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February 28, 2013

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

License Amendment Request
Revise Section 3.6.5 of the Technical Specifications, "Containment Air Temperature"

Pursuant to 10 CFR 50.90, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) hereby requests an amendment to the Ginna Renewed Facility Operating License DPR-18.

The proposed amendment would revise Technical Specifications (TS) 3.6.5, "Containment Air Temperature," to increase the allowable containment average air temperature from 120°F to 125°F. The proposed change to the TS and our determination of significant hazards have been reviewed by our Plant Operation Review Committee (PORC), and it has concluded that implementation of the change will not result in an undue risk to the health and safety of the public. Pursuant to 10 CFR 50.91(b)(1), Ginna has provided a copy of this license amendment request to the appropriate state representative.

Ginna LLC requests approval of the proposed license amendment by May 1, 2014, to avoid compensatory measures and temporary modifications that are currently planned during summer months that have a minor effect on containment temperatures. Once approved, the amendment shall be implemented within 45 days. This proposed change has been analyzed in accordance with 10 CFR 50.91(a)(1) using the criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards consideration. The bases for this determination are included in Attachment (1), which provides a description of the proposed change, a technical evaluation, a regulatory safety analysis, and an environmental review. Attachment (2) provides the existing TS pages markup to show the proposed change. Attachment (3) provides the existing TS Bases markup.

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NRR

WPLNRC-1002662

If there are any questions regarding this submittal, please contact Thomas Harding at 585-771-5219 or Thomas.HardingJr@cengllc.com.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 28 day of February 2013.

Sincerely,

Joseph E. Pacher

Attachments:

(1) Description and Assessment of Proposed Changes
(2) Proposed Technical Specification Change (Markup)
(3) Proposed Technical Specification Bases Change (Markup)

cc: M. C. Thadani, NRC
W. M. Dean, NRC
Ginna Resident Inspector, NRC
A. Peterson, NYSERDA

ATTACHMENT (1)

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGES

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1.0 SUMMARY DESCRIPTION

R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) requests Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise Renewed Facility Operating License DPR-18 for the R.E. Ginna Nuclear Plant (Ginna).

Ginna LLC proposes to revise Technical Specification 3.6.5, Containment Air Temperature, to change the allowable containment average air temperature from 120°F to 125°F.

2.0 DETAILED DESCRIPTION

System Description

The function of the containment structure is to completely enclose the entire reactor and reactor coolant system and ensure that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding for both normal and accident situations. The structure must withstand the pressure and temperatures of the design-basis accident without exceeding the design leak rate.

The Containment structure is dependent on the Containment Recirculation Fan Cooler (CRFC) and Containment Spray (CS) Systems to mitigate the post design-basis accident to prevent exceeding the design pressure and temperature of containment. The CRFC system is made up of four fan cooler units, with two units per each train, and the CS system has two pumps with one per train. The CRFC system is supported by the service water (SW) system. The SW system is a once-through system that takes suction from Lake Ontario, and discharges to the lake. The system is used to cool both safety-related and non-safety related loads. The containment coolers receive water directly from the SW pump discharges without any upstream heat addition. The containment response to a design-basis accident also depends on the initial conditions of containment that exist during normal operation.

Normal Operation

The maximum average temperature inside the Containment is limited to 120°F by operation of the CRFC cooling units. The maximum allowable SW suction temperature is 85°F. During normal operation, the CRFC SW outlet valves, which are fully opened following a LOCA, are normally throttled to control containment temperature and pressure. Occasionally, during extended periods of high outside air temperature, all four coolers are used to limit the average containment temperature to 120°F. Historically, SW temperatures peak the first week of August; however, during the summer of 2012 the SW temperature peaked two weeks sooner due to an extended period of high temperatures. Correspondingly, the containment average air temperature was approaching the 120°F limit. At this time, the ultimate heat sink (UHS)---which is Lake Ontario for Ginna--- experienced a change in the depth of the different layers of water resulting in the water temperature significantly decreasing for a period of several days and the containment average air temperature decreased. Lake Ontario is subject to a phenomenon known as thermal stratification, which causes distinct layers of specific temperature water to exist. There are three distinct layers that include the epilimnion, the thermocline, and the hypolimnion. The epilimnion is warm water heated by solar radiation and ambient temperatures, and the hypolimnion is the cold water at the bottom of the lake. The thermocline is a thin layer of water that separates the other two layers, and rapidly changes in temperature with depth. A change in depth of the thermocline can result in a rapid change in intake water. Based on the recent trend of increasing ambient and lake temperatures, there is a possibility that the containment average air temperature limit of 120°F could be exceeded if there is an extended period of high temperatures without a corresponding change in the lake temperature.

Affected Technical Specifications

Technical Specification Section 3.6.5, "Containment Air Temperature," currently states:

"Containment average air temperature shall be ≤ 120°F."

The proposed change would revise Technical Specification Section 3.6.5 by adding a different limitation on the containment average air temperature. The revised Technical Specification Section 3.6.5 would read as follows:

"Containment average air temperature shall be ≤ 125°F."

To support this proposed change, accident analyses that are impacted by either the increase in initial containment air temperature or an increase in Safety Injection (SI) accumulator temperature were evaluated and analyzed. The SI accumulators are located in the Ginna containment, and are assumed to be at the same temperature as containment. The analyses impacted by the change in containment temperature that have been re-performed include the Loss of Coolant Accident (LOCA) and a Main Steam Line break (MSLB) containment response analyses. The analyses impacted by the change in SI accumulator temperature that have been re-evaluated include the core response to a large break (LB) LOCA and a small break (SB) LOCA. The combined impact on the post-LOCA long term cooling analyses was also assessed.

3.0 TECHNICAL EVALUATION

During a design basis accident (DBA), a minimum of one containment cooling train and one CS train is required to maintain the containment peak pressure and temperature below the design limits. The CS and CRFC Cooling Systems limit the containment temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are a LOCA and a MSLB. The LOCA and MSLB are analyzed using computer codes described in the sections below that were designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively.

LBLOCA Mass and Energy Release and Containment Response

A large break LOCA is initiated by the rupture of the primary coolant system piping. The primary coolant flashes to steam and escapes through the pipe break. As the steam is released to containment, the containment atmosphere pressure and temperature quickly increases. The structures in containment will absorb energy and condense steam, counteracting the initial pressure and temperature increase. The increase in containment pressure will activate the CRFC and CS systems, and will reduce containment pressure and temperature by removing energy from the containment atmosphere as the event progresses. The postulated LOCA is analyzed coincident with a loss of offsite power (LOOP). One emergency diesel generator is assumed to fail such that only one train of engineered safeguards is available.

During a LOCA event, the initial blowdown and subsequent decay heat boil off from the primary coolant system adds mass and energy (M&E) to the containment atmosphere. Both long-term and short-term effects on the containment resulting from a postulated LOCA were considered using the operating conditions for Ginna at elevated initial containment and accumulator temperature.

The increased containment temperature has no impact on the design basis short-term LOCA M&E releases that would be used for short-term sub-compartment analyses. Likewise, the increased accumulator temperature has no impact on the design basis short-term LOCA M&E releases that would be used for short-term sub-compartment analyses because the releases are calculated for only 1 second to 3 seconds and accumulators would not begin to inject until seconds into the transient.

The long-term LOCA M&E releases were analyzed for Ginna in two stages. The first of which is from 0 to 3600 seconds, i.e. the time at which energy in the primary heat structures and steam generator secondary system is released / depressurized to atmospheric pressure, (i.e. 14.7 psia and 212°F). This portion of the long-term LOCA M&E release was analyzed using the WCAP-10325-P-A (Reference 3) methodology consisting of M&E release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. The second stage, i.e. post 3600 seconds, is analyzed using Gothic version 7.2a. This methodology for the long-term LOCA M&E release was approved for use at Ginna in the extended power uprate (EPU) safety evaluation report (SER) (Reference 1).

Westinghouse previously identified non-conservative LOCA M&E release inputs for the Ginna containment response analysis. Westinghouse determined that the reactor vessel modeling did not include all appropriate vessel metal mass available from component drawings and in the reactor vessel barrel/baffle region. The LOCA M&E release analysis was initialized at a non-conservative (low) steam generator (SG) secondary pressure condition, and an error was found in the EPITOME computer code used to determine the M&E release rate during the long-term (i.e. post-reflood) SG depressurization phase of the LOCA transient (Reference 11).

These issues involve input values and methods used in performing the LOCA M&E release analysis and have been corrected. The under-prediction of M&E released into containment occurred only for the LOCA event. The mass energy release assumed in the MSLB was not affected.

Due to the increase in containment pressure and temperature associated with these changes, an additional input change was made from the model previously approved for use during EPU to increase margin between the calculated peak pressure and the containment design pressure limit. The energy from the SG secondary metal mass above the secondary side liquid level will not rapidly transfer any stored energy to the primary side break flow. Hence, the stored energy in the secondary side metal that is above the secondary side liquid level is assumed to be released to the containment atmosphere over 24 hours. This energy was previously released during the first 3600 seconds. This is not a departure from the method described in Reference 3, because the secondary inventory is still assumed to release all available energy (i.e. down to 212°F and 14.7 psia) within the first 3600 seconds, and all SG energy is available for release. This is similar to the assumption made in the containment response for Watts Bar described in Reference 4 and approved by the NRC in Reference 5. The difference is the Watts Bar analysis neglects the addition of the energy from the metal in the SGs above the secondary liquid level for the duration of the long-term response.

There were no changes made to the containment response model other than the revised M&E input due to the change in the release of SG metal mass energy to the containment over 24 hours and the revised M&E input described above. The Gothic computer code (version 7.2a) was used consistent with the EPU analysis. The analysis results are in Table 1. This table compares the results obtained from the analyses performed in support of EPU and the analyses performed in support of this proposed amendment. During EPU the most limiting case was the double ended hot leg (DEHL) break for containment pressure, but with the changes to the M&E release inputs, the double ended pump suction (DEPS) break is most limiting for containment pressure. Both cases are analyzed with a LOOP and a single failure of one of the two emergency diesel generators (EDGs), denoted as minimum safeguards. The DEPS break was most limiting for containment temperature in the EPU analysis, and remains the most conservative. The peak containment pressure and temperature remain below the containment design pressure of 60.0 psig, and below the containment design and EQ maximum temperature of 286°F.

Table 1, LOCA Analysis Results

	Peak Pressure (psig)	Time (sec)	Peak Gas Temperature (°F)	Time (sec)	Pressure @ 24 hrs (psig)	Temp @ 24 hrs (°F)
DEHL (EPU)	54.21	15.52	280.1	15.02	N/A	N/A
DEHL (Proposed)	54.25	16.01	280.4	16.01	N/A	N/A
DEPS – MIN Safeguards (EPU)	53.88	1110	282.4	1110	7.77	159.4
DEPS – Min Safeguards (Proposed)	54.61	1220	283.6	1220	7.53	161.0

The pressure and temperature results of this analysis do not affect the offsite radiological consequences of a LOCA as previously analyzed in the UFSAR. The LOCA offsite radiological dose consequence analysis is based on the maximum allowable containment leakage rate of 0.2% by volume per day at 60 psig, and is decreased to 0.1% per day after 24 hours following a LOCA for the duration of the accident (30 days). This is consistent with the guidance provided in Regulatory Guide 1.183 (Reference 10). The containment pressure at 24 hours is significantly less than half of the peak pressure, which ensures the conservatism of the long-term leakage assumption. Since the maximum allowable containment leakage rate is not being revised, containment leakage assumed in the LOCA analysis is not impacted. Therefore, the increase in the calculated peak containment internal pressure does not impact the offsite radiological consequences of the LOCA accident analysis.

The pressure and temperature results of these analyses do not affect the analysis of radiological consequences of a LOCA with respect to radiological dose to the Control Room operators. Calculated Control Room operator dose during a LOCA is dependent on the maximum allowable containment atmosphere leakage rate and is unaffected by calculated peak containment internal pressure, as discussed above. Since the maximum allowable containment leakage rate is not being revised, dose to the Control Room operators is not affected by a change in peak containment pressure.

The pressure and temperature results of these analyses do not adversely affect environmentally qualified (EQ) equipment within Containment. The containment temperature response remains below the EQ profile (Reference 6, Section 3.11) until 24 hours. At this time the containment temperature is 9°F above the EQ temperature profile and drops below the profile within 2.3 days (of the event initiation). During EPU the temperature at 24 hours exceeded the EQ profile by 7°F for 1.3 days. During EPU a post-accident operating time (PAOT) calculation was completed using the Arrhenius equation that compared the long-term temperature profile from 24 hours to 30 days with the EQ profile by comparing equivalent times for a common temperature. This evaluation used a temperature of 120°F as a common temperature. The EQ Ginna accident profile, which is level at 152°F from 24 hours through 30-days, is equivalent to 74.43 days at 120°F. The EPU temperature profile was equivalent to 41.38-day duration at 120°F. With the increase to 125°F the temperature profile from 24 hours to 30 days is less than 43 days at 120°F. This remains significantly less than the EQ profile. Therefore, the small increase in temperature at 24 hours does not impact the qualification of equipment in containment.

The increase in normal allowable average air temperature in containment will not affect the EQ of equipment in containment. The equipment in containment has been qualified to a long-term normal operation containment temperature of 120°F. The increase to a maximum allowable containment average air temperature of 125°F will have no impact on the qualification of equipment in containment because

the time-weighted equivalent life temperature will be significantly less than 120°F. During EPU, equipment with time-limited aging analysis (TLAA) temperatures that are close to or lower than the building design temperature were identified for area temperature monitoring during operation to confirm the validity of the time-limited aging analysis. For these cases, monitoring either confirmed that the local temperature that has been established will ensure the qualified life of a component is longer than the end of extended operation or the appropriate replacement frequency has been established. Since the time-weighted equivalent life bulk air temperature will remain less than 120°F, then the time-weighted average local air temperatures will remain less than the peak air temperatures used in the TLAAs, and the conclusions of the TLAAs for extended operation will remain unchanged.

MSLB M&E Release and Containment Response

Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment that could produce high-pressure conditions for extended periods of time. The magnitude of the releases following a steam line rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. There are competing effects and credible single failures in the postulated accident scenario used to determine the worst cases for containment pressure following a steam line break.

The Ginna steam line break and containment response analysis considers a spectrum of cases that vary the initial power condition and the postulated single failure. The most recent licensing-basis steam line break analysis was completed for the extended power uprate (EPU). This analysis included changes for installation of automatically actuated main feedwater isolation valves (MFIVs) and a reduction in the minimum required shutdown margin for the core at the end of cycle conditions. That analysis received NRC approval via Reference 1.

The licensing-basis analysis presented in Reference 2, which was approved via Reference 1, accounted for many factors that influence the quantity and rate of the M&E release from a steam line break. These factors include initial power level, break size, and single failure. The current limiting steam line break containment pressure case, as documented in UFSAR Section 6.2.1.2.3 (Reference 6), is a 1.4 ft^2 break initiated from 70% power assuming a vital bus failure. This failure assumes a loss of one train of containment fan coolers and one CS pump. Only this most limiting case has been reanalyzed to support increasing the initial containment temperature to 125°F since the change impacts all cases the same based on the inputs that are being changed, i.e. containment air and structural heat sinks initial temperature. Also, the other cases had peak pressures that were considerably less than the limiting case, i.e. more than 2 psi lower than the peak case.

The analysis continues to utilize the computer codes approved in Reference 1. The RETRAN code (Version 2.0.2) is used for the secondary M&E release from the steam generators (SG), and GOTHIC (Version 7.2a) is used for the containment response analysis.

The analysis considered the increase in initial containment air temperature, containment heat sinks and SI accumulator temperature from 120°F to 125°F. The CS flow rate is reduced from a constant 1300 gpm to a spray flow as a function of containment pressure at reduced pump performance. The dose analyses and post-LOCA long term cooling analyses will continue to use 1300 gpm. The reduction in spray flow rate in the MSLB analysis is to provide for a potential future margin increase in the CS pumps. Table 2 shows the spray flow versus containment pressure curve used.

Table 2, Containment Spray Pump Performance

Containment Pressure (psig)	Spray Pump Flow (gpm)
60	1179
55	1203
50	1227
45	1250
25	1338

Due to the low margin in containment pressure resulting from the above changes, two additional inputs were adjusted from the previous analysis. In the letter dated April 29, 2005 (Reference 2) Ginna requested approval for a change to the main feedwater isolation valves (MFIVs) in support of the Ginna EPU. The current licensing basis MSLB analysis was presented in this letter. It stated that "the turbine-driven auxiliary feedwater (TDAFW) pump is assumed to be actuated due to a low-low SG level signal in both SGs for any case initiated from a power level above 50%." Plant experience has demonstrated that sufficient shrinkage occurs following a reactor trip that both SGs will reach the low-low SG level above 50% power. The analysis that supported this submittal actually used a more conservative start time for the TDAFW pump. The TDAFW pump was conservatively started coincident with the motor driven AFW (MDAFW) pumps for the limiting case. The MDAFW pumps start on the SI signal initiated from the low steam line pressure setpoint, which occurs tens of seconds prior to reaching the low-low SG level. Therefore, additional margin was gained in the limiting case by delaying the TDAFW pump start to coincide with the faulted SG reaching the low-low level setpoint. This reduces the mass available for release from the faulted SG. This change is in agreement with the assumptions in the analysis previously presented in Reference 2 and approved in Reference 1.

The second input change was to the SI system enthalpy. In the current licensing basis analysis approved in Reference 1, the enthalpy from the SI system corresponded to the maximum refueling water storage tank (RWST) temperature of 104°F and a pressure of 2250 psia. However, the SI pumps shutoff pressure is 1500 psia. Therefore, the SI system enthalpy was adjusted from 77.9 BTU/lbm to 75.9 BTU/lbm to correspond with 104°F and 1500 psia.

The limiting containment response analysis demonstrates that the peak pressure following a MSLB is 59.68 psig, which is below the containment design pressure 60.0 psig. This pressure is 0.09 psi higher than the EPU approved peak pressure. The maximum MSLB peak containment air temperature was calculated to exist for only a few seconds during the transient. Thermal analyses show that the time interval during which the containment air temperature peaked was short enough that the equipment surface temperatures remained below their design temperatures. Also, the equipment and cabling inside containment are protected against the direct effects of a SLB by concrete floors and shields. Therefore, it was concluded that the calculated transient containment air temperature following a LOCA remains limiting for environmental qualification (EQ) reasons.

The dose analysis approved for EPU in Reference 1 only considered the complete severance of the 36-inch main steam header outside containment inside the turbine building. Therefore, the changes to the MSLB in containment have no impact on the limiting MSLB dose analysis.

Large Break LOCA (LBLOCA) Core Response Analysis

The LBLOCA event is a major rupture of the reactor coolant pressure boundary (RCPB), which is defined as a breach in the RCPB with a total cross-sectional area greater than 1.0 ft^2.

A best-estimate (BE) LBLOCA analysis was completed to support the EPU. This analysis uses the ASTRUM methodology and received NRC approval in Reference 1. The results of this analysis are included in Table 3. Additional assessments were performed to determine the peak cladding temperature (PCT) impact on the R.E. Ginna BE LBLOCA analysis of HOTSPOT fuel relocation, fuel pellet thermal conductivity degradation (TCD), peaking factor burndown, and design input changes. The results were reported in Reference 14, and demonstrated that PCT was 2041°F.

The only change to the parameters for the BE LBLOCA evaluation for this amendment request is the maximum accumulator temperature, which is increased from 120°F to 125°F. An initial containment temperature of 90°F is assumed to determine a conservatively low containment backpressure, which is used to determine the break flow for the BE LBLOCA analysis; therefore, this input is not affected by the increase in maximum initial containment temperature from 120°F to 125°F.

The evaluation of an increase in maximum accumulator temperature at Ginna from 120°F to 125°F included re-execution of a subset of the analyses that included all PCT assessments performed for the Ginna BE LBLOCA analysis (HOTSPOT fuel relocation, fuel pellet TCD, peaking factor burndown, and design input changes). This evaluation resulted in a 75°F PCT penalty. The results of this evaluation, which considered all of the 10 CFR 50.46 acceptance criteria, are summarized in Table 4.

Table 3, Results of Ginna ASTRUM Analysis (EPU)

10 CFR 50.46 Requirement	Non-IFBA	IFBA
95/95 PCT (°F)	1,870	1,870
95/95 MLO (%)	2.89	3.43
95/95 CWO (%)	0.30	N/A

Note: IFBA denotes Integral Fuel Burnable Absorber, MLO denotes Maximum Local Oxidation, and CWO denotes Core Wide Oxidation.

Table 4, Results of Ginna Evaluation of the Increase in Maximum Accumulator Temperature

10 CFR 50.46 Requirement	Non-IFBA	IFBA
95/95 PCT (°F)	2,116	2,087
95/95 MLO (%)	7.38	5.70
95/95 CWO (%)	0.97	N/A
Note: This evaluation also includes all PCT assessments performed for the R.E. Ginna BE LBLOCA analysis (HOTSPOT fuel relocation, fuel pellet TCD and peaking factor burndown, and design input changes).		

Small Break LOCA (SBLOCA) Core Response Analysis

The SBLOCA event is a major rupture of the reactor coolant pressure boundary (RCPB), which is defined as a breach in the RCPB with a total cross-sectional area less than 1.0 ft^2.

Due to the increased containment temperature, the SI accumulator water temperature is also assumed to increase to 125°F. The current SBLOCA analysis-of-record (AOR) assumed an accumulator water temperature of 120°F (approved in Reference 1). An evaluation of the increase in accumulator water temperature to 125°F has been performed, which is based on the calculations completed using computer codes NOTRUMP and SBLOCTA approved in Reference 1.

The results of the analyses included in the Ginna UFSAR include the most limiting breaks 1.5-inch, 2-inch, and 3-inch diameter breaks. The results of break sizes 4, 5, 6, 8.75, and 9.75-inch diameter breaks have not been included in the UFSAR, but were included in Reference 1.

For the 1.5-inch break (1011°F PCT) and limiting 2-inch break (1167°F PCT), accumulator injection does not occur until (greater than 1000 seconds) after the peak cladding temperature (PCT) is reached; as such, the change in accumulator water temperature does not impact the PCT for these break sizes.

For breaks where accumulator injection is required to preclude core uncovery (4-inch break and larger), the integrated accumulator flow rate is more important than a small variation in accumulator water temperature. For these breaks, a small change in accumulator water enthalpy is far outweighed by the large amount of water mass entering the reactor coolant system, ultimately filling the downcomer and core. Therefore, the results will be minimally impacted, and the breaks less than 4-inches will remain limiting.

For the 3-inch break (1117°F PCT), the core begins to recover on accumulator injection, which occurs 75 seconds prior to the time of PCT. For SBLOCAs the timing of injection is the key parameter associated with the accumulators. The AOR conservatively assumes minimum accumulator cover pressure; therefore, there is no impact on the timing of initial accumulator injection. While a change in accumulator water enthalpy may cause minor changes in the transient response following accumulator injection and prior to the PCT time, as for the larger breaks, a minor change in accumulator injection enthalpy is outweighed by the beneficial water mass entering the system. Furthermore, the 5°F increase would negligibly impact the water enthalpy entering the downcomer and core, as the injecting accumulator water is mixing with cold leg fluid that is hundreds of degrees higher. Based on this, the core recovery response and cladding heat-up for the 3-inch break are negligibly impacted.

As denoted in Reference 1, break sizes intermediate to the integer sizes (for example, break sizes between 2 and 3 inches, and between 3 and 4 inches) can result in PCT increases by as much as 150°F. However, the NRC staff concluded in Reference 1 that, since the SBLOCA PCTs are very low (compared to the 10 CFR 50.46 limit of 2200°F) due to the high capacity of the Ginna high head safety injection (HHSI) pumps relative to the core power level and the high pressure of the accumulators, further analyses of breaks between 2 and 3 inches and 3 and 4 inches was not warranted. The 5°F increase in accumulator water temperature does not impact the 1.5-inch break and limiting 2-inch break, and only negligibly impacts breaks larger than 2-inches; therefore, the 2-inch break remains the limiting case for Ginna.

Post-LOCA Long-Term Cooling

The evaluation of post-LOCA long-term cooling consists of post-LOCA subcriticality and post-LOCA boric acid precipitation control.

The containment sump boron concentration model used for post-LOCA subcriticality analyses are performed in accordance with the Westinghouse Reload Safety Evaluation (RSE) methodology (Reference 12) and is based on the following:

- The calculation of the containment sump mixed-mean boron concentration assumes minimum mass and minimum boron concentrations for significant boron sources such as the SI accumulators and maximum mass and minimum boron concentration for significant dilution sources such as the Spray Additive Tank (SAT).
- Boron is uniformly distributed in the containment sump liquid. The post-LOCA sump inventory is made up of constituents that are equally likely to return to the containment sump; selective holdup in containment is neglected.
- The sump mixed-mean boron concentration is calculated as a function of the pre-trip Reactor Coolant System (RCS) conditions.
- There are no specific acceptance criteria when calculating the post-LOCA sump mixed-mean boron concentration. The resulting sump boron concentration, which is calculated as a function of the pre-LOCA RCS boron concentration, is reviewed for each cycle-specific core design to confirm that adequate boron exists to maintain subcriticality in the long-term post-LOCA

A post-LOCA subcriticality evaluation was performed using the Westinghouse RSE methodology (Reference 12) and the assumption that a 5°F increase in containment air temperature results in a 5°F increase in accumulator liquid temperature. The increase in temperature results in a small decrease in accumulator liquid mass. The resulting change in the sump mixed-mean boron concentration curve is negligible. Therefore, operation of Ginna with a maximum containment air temperature limit of 125°F is supported.

The results of the post-LOCA subcriticality evaluation will be reviewed each subsequent cycle in accordance with the Westinghouse Reload Safety Evaluation methodology (Reference 12) to ensure the post-LOCA core will remain subcritical upon entering the sump recirculation phase of ECCS injection.

4.0 REGULATORY SAFETY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

General Design Criterion 4, "Environmental and Dynamic Effects Design Bases," states that structures, systems and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including LOCAs.

General Design Criterion 16, "Containment Design," states that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

General Design Criterion 19, "Control Room," states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe shutdown under accident conditions, including LOCAs, and that adequate radiation protection shall be provided.

General Design Criterion 27, "Combined Reactivity Control System Capability," states that the reactivity control systems shall be designed to have a combined capability in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes to assure that under accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

General Design Criterion 35, "Emergency Core Cooling," states that a system will be provided with abundant emergency core cooling. The system safety function shall be to transfer heat from the reactor following any LOCA such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

General Design Criterion 38, "Containment Heat Removal," states that a system to remove heat from the reactor containment shall be provided that rapidly reduces, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable low levels.

General Design Criterion 50, "Containment Design Basis," states that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by § 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires that the following criteria be met in response to a LOCA: (1) peak cladding temperature is < 2200°F, (2) maximum cladding oxidation is less than < 0.17 times the total cladding thickness before oxidation, (3) maximum hydrogen generation is less than <0.01 (1%) times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, (4) calculated changes in core geometry shall be such that the core remains amenable to cooling, and (5) after any successful initiation of ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

10 CFR 54.21, "Contents of application – technical information," requires that the SSCs within the scope of life extension have an aging management review, which includes time-limited aging analyses be completed for the period of extended operation.

Although the Ginna licensing basis is the Atomic Industrial Forum (AIF) General Design Criteria, these General Design Criteria continue to be met following the revised containment temperature and pressure analyses. The EQ of equipment within Containment is not affected by these analyses. These analyses do not impact the maximum allowable containment leakage rate and therefore does not impact Control Room operator dose. The calculated peak containment pressure remains below containment design pressure. All 10 CFR 50.46 criteria continue to be met. The TLAA performed in accordance with 10 CFR 54.21 for extended life operation and EPU are not impacted by the change.

4.2 Precedent

This request is similar to the license amendments authorized by the NRC on March 27, 1990, for Indian Point Unit 2 (Reference 7), and on May 7, 1990, for Indian Point Unit 3 (Reference 8). These two amendments, in addition to approving an increase to the allowable containment air temperature, also

approved an increase to the Ultimate Heat Sink (UHS) temperature, which Ginna is not requesting. There are, however, differences in methodology (WCAP-12269 for Indian Point Unit 3, and the use of the code COCO for Indian Point Unit 2).

The change to the LBLOCA M&E release analysis to change the time that the energy from the SG metal mass above the secondary side liquid level is added to containment is similar to the assumption made in the M&E release analysis for Watts Bar described in Reference 4 and approved by the NRC in Reference 5. Watts Bar has an ice-condenser containment, and Ginna has a dry containment. However, this should not impact the M&E release, only the containment response. Ginna uses the same computer codes as Watts Bar for the long-term M&E release, i.e., SATAN-VI, WREFLOOD, and FROTH. Ginna also uses EPITOME as previously described. Watts Bar does not include the energy from this metal mass in the analysis, and Ginna includes it over a period of 24 hours, which is more conservative.

The changes other than the input noted above to the M&E release analysis is similar to the license amendment authorized by the NRC on January 19, 2012 for the Palisades Nuclear Plant (Reference 9). Both Ginna and Palisades used GOTHIC version 7.2a to analyze the LOCA containment response. Palisades used the code CEFLASH-4A to calculate M&E releases. Ginna has used the codes SATAN VI, WREFLOOD, FROTH, and EPITOME.

4.3 No Significant Hazards Consideration

R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) is proposing a license amendment to Renewed Operating License DPR-18 for the R.E. Ginna Nuclear Power Plant (Ginna), Technical Specification (TS) Section 3.6.5, "Containment Air Temperature." The proposed amendment would increase the allowable containment average air temperature from 120°F to 125°F.

Ginna LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to increase the containment average air temperature limit to 125°F, from 120°F, does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change. This proposed change will not affect radiological dose consequence analyses. The radiological dose consequence analyses assume a certain containment atmosphere leak rate based on the maximum allowable containment leakage rate, which is not affected by the change in allowable average containment air temperature resulting in a higher calculated peak containment pressure. The 10 CFR Part 50, Appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The acceptable leakage corresponds to the peak allowable containment pressure of 60 psig. The radiological dose consequence analyses assume a certain source term, which is not affected by the change in allowable average containment air temperature. All core limitations set forth in 10 CFR 50.46 continue to be met. The consequences of an accident previously evaluated will not be increased by this proposed change.

Therefore, operation of the facility in accordance with the proposed change to the containment average air temperature limit will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change provides for a higher allowable containment average air temperature to that currently in the TS Section 3.6.5. The calculated peak containment temperature and pressure remain below the containment design temperature and pressure of 286°F and 60 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Therefore, operation of the facility in accordance with the proposed change to TS Section 3.6.5 would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The calculated peak containment pressure and temperature remain below the containment design pressure and temperature of 60 psig and 286°F, respectively. The penalties applied to the BE LBLOCA analysis result in the limitations set forth in 10 CFR 50.46 continuing to be met. Since the radiological consequence analyses are based on the maximum allowable containment leakage rate, which is not being revised, the change in the calculated peak containment pressure and temperature and changes in core response do not represent a significant change in the margin of safety. The long-term impact of the peak containment temperature following a design basis accident exceeding the EQ profile by 2°F with respect to the current licensing basis is negligible.

   Therefore, operation of the facility in accordance with the proposed change to increase the allowable containment average air temperature from 120°F to 125°F does not involve a significant reduction in the margin of safety.

4.4 Conclusions

Ginna has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installed facility components located within the restricted area of the plant as defined in 10 CFR Part 20. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types

or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Safety Evaluation Related to Extended Power Uprate At R.E. Ginna Nuclear Power Plant, July 11, 2006. (ADAMS Accession Number ML061380249)

2. License Amendment Request Regarding Main Feedwater Isolation Valves, R.E. Ginna Nuclear Power Plant, Docket No. 50-244, April 29, 2005. (ADAMS Accession Number ML051260236)

3. WCAP-10325-P-A, Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version, May 1983.

4. WCAP-15699, Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, Containment Integrity Analyses for Ice Weight Optimization, Engineering Report, Revision 1, August 2001. (ADAMS Accession Number ML012700153)

5. Watts Bar Nuclear Plant Unit 1, Issuance of Amendment Regarding Reduction of Ice Condenser Ice Weight (TAC MB2969), November 29, 2001. (ADAMS Accession Number ML013330646)

6. R.E. Ginna Nuclear Power Plant, Updated Final Safety Analysis Report, Revision 23.

7. Indian Point Nuclear Generating Unit 2, Issuance of Amendment (TAC No. 73764), March 27, 1990. (ADAMS Accession Number ML003778344)

8. Indian Point Nuclear Generating Unit 3, Issuance of Amendment (TAC No. 78334), May 7, 1990. (ADAMS Accession Number ML003779051)

9. Palisades Nuclear Plant, Issuance of Amendment Re: Revise Calculated Peak Containment Internal Pressure (TAC No. ME6875), January 19, 2012. (ADAMS Accession Number ML113220370)

10. USNRC, Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000. (ADAMS Accession Number ML003731665)

11. NSAL-11-5, Westinghouse LOCA Mass and Energy Release Calculation Issues, July 25, 2011.

12. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985.

13. RGE-13-5, Westinghouse Inputs to License Amendment Request to Support Elevated Initial Containment and Accumulator Temperature at R.E. Ginna Nuclear Power Plant, Revision 0

14. Letter from T. Mogren (CENG) to U.S. NRC, ECCS 30-Day Report for Thermal Conductivity Degradation Impact on R.E. Ginna Large Break Loss of Coolant Accident Analysis with ASTRUM, dated August 16, 2012. (ADAMS Accession Number ML12233A621)

ATTACHMENT (2)

PROPOSED TECHNICAL SPECIFICATION CHANGE (MARKUP)

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ ~~120°F~~ 125°F

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	12 hours

ATTACHMENT (3)

PROPOSED TECHNICAL SPECIFICATION BASES CHANGE (MARKUP)

R.E. Ginna Nuclear Power Plant, LLC
February 28, 2013

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) and steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the Containment Spray (CS) and Containment Recirculation Fan Cooler (CRFC) Systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses to ensure that the total amount of energy within containment is within the capacity of the CS and CRFC Systems. The containment average air temperature is also an important consideration in establishing the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to the capability of the Engineered Safety Feature (ESF) systems to mitigate the accident, assuming the worst case single active failure. Consequently, the ESF systems must continue to function within the environment resulting from the DBA which includes humidity, pressure, temperature, and radiation considerations.

The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is ~~120°F~~. The postulated SLB accident results in maximum containment air temperatures that can exceed 350°F.

125°F

The initial temperature limit specified in this LCO is also used to establish the environmental qualification operating envelope for containment. The maximum SLB peak containment air temperature was calculated to exist for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses show that the time interval during which the containment air temperature peaked was short enough that the equipment surface temperatures remained below their design temperatures. Also, the equipment and cabling inside containment are protected against the direct effects of a SLB by concrete floors and shields. Therefore, it was concluded that the calculated transient containment air temperature following a LOCA (~~282.4°F~~) becomes limiting for environmental qualification reasons and is below the containment analysis criteria of the containment design temperature of 286°F.

283.6°F

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a SLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum allowable containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured and the OPERABILITY of equipment within containment is maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within the limit within 24 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 24 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. There are 6 containment air temperature indicators (TE-6031, TE-6035, TE-6036, TE-6037, TE-6038, and TE-6045) such that a minimum of three should be used for calculating the arithmetic average. The 12 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal containment temperature condition.

Calibration of these temperature indicators shall be performed in accordance with industry standards.

REFERENCES

1. UFSAR, Section 6.2.1.2.

2. 10 CFR 50.49.
