG-1793, Chapter 20. GL 85-05 was resolved based on the analyses of inadvertent boron dilution events described in AP1000 DCD Section 15.4.6, which show that in all modes of operation the inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. The staff also stated that COL applicants should develop plant-specific emergency operating procedures (EOPs) that address the boron dilution events. The development of EOPs is identified as COL Information Item 13.5-1, Plant Procedures, which is addressed in BLN FSAR Section 13.5. Therefore, based on the above, the applicant needs to reinsert a reference to GL 85-05 in FSAR Table 1.9-204 and provide a cross reference to COL Information Item 13.5-1. This is **Open Item 15.4-1**.*

*Resolution of Standard Content Open Item 15.4-1*

*To address Open Item 15.4-1 in the BLN SER with open items, the VEGP applicant stated in its letter dated January 22, 2010, that VEGP COL FSAR*

*Table 1.9-204, "Generic Communications Assessment," would be revised to list GL 85-05 with a cross-reference to VEGP COL FSAR Section 13.5. Until this change is incorporated in a future version of the VEGP COL FSAR, this item is being tracked as **Confirmatory Item 15.4-1**.*

#### **15.4.5 Post Combined License Activities**

There are no post-COL activities related to this section.

#### **15.4.6 Conclusion**

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to reactivity and power distribution anomalies, and there is no outstanding information expected to be addressed in the VCSNS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes, pending closure of **Confirmatory Item 15.4-1**, that the relevant information presented in the VCSNS COL FSAR related to GL 85-05 is acceptable. Plant-specific EOPs, which will include responding to abnormal events such as the boron dilution events discussed in GL 85-05, are evaluated by the staff in Section 13.5 of this SER.

### **15.5 Increase in Reactor Coolant Inventory**

Analyses focused on the increase in reactor coolant inventory address AOOs that could result in an increase in RCS inventory. Increased inventory can be caused by:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunctions that increases reactor coolant inventory

Section 15.5 of the VCSNS COL FSAR, Revision **2**, incorporates by reference, with no departures or supplements, Section 15.5, "Increase in Reactor Coolant Inventory," of Revision **17** of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

### **15.6 Decrease in Reactor Coolant Inventory**

Analyses focused on the decrease in reactor coolant inventory address AOOs and accidents that could result in a decrease in RCS inventory. Decreased inventory can be caused by the following:

- Inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system
- Failure of small lines carrying primary coolant outside containment
- Steam generator tube failure
- LOCA resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB)

Section 15.6 of the VCSNS COL FSAR has one item, VCS COL 2.3-4, related to site-specific  $\chi/Q$  values. The effect of VCS COL 2.3-4 on the design-basis accident (DBA) radiological consequences analyses is addressed in Section 15A of this SER.

With the exception of the item noted above, Section 15.6 of the VCSNS COL FSAR, Revision 2, incorporates by reference Section 15.6, "Decrease in Reactor Coolant Inventory," of Revision 17 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

## **15.7 Radioactive Release From a Subsystem or Component**

### **15.7.1 Introduction**

The group of events considered includes the following:

- Gas waste management system leak or failure
- Liquid waste management system leak or failure (atmospheric release)
- Release of radioactivity to the environment via liquid pathways
- Fuel handling accident
- Spent fuel cask drop accident

### **15.7.2 Summary of Application**

Section 15.7 of the VCSNS COL FSAR, Revision 2, incorporates by reference Section 15.7 of the AP1000 DCD, Revision 17.



In addition, in VCSNS COL FSAR Section 15.7, the applicant provided the following:

AP1000 COL Information Item

- VCS COL 15.7-1

The applicant provided additional information in VCS COL 15.7-1 to address COL Information Item 15.7-1, “Consequences of Tank Failures.” This COL item is addressed by the applicant in VCSNS COL FSAR Section 2.4.13.

**15.7.3 Regulatory Basis**

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the radioactive release from a subsystem or component are given in Section 11.2 of NUREG-0800, including Branch Technical Position (BTP) 11-6, and Section 2.4.13 of NUREG-0800, Acceptance Criterion Number 5.

The regulatory basis for acceptance of the supplementary information on consequences of a tank failure is established in:

- 10 CFR Part 20, “Standards for protection against radiation,” Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage”
- 10 CFR 20.1301, “Dose limits for individual members of the public”
- 10 CFR 20.1406, “Minimization of contamination”
- 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criteria (GDC) 60, “Control of Releases of Radioactive Materials to the Environment,” and GDC 61, “Fuel Storage and Handling and Radioactivity Control”
- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors”
- 10 CFR 50.36a, “Technical specifications on effluents from nuclear power reactors”
- 10 CFR 52.80(a), “Contents of applications; additional technical information”
- Regulatory Guide (RG) 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning”

- RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1
- RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," Revision 1
- RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2, Regulatory Position C.1.1

#### **15.7.4 Technical Evaluation**

The NRC staff reviewed Section 15.7 of the VCSNS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.<sup>1</sup> The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the radioactive release from a subsystem or component. The results of the NRC staff's evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the VCSNS COL FSAR:

##### AP1000 COL Information Item

- VCS COL 15.7-1

COL Information Item 15.7-1 states:

Combined License applicant referencing the AP1000 certified design will perform an analysis of the consequences of potential release of radioactivity to the environment due to a liquid tank failure as outlined in subsection 15.7.3.

The applicant addresses the consequence of a liquid waste tank failure in VCSNS COL FSAR Section 2.4.13. The staff's evaluation of liquid waste tank failure is described in Section 11.2, "Liquid Waste Management Systems," of this SER.

#### **15.7.5 Post Combined License Activities**

There are no post-COL activities related to this section.

#### **15.7.6 Conclusion**

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to radioactive release from a subsystem or component, and there is no outstanding information expected to be addressed in the VCSNS COL FSAR related to this section. The results of the NRC staff's

technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the VCSNS COL FSAR is acceptable and meets the regulatory guidance in Sections 2.4.13 and 11.2 of NUREG-0800. The staff based its conclusion on the following:

- VCS COL 15.7-1 is acceptable based on the evaluations in Sections 2.4.13 and 11.2 of this SER.

### **15.8 Anticipated Transients Without Scram**

Analyses focused on anticipated transients without scram (ATWS) address an AOO during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system.

Section 15.8 of the VCSNS COL FSAR, Revision 2, incorporates by reference, with no departures or supplements, Section 15.8, "Anticipated Transients Without Scram," of Revision 17 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

## **Appendix 15A Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents**

### **15A.1 Introduction**

This appendix includes the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

### **15A.2 Summary of Application**

In the VCSNS COL FSAR, Revision 2, Chapter 15, "Accident Analyses," the applicant incorporated by reference Appendix 15A to Chapter 15, "Accident Analysis," of the AP1000 DCD, Revision 17.

In addition, the applicant provided the following:

#### **AP1000 COL Information Item**

- VCS COL 2.3-4

In VCSNS COL FSAR Sections 15.6 and 15A, the applicant provided additional information in VCS COL 2.3-4 on site-specific  $\chi/Q$  values to partially resolve COL Information Item 2.3-4. The applicant provided additional information in VCSNS COL FSAR Section 2.3.4 to resolve the remaining portion of COL Information Item 2.3-4, and the staff's review of this portion is in Section 2.3.4 of this SER.

### **15A.3 Regulatory Basis**

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the accident analyses are given in Section 15.0.3 of NUREG-0800.

Requirements for the technical information in the FSAR for the application for a COL are given in 10 CFR 52.79. In particular, 10 CFR 52.79(a)(1)(vi) requires a description and safety assessment of the site on which the facility is to be located, including an evaluation of the offsite radiological consequences of postulated accidents to show that the site characteristics comply with the following offsite radiological consequence evaluation factors:

- (A) An individual located at any point on the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sievert (Sv) (25 roentgen equivalent man (rem)) total effective dose equivalent (TEDE).
- (B) An individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission

product release (during the entire period of its passage) would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

Applications for DCs must include similar evaluations to show compliance with 10 CFR 52.47(a)(2), which includes the same offsite radiological consequence evaluation factors as given in 10 CFR 52.79(a)(1). In other words, both the AP1000 DCD and the COL FSAR must have DBA radiological consequences analyses that estimate a dose at or below 0.25 Sv (25 rem) TEDE at the EAB and LPZ receptors.

Compliance with the control room habitability dose requirements of 10 CFR Part 50, Appendix A, GDC 19, "Control Room," requires that the applicant show that, for a plant located at the VCSNS site, the control room provides adequate radiation protection to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) TEDE to permit access and occupancy of the control room under accident conditions for the duration of the accident.

#### **15A.4 Technical Evaluation**

The NRC staff reviewed Appendix 15A to Chapter 15 of the VCSNS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.<sup>1</sup> The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to radiological consequences of accidents. The results of the NRC staff's evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the VCSNS COL FSAR:

##### AP1000 COL Information Item

- VCS COL 2.3-4

In VCSNS COL FSAR Sections 15.6 and 15A, the applicant stated that it provided additional information in VCS COL 2.3-4 to partially resolve COL Information Item 2.3-4, which states:

Combined License applicants referencing the AP1000 certified design will address the site-specific  $\chi/Q$  values specified in [DCD] subsection 2.3.4. For a site selected that exceeds the bounding  $\chi/Q$  values, the Combined License applicant will address how the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values given in 10 CFR Part 50.34 and control room operator dose limits given in General Design Criteria 19 using site-specific  $\chi/Q$  values. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.

With regard to assessment of the postulated impact of an accident on the environment, the COL applicant will provide  $\chi/Q$  values for each cumulative frequency distribution which exceeds the median value (50 percent of the time).

The commitment was also captured as COL Action Items 2.3.4-1, 2.3.4-2, and 2.3.4-3 in Appendix F of NUREG-1793, which states:

The COL applicant will determine the site specific  $\chi/Q$  values. If the site-specific values exceed the bounding  $\chi/Q$  values, the COL applicant will address how the radiological consequences associated with the controlling DBA continue to meet the radiological dose consequence criteria given in Title 10, Section 50.34(a)(1)(ii)(D)(1) and (2), of the *Code of Federal Regulations* (10 CFR 50.34), using site-specific  $\chi/Q$  values.

The COL applicant will determine the site specific  $\chi/Q$  values. If the site-specific values exceed the bounding  $\chi/Q$  values, the COL applicant will address how the radiological consequences associated with the controlling DBA continue to meet the control room operator dose limits given in General Design Criteria 19, using site -specific  $\chi/Q$  values.

The COL applicant will provide  $\chi/Q$  values for each cumulative frequency distribution that exceeds the median value (50 percent of the time).

VCS COL 2.3-4 added text to the end of Section 15.6.5.3.7.3 and Section 15A.3.3 of the AP1000 DCD to state that the site-specific atmospheric dispersion ( $\chi/Q$ ) values provided in VCSNS COL FSAR Section 2.3 are bounded by the values given in AP1000 DCD Table 15A-5, "Offsite Atmospheric Dispersion factors ( $\chi/Q$ ) For Accident Dose Analysis," (offsite receptors) and Table 15A-6, "Control Room Atmospheric Dispersion Factors ( $\chi/Q$ ) For Accident Dose Analysis" (control room receptors).

The NRC staff reviewed the impact of the site-specific  $\chi/Q$  values given in response to VCS COL 2.3-4 on the radiological consequences of DBAs. The applicant did not provide site-specific doses at the EAB, LPZ, or control room for the DBAs referenced in AP1000 DCD, Chapter 15, but instead incorporated by reference the analysis of the radiological consequences in AP1000 DCD, Chapter 15.

AP1000 DCD, Chapter 15, over several sections, describes and provides results of the radiological consequences analyses for the DBAs applicable to the AP1000 design. A list of the DBAs analyzed for radiological consequences and the corresponding sections where the radiological consequences analyses for those DBAs are discussed in the AP1000 DCD is given below.

<u>DCD Section</u>	<u>Design Basis Accident</u>
15.1.5.4	Main Steam Line Break
15.3.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)
15.4.8.3	Control Rod Ejection
15.6.2	Small Line Break
15.6.3.3	Steam Generator Tube Rupture
15.6.5.3	Loss of Coolant Accident (LOCA)
15.7.4.3	Fuel Handling Accident

The DBA radiological consequences analyses in the AP1000 DCD used design reference values for the accident atmospheric dispersion factors in place of site-specific values. The  $\chi/Q$  values are the only input to the DBA radiological consequences analyses that are affected by the site characteristics. To resolve VCS COL 2.3-4, the applicant discussed the VCSNS site-specific short-term (accident)  $\chi/Q$  values in VCSNS COL FSAR Section 2.3.4. The VCSNS site-specific EAB and LPZ  $\chi/Q$  values for DBAs are given in VCSNS COL FSAR Table 2.0-201, and the control room  $\chi/Q$  values for DBAs are given in VCSNS COL FSAR Table 2.0-202. In Section 2.3.4 of this SER, the NRC staff discusses its review of the VCSNS site-specific  $\chi/Q$  values and resolution to VCS COL 2.3-4.

The estimated DBA dose calculated for a particular site is affected by the site characteristics through the calculated  $\chi/Q$  input to the analysis; therefore, the resulting dose would be different than that calculated generically for the AP1000 design in the DCD. All other inputs and assumptions in the radiological consequences analyses remain the same as in the DCD. Smaller  $\chi/Q$  values are associated with greater dilution capability, resulting in lower radiological doses. When comparing a DCD site parameter  $\chi/Q$  value and a site characteristic  $\chi/Q$  value, the site is acceptable for the design if the site characteristic  $\chi/Q$  value is smaller than the site parameter  $\chi/Q$  value. Such a comparison shows that the site has better dispersion characteristics than that required by the reactor design.

For each of the DBAs, the VCSNS site-specific  $\chi/Q$  values for each time averaging period are less than the comparable design reference  $\chi/Q$  values used by Westinghouse in the AP1000 DCD radiological consequences analyses. Since the result of the radiological consequences analysis for a DBA during any time period of radioactive material release from the plant is directly proportional to the  $\chi/Q$  for that time period, and because the VCSNS site-specific  $\chi/Q$  values are less than the comparable AP1000 DCD design reference  $\chi/Q$  values for all time periods and all accidents, then the VCSNS site-specific estimated total dose for each DBA is, therefore, less than the AP1000 DCD estimated total dose for each DBA.

Since the AP1000 DCD Chapter 15 DBA radiological consequences analyses show that the offsite radiological consequences meet the regulatory dose requirements of 10 CFR 52.47(a)(2) and the control room consequences meet the regulatory dose requirements of GDC 19, and since, by the logic above, the VCSNS site-specific DBA radiological consequences are estimated to be less than those calculated in AP1000 DCD, then the applicant has sufficiently shown that the DBA offsite radiological consequences meet the requirements of

10 CFR 52.79(a)(1) and the DBA control room radiological consequences meet the requirements of GDC 19.

The effect of the site-specific  $\chi/Q$  values on the Technical Support Center radiological habitability is evaluated by the NRC staff in SER Section 13.3 as part of its evaluation of VCS DEP 18.8-1.

#### **15A.5 Post Combined License Activities**

There are no post-COL activities related to this section.

#### **15A.6 Conclusion**

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the evaluation models and parameters for analysis of radiological consequences of accidents, and there is no outstanding information expected to be addressed in the VCSNS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the VCSNS COL FSAR is acceptable and meets the requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 50, Appendix A, GDC 19. The staff based its conclusion on the following:

- VCS COL 2.3-4 is acceptable because the DBA offsite radiological consequences meet the requirements of 10 CFR 52.79(a)(1) and the DBA control room radiological consequences meet the requirements of GDC 19.



## **Appendix 15B Removal of Airborne Activity from the Containment Atmosphere Following a LOCA**

This appendix includes information related to the AP1000 design, which does not depend on active systems to remove airborne particulates or elemental iodine from the containment atmosphere following a postulated LOCA with core melt. The AP1000 applicant stated that naturally occurring passive removal processes provide significant removal capability such that airborne elemental iodine is reduced to very low levels within a few hours and the airborne particulates are reduced to extremely low levels within 12 hours.

Appendix 15B of the VCSNS COL FSAR, Revision 2, incorporates by reference, with no departures or supplements, Appendix 15B, "Removal of Airborne Activity from the Containment Atmosphere Following a LOCA," of Revision 17 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.<sup>1</sup> The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VCSNS COL application are documented in NUREG-1793 and its supplements.

**Table 15.0-1. Power Calorimetric Uncertainty Methodology**

<b>Design Commitment</b>	<b>Inspections, Tests, Analyses</b>	<b>Acceptance Criteria</b>
<p>4. The plant calorimetric uncertainty and plant instrumentation performance is bounded by the 1 percent calorimetric uncertainty value assumed for the initial reactor power in the safety analysis.</p>	<p>Inspection will be performed of the plant operating instrumentation installed for feedwater flow measurement, its associated power calorimetric uncertainty calculation, and the calculated calorimetric values.</p>	<p>a) the as-built system takes input for feedwater flow measurement from a Caldon [Cameron] LEFM CheckPlus™ System;            b) the power calorimetric uncertainty calculation documented for that instrumentation is based on an NRC-accepted Westinghouse methodology and the uncertainty values for that instrumentation are not lower than those for the actual installed instrumentation; and            c) the calculated calorimetric power uncertainty measure values are bounded by the 1 percent uncertainty value assumed for the initial reactor power in the safety analysis.</p>