Need to be revised to Rev. 2. (Typical Comment)
This TR was finally revised into Rev. 2 which is consistent with DCD TIER 2, 6.2.9(References) in Rev. 3.

LOCAs and MSLBs listed in D[illegible]ned the containment analysis subseq[illegible]de GOTHIC (Reference 1), which calculates containment pressure and temperature during the course of the transient. Appendices A through G to the TR (TR) APR1400-Z-A-NR-14007, ~~Revision 0~~, “LOCA Mass and Energy Release Methodology,” (Reference 2) describe the methodology and the GOTHIC containment model used to predict the containment pressure and temperature response to a spectrum of high-energy line break DBAs in the large, dry APR1400 containment building. The sources of generated and stored energy in the RCS and secondary coolant system considered in the LOCA analyses include: primary coolant, secondary coolant, primary walls (including reactor internals), secondary walls, SI water, core power, and decay heat.

The containment design pressure of 4.218 kg/cm 96 (60 psig 93 15 kPa) is based on the worst-case LOCA and bounds all of the postulated secondary system piping rupture events for peak containment pressure. The DCD Tier 2 Section 6.2.1.1.1 reports that a design margin of 10 percent has been taken into account in the determination of this value. The containment design temperature is 143.3 ℃ (290.0 ℉). Per Reference 2, the applicant calculated a DBA LOCA peak saturation temperature of 134.~~59~~ ℃ (274.~~25~~ ℉). The APR1400 containment is also designed for a limiting containment pressure reduction event to withstand an external pressure loading of 0.28 kg/cm^{2}G (4.0 psig or ~~27.5 kPa~~) relative to ambient pressure.

27.5 kPaG or 128.8 kPa?, (Typical Comment for unit conversion.)

ITAAC: The ITAAC associated with this evaluation of AP[illegible] provided in DCD Tier 1 Table 2.2.1-2, and Table 2.11.1-2[illegible]

TS: The TS associated with this evaluation of DCD Tier 2 Section 6.2.1.1 are provided in DCD Tier 2, Chapter 16, “Technical Specifications,” Section 3.6, “Containment Systems.” The TS that apply specifically to this review area include Sections 3.6.1, “Containment,” 3.6.4, “Containment Pressure,” and 3.6.5, “Containment Air Temperature.”

6.2.1.1.3 Regulatory Basis

The associated acceptance criteria for this area of review are given in NUREG–0800 Section 6.2.1.1.A. Review interfaces with other SRP sections can also be found in NUREG–0800, Section 6.2.1.1.A. The acceptance criteria are based on meeting the relevant regulatory requirements as summarized below:

- GDC 13, “Instrumentation and control,” requires instrumentation to be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety. This means that instrumentation shall be capable of operating in the post-accident environment in order to monitor the containment atmosphere pressure and temperature, and the sump water level and temperature following an accident. It shall have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Meeting this requirement helps ensure that the containment precludes the release of radioactivity to the environment.

- GDC 16, “Containment design,” as it relates to the reactor containment and associated systems being designed to ensure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Since the primary reactor containment is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environment, preserving containment integrity under the dynamic conditions imposed by postulated LOCAs is is conservative as it would increase the system flow rate and maximize the break flow during the post-reflood period. The response also stated that the assumption of steam quenching and the description of the removal of the remaining stored energy in the primary and secondary systems are identical to that of the reflood period. The staff agrees that not taking credit for condensation after the turndown to low SIT flow during post-reflood period was conservative. The response also provided details of the lumped-parameter GOTHIC model for the RCS used to calculate the M&E release through the break during the decay heat phase. The RCS model has three lumped-parameter volumes that represent a RCS core, a downcomer and a hot/cold leg piping segment. The response also submitted the associated markups to appropriately revise DCD Tier 2 Section 6.2.1.3.5, "Description of Post Reflood Model," and TR (Reference 1) Section 3.7, "Description of Post-Reflood Model," accordingly. However, in the July 7, 2016 public teleconference, the staff inquired whether the use of the GOTHIC code for the decay heat phase M&E release analysis can be considered appropriate for this application, as neither the DCD nor the TR discussed it. The applicant submitted a supplemental response to RAI 394-8460, Question 06.02.01.03-10, dated November 18, 2016 (ML16323A494) that characterized the decay heat phase to be a relatively stable period due to the release of core decay heat and the sensible energy of the RCS and SGs metal under thermal equilibrium, to the containment and, thus, does not require specific conservative models to calculate the break flow. The response also cited that the use of GOTHIC for the M&E release calculations for the Dominion's power plant during the post-reflood decay heat phase in their containment analysis methodology, was previously approved by the NRC in their containment analysis methodology (Reference 7). The staff found the previous NRC approval of the use of the GOTHIC code for the decay heat phase M&E release analysis to be an appropriate justification. The staff finds the applicant's response acceptable and, based on the review of the DCD and the TR, the staff has confirmed incorporation of the changes described above; therefore, RAI 394-8460, Question 06.02.01.03-10, is resolved and closed.

According to SRP Section 6.2.1.3 Acceptance Criterion No. 1C(v), the fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in SRP Section 9.2.5. However, the DCD or TR (Reference 1) provides no information to ascertain how conservative the decay energy model is compared to the one given in SRP Section 9.2.5. SRP Section 6.2.1.3 Acceptance Criterion No. 1C(v) also states that steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water. No such description is found in the DCD or the TR. On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-11, to request the applicant to clarify these two aspects in the DCD. In its response to RAI 394-8460, Question 06.02.01.03-11 (ML16160A034), dated June 8, 2016, the applicant explained that the decay energy model in the LOCA M&E calculation uses two different decay heat curves both of which are based on the ANSI/ANS 5.1-1994, "Decay Heat Power in Light Water Reactors," standard decay heat curve. One is ANS 5-1971, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors," decay heat curve, which is chosen for decay energy releases during the earlier phases of a LOCA (blowdown, reflood and post-reflood), and the other is ANSI/ANS 5.1-1979 decay heat curve, which is used for the decay heat phase. The staff determined that, while the ANS 5-1971 is the required decay heat model for Appendix K analysis, the ANS 5.1-1979 model is appropriate for the containment response analysis for the decay heat phase. For the earlier phases of a LOCA until the end of post-reflood, the ANS standard decay curve corrected for decay of the heavy elements U-239 and NP-239 has been incorporated into the CEFLASH-4A. The decay heat contribution from actinides other than U-239 and NP-239[9] is additionally taken into account to the decay heat curve for conservatism. The staff agrees that the decay heat model used in the

NP−239

analysis is more conservative compared to that provided in the SRP Section 9.2.5, “Ultimate Heat Sink.”

The response also stated that in the RCS model for the LOCA M&E release during decay heat phase, energy release from coolant and metal of the SGs’ secondary side are modeled using the GOTHIC heater components, which are submerged in the RCS core volume. In the RCS model, the downcomer receives the IRWST water through the SIP, and then feeds it to the core as needed to make up for steaming and returns the remaining water to the IRWST as spillage without temperature increase. This modeling approach basically excludes mixing of the steam with the SI injection water and maximizes steaming in the core. The staff accepts that APR1400 methodology conservatively assumes that there is no mixing of steam and SI water in the RCS during the decay heat phase. The response also submitted the associated markups to revise DCD Tier 2 Section 6.2.1.3.2, “Energy Sources,” and TR Section 3.4, accordingly. The staff accepts the applicant’s response as it has provided the required information. Based on the review of the DCD and the TR, the staff has confirmed incorporation of the changes described above; therefore, RAI 394-8460, Question 06.02.01.03-11, is resolved and closed.

Initial testing for the SIS (specifically, DCD Tier 2, Sections 14.2.12.1.21, “Safety Injection System Test,” and 14.2.12.1.22, “Safety Injection Tank Subsystem Test”) are addressed in Section 6.3 of this SER.

ITAAC: The regulations in 10 CFR 52.47(b)(1) requires, in part, that a DC application contain the ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the as-built plant incorporating the ITAAC will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC’s regulations. ITAAC for the SIS is evaluated in Section 6.3 of this SER. The discharge values for the SITs and SIPs are such that they meet the values assumed in the safety analysis related to this section, and are verified by ITAAC items 9.a.i and 9.a.iii in Table 2.4.3-4. The staff finds the ITAAC acceptable.

The staff determined DCD Tier 1 Table 2.11.1-2, provides ITAAC No. 4 and ITAAC No. 5 to demonstrate two key containment functional design commitments, structural integrity and acting as a barrier against uncontrolled fission products release, required by GDC 16, GDC 38, and GDC 50. ITAAC No.4 ensures that an analysis will be performed that concludes that the as-built containment peak pressure following a high energy line break remains below its design pressure of 4.218 kg/cm^{2}G (60 psig or 515 kPa) with more than 10 percent margin above the maximum calculated peak pressure. ITAAC No.5 ensures that an analysis will be performed that concludes that the as-built containment pressure is reduced to less than 50 percent of its maximum calculated peak pressure for the design basis LOCA within 24 hours after the postulated accident.

TS: TS for initial containment pressure and temperature are specified in APR1400 DCD Tier 2 Chapter 16, Sections 3.6.4 and 3.6.5 respectively. TS 3.6.4 specifies the LCO, “Containment pressure shall be ≥ -0.007 kg/cm^{2}G (−0.1 psig) and ≤ +0.07 kg/cm^{2}G (+1.0 psig).” Similarly, for temperature the LCO requires that “Containment average air temperature shall be ≤ 49 − C (120 ℉).” The safety analyses in DCD Tier 2 Section 6.2.1.3, assume a scenario-dependent SI flow. TS Surveillances 3.5.1.2 and 3.5.2.5 ensure that during normal operation, the SIT volume and SIP flow, respectively, are greater than the minimum values assumed in the safety analysis. As part of the TS for the APR1400, SRs provided for the differential pressure developed by the pump at minimum flow rate (3.5.2.4) and for the design flow rate at design pressure (3.5.2.5). However, no provisions are included to ensure that the SIPs are capable of maintaining the flow

°C