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April 29, 2016

10CFR50.59(d)(2)
10CFR72.48(d)(2)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

R.E. Ginna Nuclear Power Plant
Facility Operating License No. DPR-18
NRC Docket No. 50-244

Subject: Report of Facility Changes, Tests and Experiments Conducted Without Prior Commission Approval and Report of Commitment Changes Performed in Accordance with NEI 99-04

The subject report is hereby submitted as required by 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2). The enclosed report (Attachment 1) contains descriptions and summaries of the 10 CFR 50.59 or 10 CFR 72.48 evaluations conducted in support of proposed changes to the facility and procedures described in the UFSAR and special tests, from July 2014 through March 2016, performed under the provisions of 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2). Also included is a report with a summary of commitment changes performed in accordance with NEI 99-04, "Guidelines for Managing NRC Commitment changes" during the period from July 2014 through March 2016 that have not been previously provided to the NRC.

There are no new regulatory commitments contained in this letter.

Should you have any questions regarding this submittal, please contact Thomas Harding at (315) 791-5219.

Respectfully,

Joe Pacher

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Attachments: (1) Report of Facility Changes, Tests, and Experiments Conducted Without Prior NRC Approval from July 2014 through March 2016 under the Provisions of 10 CFR 50.59 or 10 CFR 72.48

(2) Report of Commitment Changes from July 2014 through March 2016 Performed in Accordance with NEI 99-04

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WPLNRC-1003072

cc: NRC Regional Administrator, Region 1
NRC Project Manager, Ginna
NRC Resident Inspector, Ginna

Attachment (1)

Report of Facility Changes, Tests, and Experiments Conducted Without Prior NRC Approval from July 2014 through March 2016, under the Provisions of 10 CFR 50.59 or 10 CFR 72.48

Report of Facility Changes, Tests, and Experiments Conducted Without Prior NRC Approval from July 2014 through March 2016 under the Provisions of 10 CFR 50.59 or 10 CFR 72.48

50.59 Evaluation No:	5059EVAL-2014-0001
Title of Change:	Containment Air Temperature Increase and Associated Changes
Implementation Document:	ECP-13-000048

Description of Change:
ECP-13-000048 increased the allowable containment average air temperature during normal operation from 120 °F to 125 °F, which was addressed in NRC safety evaluation for Ginna License Amendment 116. This current 50.59 evaluation specifically evaluates changes and impacts that were not addressed in the NRC safety evaluation for Ginna License Amendment 116.

Evaluation Summary:
The Ginna reactor containment average air temperature limit during normal operation was increased from 120°F to 125°F. The allowable minimum containment spray flow rate was reduced, and updates to the EQ and LBLOCA dose analyses were made to accommodate the former changes and inaccurate spray removal coefficients. The increase in temperature from 120°F to 125°F was subject to NRC approval. For the changes not approved by the NRC, evaluations have been completed to ensure all systems, structures and components that are in containment or interact with containment or the containment spray system will continue to perform the specified design function as described within the UFSAR. In addition, evaluations have demonstrated that no Design Basis limits will be exceeded, and the changes are consistent with UFSAR methods of evaluation.

Each of the eight questions in 10 CFR50.59(c)(2)(i) through 10 CFR 50.59(c)(2)(viii) were answered "NO" such that Nuclear Regulatory Commission permission was not required to make the change.

Report of Facility Changes, Tests, and Experiments Conducted Without Prior NRC Approval from July 2014 through March 2016 under the Provisions of 10 CFR 50.59 or 10 CFR 72.48

50.59 Evaluation No:	C-2015-001
Title of Change:	Turbine Stop, Intercept, and Reheater Stop Valve Testing Extension
Implementation Document:	ECP-15-000379

Description of Change:
The activity extends the operational testing interval of the high-pressure turbine stop valves and the reheater intercept and reheater stop valves from 12 months to a frequency of 440 days.

Evaluation Summary:
Ginna has established an UFSAR specified regulatory commitment to test high-pressure turbine stop valves, reheater intercept valves, and reheater stop valves annually (i.e. 12 months). MPR Calculation 0236-0070-01 concluded that the overall probability of a turbine missile ejection event as a function of 14.5 month testing (i.e. 441.3 days) is at the NRC acceptance criterion of 1E-05 (Yr.$^{-1}$). Regulatory Guide 1.115 requires that plants maintain an annual turbine missile probability less than 1E-05 (Yr.$^{-1}$). Therefore, turbine valve testing shall occur at a frequency of 440 days. Maintaining a turbine missile probability less than 1E-05 (Yr.$^{-1}$) provides assurance that the turbine overspeed protection system will function as designed thereby limiting the likelihood of overspeed past design conditions.

Each of the eight questions in 10 CFR50.59(c)(2)(i) through 10 CFR 50.59(c)(2)(viii) were answered "NO" such that Nuclear Regulatory Commission permission was not required to make the change.

Report of Facility Changes, Tests, and Experiments Conducted Without Prior NRC Approval from July 2014 through March 2016 under the Provisions of 10 CFR 50.59 or 10 CFR 72.48

50.59 Evaluation No:	C-2015-002
Title of Change:	Implementation of WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO"
Implementation Document:	ECP-15-000081

Description of Change:
This activity implements an updated Nuclear Regulatory Commission (NRC) approved Integral Form ZIRLO Cladding Corrosion Model as described in Westinghouse WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO," October 2013. The new model replaces the cladding corrosion model that was referred to in WCAP-15063-P-A, Revision 1, with Errata and more accurately reflects the temperature profile in the boiling region required to predict the measured oxides. The new clad corrosion model as described in the Topical Report will replace the current corrosion models one for one in the corrosion model applications. The new corrosion model will be used according to the currently licensed methodologies for Westinghouse NSSS plants.

Evaluation Summary:
The corrosion model described in WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A has been approved by the NRC. Ginna meets the limitations and conditions of Section 5.0 of the NRC SER for implementation of the new corrosion model for ZIRLO and Optimized ZIRLO cladding. The new clad corrosion model does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, and thus does not require NRC approval prior to its use.

The Evaluation concluded that the new clad corrosion model has been approved by the NRC for use on a forward fit basis, within the constraints described in the associated SER. WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A is acceptable for Ginna since the new clad corrosion model has been approved by the NRC for all Westinghouse NSSS plants that use ZIRLO for their cladding material with consideration of the conditions as described in Section 5.0 of the SER. Additionally, Technical Specification changes are not required since the Ginna Technical Specification Section 5.6.5 does not contain the corrosion model related topical report. The new corrosion model topical report does not define the technical basis for a Core Operating Limits Report (COLR) limit.

Each of the eight questions in 10 CFR50.59(c)(2)(i) through 10 CFR 50.59(c)(2)(viii) were answered "NO" such that Nuclear Regulatory Commission permission was not required to make the change.

Report of Facility Changes, Tests, and Experiments Conducted Without Prior NRC Approval from July 2014 through March 2016 under the Provisions of 10 CFR 50.59 or 10 CFR 72.48

50.59 Evaluation No:	C-2016-001
Title of Change:	Revision of Ginna SFP Criticality Analysis for EPU Conditions and Pu-241 Half-Life Corrections
Implementation Document:	ECP-15-000283

Description of Change:
The purpose of this activity is to revise calculation A-RGE-FE-003 Rev. 000 to include two conditions: 1) apply the current Pu-241 half-life, as established by the Journal of Radioanalytical and Nuclear Chemistry, to pre-Extended Power Uprate (EPU) burned fuel and document the impact of said change on Spent Fuel Pool (SFP) criticality and 2) apply the current Pu-241 half-life and updated isotopic number densities to EPU burned fuel and document the impact of said change on SFP criticality.

Evaluation Summary:
Although the revised analysis uses the same methodology as the existing analysis of record, changes to design inputs and elements of the methodology were required. The following is a list of design inputs and elements of the methodology that are impacted: the change from SCALE 4.3 to SCALE 4.4; application of the double contingency principle; changes to target Keff; EPU inputs; and the change in Pu-241 half-life. A 50.59 Evaluation was performed and concluded that the changes to the associated design inputs and elements of the methodology are acceptable as they do not result in a departure from a method of evaluation described in the UFSAR used to establish design bases or used in safety analyses. In addition as specified in the NRC Safety Evaluation related to Ginna Amendment No. 79, when analyzing SFP criticality the NRC allows for crediting the existing soluble boron within the SFP (i.e. 2300ppm maintained per TS 3.7.12). It was concluded in the aforementioned Safety Evaluation that the minimum SFP boron concentration value of 2300ppm required by TS 3.7.12 is more than sufficient to maintain Keff less than or equal to 0.95 for the most limiting accident condition within the pool. By virtue of the double contingency principle, which has been endorsed by the NRC, two unlikely independent and concurrent events are beyond the scope of the required analysis. Therefore, credit for the presence of the entire 2300ppm of soluble boron may be assumed in evaluating the most limiting accident conditions (i.e. fuel misplacement). As a result, the current total design basis soluble boron concentration for achieving a 95/95 value of Keff less than or equal to 0.95 (i.e. 975ppm) remains bounding for pre-EPU and EPU conditions using a Pu-241 half-life of 14.4 years. Based on these conclusions it was determined that this activity shall be implemented per plant procedures without obtaining a License Amendment.

Each of the eight questions in 10 CFR50.59(c)(2)(i) through 10 CFR 50.59(c)(2)(viii) were answered "NO" such that Nuclear Regulatory Commission permission was not required to make the change.

Attachment (2)

Report of Commitment Changes from July 2014 through March 2016 Performed in Accordance with NEI 99-04.

Report of Commitment Changes from July 2014 through March 2016 Performed In Accordance With NEI-99-04

Commitment Change Evaluation No.: 2015-01

Source Document: Ginna correspondence, WPLNRC-RG005481, "Letter from R. W. Kober, RG&E, to D. M. Crutchfield, NRC, Subject: Control of Heavy Loads, dated March 26, 1984," Section 2.4-1.

Original Commitment: Section 2.4-1 states, "RG&E has investigated the operation of the upper intermediate building monorail during plant operation and has restricted the use of this monorail to periods of time when the plant is in a cold shutdown mode.

Revised Commitment: The following exception is being made to the requirement to only use the monorail during cold shutdown: "The upper intermediate building monorail shall only be used during cold shutdown except for lifts that can be demonstrated to meet one of the following criteria, as required by NUREG-0612:

1. A single failure-proof handling system is utilized.
2. Analysis of the effects of a load drop ensures that a load drop will not cause a loss of safe shutdown equipment.
3. Procedures and interlocks are used to keep loads away from safe shutdown equipment."

Justification Summary:

Ginna committed to restricting use of the monorail in modes higher than cold shutdown in response to NRC Generic Letter 80-113 to ensure compliance with NUREG-0612. This broad restriction was made because the Safety Functions of the equipment below the monorail are only required in operational modes higher than cold shutdown. At the time the commitment was made, the monorail was not anticipated to be needed during any higher modes, since the majority of equipment inside the Intermediate Building is required to be in-service during higher modes.

This commitment change is being implemented to make an exception to this restriction, so that the monorail can be used during higher modes of operation. This exception maintains the requirements of NUREG-0612. The exception precludes the loss of Safety-Related equipment by load drops.