RAIO-0817-55624

August 25, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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**SUBJECT:** NuScale Power, LLC Response to NRC Request for Additional Information No. 77 (eRAI No. 8895) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 77 (eRAI No. 8895)," dated June 28, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 8895:

- 09.02.05-2

NuScale requests that the security-related information in Enclosure 1 be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. Enclosure 2 contains a public version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8895, nonpublic

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8895, public

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 8895, nonpublic Security-Related Information - Withhold Under 10 CFR §2.390

**Enclosure 2:**

NuScale Response to NRC Request for Additional Information eRAI No. 8895, public

# Response to Request for Additional Information
# Docket No. 52-048

**eRAI No.:** 8895
**Date of RAI Issue:** 06/28/2017

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**NRC Question No.:** 09.02.05-2

10 CFR 52.47(a)(2) requires that a standard design certification applicant provide a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.

The staff reviewed FSAR Tier 2, Section 9.2.5, on the ultimate heat sink (UHS) design capacity for abnormal and accident conditions including the size and heat loads of the UHS to verify the adequacy of the long term UHS capacity. The heat loads identified in FSAR Table 9.2.5-2, "Ultimate Heat Sink Heat Loads: Boil off Event," as indicated in footnote 3, account for decay heat only. Table 9.2.5-2 does not indicate the inclusion of the sensible heat nor the heat loads from the stored spent fuels, which are discussed in FSAR Tier 2, Sections 9.2.5.2.3 and 9.2.5.4.

In order to better describe the UHS design capacity calculation, the applicant is requested to:

- clarify FSAR Tier 2, Table 9.2.5-2, on the heat loads used for the UHS design.
- provide in FSAR Tier 2, Section 9.2.5, "Ultimate Heat Sink," the amount of water in the UHS relating to the UHS design capacity.
- provide the thermal analysis for the UHS under the limiting case, including the methodology, and all the assumptions (e.g., the initial pool temperature, water level, etc.).

The FSAR should be modified accordingly.

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**NuScale Response:**

FSAR Section 9.2.5 has been revised to provide additional details in Section 9.2.5.4, Safety Evaluation, to clarify the capability of ultimate heat sink (UHS) to safely cool a NuScale Power Plant for abnormal and accident conditions. The revisions address the limiting case initial conditions and assumptions, including initial volume of water in the UHS, and demonstrate the

adequacy of the long term cooling capacity of the UHS.

The revisions in FSAR Section 9.2.5.4 expand and clarify the limiting case analysis of boiling of the UHS pool water assuming an accident in one NuScale Power Module (NPM) and coincident loss of electric power causing a shutdown of the remaining NPMs in operation. The initial conditions use the UHS pool water starting temperature and level based on the limiting conditions for operation in Section 3.5.3 of Part 4, Generic Technical Specifications, of the NuScale Design Certification Application. In addition, the revisions clarify that the following heat loads are included in the limiting case analysis: sensible heat from the water and metal in each NPM, and decay heat from the start of the event for the NPMs and the spent fuel assemblies in the fuel storage racks.

FSAR Section 9.2.5.4 has been revised as described above and shown in the attached markups. FSAR Table 9.2.5-2 has been revised with the updated results of the evaluation. Also for this table, the footnotes were replaced and now refer to FSAR Section 9.2.5.4 for a description of the initial conditions and assumptions for the evaluation. In addition, FSAR Section 15.0.3.7.2 has been revised as shown in the attached markups to be consistent with the limiting case analysis of UHS pool water boiling in revised Table 9.2.5-2. Other sections of FSAR Section 9.2.5 have been revised as shown in the attached markups to consolidate the information on the limiting case analysis in Section 9.2.5.4, and to clarify the text for consistency.

**Impact on DCA:**

FSAR Sections 9.2.5.2.2, 9.2.5.2.3, 9.2.5.4, and 15.0.3.7.2; and Tables 9.2.5-1 and 9.2.5-2 have been revised as described in the response above and as shown in the markup provided with this response.

The SFP weir is designed to maintain a minimum of 10 feet of water over spent fuel stored in the storage racks. The structural components forming the UHS are the pool walls and floors and are considered part of the RXB structure. The functional and code requirements, and associated bases, relevant to the pool structural components are described in Section 3.8.1.

UHS water surge control is provided by the pool surge control system (PSCS) during dry dock operations. For additional information, refer to Section 9.1.3.

The UHS water is circulated through the pool cleanup system to remove impurities to maintain water quality and reduce radionuclide contaminants. Water removed from the pools by the PSCS, spent fuel cooling system, or reactor pool cooling system can be routed for cleanup in the pool cleanup system filters and resin beds and returned to the pools. Refer to Section 9.1.3 for additional information.

Leakage of UHS pool water is collected by the pool leakage detection system to provide appropriate containment and confinement of radioactivity, and minimize contamination. Leakage from a pool liner results in flow out of the leakage detection channels. The channels are routed to collect and quantify pool leakage. Refer to Section 9.1.3 for additional information.

The concentration of radionuclides in the UHS water is monitored by the process sampling system. Refer to Section 9.3.2 for additional information on process sampling.

#### 9.2.5.2.2 Normal Operation

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The UHS pool water level, temperature, and quality are maintained within an operational control band to provide assurance that water is available to provide personnel and public safety during normal plant operation. Level is maintained through interface with the SFP cooling system. Temperature is maintained through interface with the SFP cooling and the reactor pool cooling systems. The heat load evaluation in the UHS for normal operation with power available for the pool cooling systems is presented in Section 9.1.3. UHS parameters are provided in Table 9.2.5-1.

Dry dock operations maintain water level in the dry dock to match the pool level prior to opening the dry dock gate. The PSCS maintains the level in the dry dock without affecting the UHS level. The PSCS operation is discussed in Section 9.1.3.

#### 9.2.5.2.3 Operation During Abnormal and Accident Conditions

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During abnormal conditions, when electric power is available, the reactor pool cooling and SFP cooling systems remove the heat transferred from the NPMs and spent fuel to the UHS. The heat that is rejected through the decay heat removal system or the containment vessel is removed by the reactor pool cooling system

via the UHS water. ~~The heat load evaluation with power available to pool cooling systems is presented in Section 9.1.3.~~For an abnormal condition where one NPM is cooled in this manner, the UHS pool water temperature would initially increase above the normal operating temperature shown in Table 9.2.5-1. With AC power available, the active pool cooling systems operate to return the water temperature to the normal operating value.

During an event where loss of electric power occurs, the volume of water already in the pool provides the inventory for the necessary heat removal. Upon loss of power, the reactor pool cooling and SFP cooling systems shut down. The UHS water expands as it heats and eventually begins to boil. Heat continues to be removed from the pool through boiling and evaporation, removing enough heat to maintain the spent fuel and fuel in the NPMs sufficiently cool to prevent fuel damage. The design is such that UHS water boil-off will continue to remove heat from the power modules and spent fuel for greater than 30 days without the need for operator action, makeup water, or electric power.

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~~An accident in one NPM concurrent with a loss of all AC power is assumed to result in a shutdown of all NPMs (as a result of loss of power). In this event, all NPMs will be isolated from the circulating water system and dissipating decay heat to the UHS. The scenario does not credit the pool cooling systems or the RXB ventilation, except for passive steam release, allowing energy from the NPMs and spent fuel to heat up and eventually boil the water in the UHS. The decay heat loads are calculated using American National Standards Institute (ANSI)/American Nuclear Society (ANS) 5.1 (Reference 9.2.5-1) and are presented in Table 9.2.5-2 as a function of time. The UHS level that provides 30 days of water without additional makeup is listed in Table 9.2.5-1.~~

~~The evaluation of pool level for a pool boil off event assumed: the heat load of 12 NPMs initially at full power (160 MWt); the SFP contains 1404 fuel assemblies representing a nominal 18 years of power operation; and 13 assemblies from a recent refuel. Each NPM has a 24-month refueling cycle; therefore, for a 12-NPM plant, a staggered refueling scheduled of every 2 months is assumed. No water is assumed to return to the pool after evaporating from the UHS and no heat is assumed to be transferred to the surrounding pool liner and walls.~~

~~Without~~Section 9.2.5.4 describes the evaluation of the capability of the UHS to cool the NPMs and the spent fuel following an accident or transient, including a loss-of-coolant accident (LOCA) in an NPM. As shown in the evaluation, without electrical power to supply normal pool cooling and makeup systems, the large UHS water volume provides sufficient time for actions to restore UHS water level using defense-in-depth design provisions. These defense-in-depth design provisions include: the makeup water connection and associated piping that provides a pathway for makeup water, and connection capability to the fire protection system.

The level of water over the spent fuel is monitored and in the event that level continues to drop, the qualified makeup line is used to add water to the pool. The

qualified makeup line provides a connection outside the RXB for a source of water to fill the SFP area and the UHS pool. The weir assures that water is available to cover the spent fuel first, before adding water to the rest of the pool complex.

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To prevent pressurization in the UHS area of the RXB, credit is taken for a passive vent path (RXB exhaust ventilation system). The system filters and controls the release of airborne radioactive material from inside of the RXB, including from pool water evaporation for loss of normal power supply (see Section 9.4.2). Section 15.0.3 addresses the radiological consequences of the UHS pool boiling.

### 9.2.5.3 Refueling Operations

The NPMs are located in the reactor pool during power operations, and one is moved by the RXB crane to the RFP to perform refueling and maintenance operations. The upper reactor vessel and upper containment vessel are placed in the module inspection rack located in the dry dock for inspections. Spent fuel is moved from the open reactor vessel, which is staged in the RFP, through an open passage over the weir wall into the SFP where the fuel storage racks are located.

The dry dock temporarily houses an upper NPM for inspection, testing, and maintenance. The PSCS controls the water level in the dry dock to provide access to the various components for inspections and maintenance. The dry dock gate is closed and the water volume separated from the UHS to allow level adjustment in the dry dock.

The boron concentration in the UHS pool, including the SFP, is maintained at a level that will preclude criticality during refueling operations (refer to Table 9.1.3-2). The UHS contains the minimum boron concentration for reactivity control of fuel during the refuel evolution when the reactor vessel is open to the pool. Boron concentration in the UHS is monitored. The SFPCS has the ability to add boron to the UHS along with the option to add boron directly to the pool manually. Boron concentration reduction is performed when necessary by using the SFPCS to add low boron concentration water while bleeding off pool water to the liquid radioactive waste system.

Water level in the UHS is capable of maintaining core cooling during refueling operations. The volume of water is able to perform the function of ultimate heat sink without the need to transfer the heat to a secondary system.

While the upper NPM is in the dry dock for inspections, radiologically activated equipment located on the lower sections of the module will be shielded by water in the dry dock. The water level in the dry dock is adjusted to provide radiation shielding depending on the work being performed.

### 9.2.5.4 Safety Evaluation

RAI 09.02.05-2

The UHS is a passive system and does not require electric power (AC or DC) to remove heat. ~~The use of non-safety equipment enables makeup water to be available during~~

~~the post-72 hour period following the onset of a station blackout, which stabilizes pool water inventory.~~Following a postulated accident and the assumed onset of a station blackout, makeup water could be added through the qualified UHS makeup line from outside of the RXB using nonsafety-related equipment to stabilize pool water inventory. However, no credit is needed for pool water additions for more than 30 days as shown by the following evaluation of the boil off of the initial UHS pool water inventory.

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An accident in one NPM concurrent with a loss of AC power is assumed to result in a shutdown of up to 11 NPMs (as a result of loss of power). For these conditions, a total of 12 NPMs are assumed to be isolated from the feedwater and main steam systems, and transferring heat to the UHS. The evaluation of UHS pool water boil off does not credit the active pool cooling systems or the RXB ventilation, except for passive steam release, allowing energy from the NPMs and spent fuel to heat up and boil the water in the UHS.

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~~A~~Without the addition of makeup water, a reduction in UHS water level ~~reduction~~ would result from an extended unavailability of the reactor pool cooling and SFP cooling systems (e.g., initiated by an extended loss of AC power). ~~A pool boil off event~~Boiling of the water in the UHS pools is evaluated assuming a prolonged unavailability of these systems. For these conditions~~this event~~, UHS water temperature and level would initially increase, and UHS water level would then decrease as a result of pool water evaporation and boiling.

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The normal operating range of the water level in the UHS pools is provided in Table 9.2.5-1. At the upper end of this range, the volume of the water in the UHS is more than 6.4 million gallons. The water level in UHS prior to the start of the evaluation of UHS pool water boiling is assumed to be at the lower end of this range and is the minimum value allowed by Technical Specifications before actions must be taken as a Limiting Condition for Operation. The volume of water in the UHS pools for this lower water level is more than 6.3 million gallons.

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The water temperature in UHS prior to the start of the analysis of pool water boiling is conservatively assumed to be 140 degrees F. This temperature corresponds to the upper end of the normal operating range for UHS pool water temperature and is the maximum value allowed by Technical Specifications before actions must be taken as a Limiting Condition for Operation.

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The heat loads for this evaluation are assumed to be from the 12 NPMs and the stored spent fuel assemblies. No other heat loads or systems are cooled by the UHS. The decay heat loads for the NPMs and spent fuel are calculated using American National Standards Institute (ANSI)/American Nuclear Society (ANS) 5.1 (Reference 9.2.5-1) and

the total cumulative heat energy cooled by the UHS is presented in Table 9.2.5-2 as a function of time.

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Prior to the start of the evaluation of UHS pool water boiling, 12 NPMs are assumed to be in operation at their normal operating power level of 160 MWt per NPM. An accident is assumed to occur in one NPM with a coincident loss of AC power and the shutdown of 11 NPMs. Based on a conservative approach to modeling the transfer of sensible heat from the metal and the water in an NPM to the UHS water, the total heat rejected from the NPM with the assumed accident is limiting for any type of accident. That is, the sensible heat in the NPM is addressed in the analysis by converting the energy associated with the cooling of the metal and water from the reactor operating temperature down to boiling, and adding this energy to the UHS pool water at the start of the evaluation. This is equivalent to assuming that the containment system isolation valves close immediately at the start of the assumed accident, which is consistent with the inadvertent main steam system isolation valve (MSIV) closure event analyzed in Section 15.2.4.2. This approach is conservative because the MSIVs are designed to close within 5 seconds and during this closure time some energy would transfer from the water in the reactor coolant system to the steam generators and then to the secondary side of the unit before the MSIVs close. Accident sequences in Chapter 15 without closure of the MSIVs would lose even more energy to the secondary side and are not limiting for UHS pool cooling capacity.

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Use of this approach for cooling of sensible heat is also applied to the 11 NPMs assumed to shut down at the start of the accident due to a loss of electric power. The sensible heat load for each of the other 11 NPMs is the same as for the NPM with the accident because there is the same assumed transfer of metal and water sensible heat from each shutdown NPM to the UHS water. With the above assumptions, the sensible heat from each of the 12 NPMs enters the UHS water through the decay heat removal system or ECCS, and is not cooled by the secondary side of the plant.

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In addition to the sensible heat, the decay heat load from each NPM is also cooled through the decay heat removal system or ECCS. The heat load to the UHS from each NPM is based on the rate of decay heat generation determined using ANSI/ANS 5.1 (Reference 9.2.5-1).

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The heat input to the UHS from the 12 NPMs for the conditions described above is the maximum that could occur for 12 NPMs during plant operations including one NPM in refueling and 11 NPMs at full power operations. For a plant with one NPM in refueling operations at the time of an accident, the total heat load to the UHS would be less than the limiting case with 12 units in operation. Once the NPM to be refueled starts the normal shutdown sequence, the unit is continually cooling and has less and less energy available to transfer to the UHS at the start of the assumed accident. The limiting case is 12 NPMs operating at the start of an accident in one unit.

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~~The heat load for this event is from 12 NPMs and~~ For the analysis of a pool water boil off, the decay heat load to the UHS pool also includes the heat added by the stored spent fuel assemblies. Each NPM has a 24-month refueling cycle; therefore, for a 12-NPM plant, a staggered schedule with a refueling every 2 months is assumed. As described in Section 9.1.2, the SFP can contain 1,404 stored spent fuel assemblies, or 18 years of stored spent fuel. For this analysis, ~~including~~the decay heat from those assemblies plus an additional 13 freshly off-loaded ~~NPM~~spent fuel assemblies is assumed.

RAI 09.02.05-2

The cumulative heat ~~load~~energy transferred to the UHS over time from these conditions is presented in Table 9.2.5-2. The analysis of UHS pool water boil off without makeup or active cooling addresses the limiting total heat load into the UHS from the NPMs and stored spent fuel.

RAI 09.02.05-2

~~The~~Other assumptions for this ~~event are as~~evaluation include the following~~s~~. As the pool heats up and boils, evaporated water is assumed to not return to the pool. No credit is assumed for heat dissipation into the surrounding pool liner, walls, and building ~~is assumed~~. Evaporation from the pools to the RXB atmosphere is not considered before the pool reaches the boiling temperature. A sensible heat of 9.2E+08 BTU from the metal and water starting at normal operating conditions in 12NPMs is assumed to be added to the water in the UHS pool before the pool water begins to boil.

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The large UHS water volume provides sufficient time for actions to restore UHS water level using defense-in-depth design provisions. The UHS level that provides 30 days of water without additional makeup or active cooling systems is listed in Table 9.2.5-1. Table 9.2.5-2 provides the times for the UHS to start boiling and to boil down to various levels.

The UHS is designed to support up to 12 NPMs with no impairment of its ability to perform required safety functions. The UHS has sufficient capacity to remove the heat energy from a design basis accident and decay heat in one unit and to achieve an orderly shutdown and cooldown of the remaining units. Water makeup to the UHS is not required to achieve the UHS safety functions.

RAI 09.02.05-2

The UHS design assures heat transfer from the containment ~~vessel~~vessels and spent fuel to the UHS under normal operating and accident conditions as an inherent consequence of the UHS physical configuration. Each containment vessel is partially immersed in the UHS, and each spent fuel assembly is submerged. Thus, the UHS provides passive cooling to transfer heat from components without reliance on active components or reliance upon AC or DC electrical power. There are no components that require alignment or isolation for the UHS to perform its safety functions.

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## Table 9.2.5-1: Relevant Ultimate Heat Sink Parameters

| UHS Parameter | | |
|---|---|---|
| **Level** | **Building Elevation (ft)** | **Pool Level (ft)** |
| Normal operating level range | Withheld - See Part 9 | 68-69 |
| Minimum level assumed for reactor building crane operation[1] | Withheld - See Part 9 | 66 |
| Minimum level for 30 day coverage for DHRS[2] | Withheld - See Part 9 | ~~61.1~~63.4 |
| Minimum level for long term cooling | Withheld - See Part 9 | 55 |
| Minimum level for FHA scrub[3] | Withheld - See Part 9 | 52 |
| Spent fuel pool weir wall | Withheld - See Part 9 | 20 |
| Minimum level to support radiation shielding[4] | Withheld - See Part 9 | 20 |
| Top of spent fuel rack | Withheld - See Part 9 | 10 |
| Reactor pool and spent fuel pool floor | Withheld - See Part 9 | 0 |
| **Temperature** | **Temperature (°F)** | |
| Minimum operating | 40 | |
| Normal operating | 100 | |
| Maximum operating | 140 | |

Notes:

[1] Maximum Reactor Building crane lifting capacity is calculated assuming a pool level of 66 ft and a pool temperature of 140 degrees F for calculating water density.

[2] ANSI/ANS 5.1-2014 is used to calculate decay heat for up to 12 NPMs and stored spent fuel assemblies with a pool water starting temperature of 140 degrees F.

[3] Level for iodine scrubbing includes: weir height + 8 ft damaged fuel + 1 ft weir clearance + 23 ft scrub

[4] ANSI/ANS 57.2-1983 maximum radiation dose of 2.5 mrem/hr

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**Table 9.2.5-2: Ultimate Heat Sink Heat Loads: Boil Off ~~Event~~Evaluation[1] Results**

| Description | Cumulative Energy[3] (BTU) | Cumulative Time (Days) | Pool Elevation (ft) |
|---|---|---|---|
| Time to UHS boiling[1] | 3.72~~5.20~~E+09 | 2.55~~4.52~~ | 68~~61.11~~ |
| Boil off to top of DHRS | 2.02~~1.66~~E+10 | 40.72~~30~~ | 45.82 |
| Boil off to top of SFP weir[2] | 3.95~~3.58~~E+10 | 119.23~~101.59~~ | 20.00 |
| Boil off to top of spent fuel rack [2] | 4.05~~3.69~~E+10 | 139.26~~119.94~~ | 10.00 |

Notes:

[1]~~Pool level providing 30 days to top of DHRS~~Initial conditions and assumptions are described in Section 9.2.5.4

- ~~[2]Assumes no operator action is taken~~
- ~~[3]ANSI/ANS 5.1-2014~~

- 10-cfm of in-leakage is assumed for ingress and egress. An additional 56 cfm of in-leakage is also assumed.

RAI 15.00.03-7

The technical support center ventilation system design modeling assumptions are provided in Table 15.0-18.

RAI 15.00.03-7

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

RAI 15.00.03-7

Potential shine radiological exposures to operators within the TSC following a radiological release event are evaluated. Direct shine, sky-shine and shine from filters are evaluated using MCNP, as described in Section 15.0.2.4.7. Reference 15.0-4 provides additional details regarding the calculation of shine doses.

RAI 09.02.05-2

#### 15.0.3.7.3 Reactor Building Pool Boiling Radiological Consequences

Without available power for the active cooling systems, the addition of makeup water, or operator action, the sensible and decay heat from the ~~reactors~~NPMs and spent fuel would heat the pool water and could eventually cause the ~~reactor pool~~water in the UHS pools to boil. Table 9.2.5-2~~, in Section 9.2.5~~ shows that it takes longer than ~~72 hours~~61 hours for the pool to reach boiling after a loss of normal AC power event. However, if the pool were to boil, the dose would be less than 0.5 rem TEDE onsite and offsite.

### 15.0.3.8 Consequence Analyses of Category 1 Events

#### 15.0.3.8.1 Failure of Small Lines Carrying Primary Coolant Outside Containment

Failure of small lines carrying primary coolant outside containment is not an event addressed in RG 1.183. The methodology used for determining dose consequences, including the iodine spiking assumptions for this event, is similar to that used for the MSLB and SGTF. The event-specific transient analysis described in Section 15.6.2 defines the time-dependent release of activity into the RXB.

The small-line break outside containment can be a break in the chemical and volume control system (CVCS) letdown line or makeup line, or the pressurizer spray line. A non-mechanistic line break occurs in the RXB allowing primary coolant from the reactor to be released into the RXB. In addition, primary coolant in the CVCS equipment (heat exchangers, filters, etc.) and piping within the RXB flows out of the other side of the break contributing less than 15,000 lbm additional primary coolant to the release. The limiting radiological scenarios identified in Section 15.6.2 are: