Exelon Generation.

10 CFR 50.59(d)(2)
10 CFR 72.48(d)(2)

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

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352, 50-353 and 07200065

Subject: 10 CFR 50.59 and 10 CFR 72.48 Evaluation 24-Month Summary Report for the Period July 1, 2014 through June 30, 2016

Attached is the 24-Month 10 CFR 50.59 and 10 CFR 72.48 Evaluation Summary Report for Limerick Units 1 and 2 for the period of July 1, 2014 through June 30, 2016, forwarded pursuant to 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2). The report includes brief descriptions of any changes, tests and experiments, including a summary of the evaluation of each. Four plant changes were approved and/or implemented using 10 CFR 50.59 Evaluations during this 24-month period. There were no plant changes implemented using 10 CFR 72.48 Evaluations during this 24-month period. The summaries of these changes are included in this report.

There are no regulatory commitments contained in this letter.

If you have any questions, please contact Robert B. Dickinson at (610) 718-3400.

Respectfully,

Richard W. Libra
Vice President – Limerick Generating Station
Exelon Generation Company, LLC

Attachment: Limerick Generating Station 10 CFR 50.59 and 10 CFR 72.48 Evaluation 24-Month Summary Report, 2016

cc: NRC Regional Administrator - Region I
NRC Senior Resident Inspector – Limerick Generating Station

ATTACHMENT
Limerick Generating Station

10 CFR 50.59 Evaluation and
10 CFR 72.48 Evaluation

24-Month Summary Report

2016

Note: This report summarizes 10 CFR 50.59 and 10CFR72.48 Evaluations that were approved between July 1, 2014 and June 30, 2016.

Title: Transition to GNF2 Fuel - Impact on EAB, LPZ and CR Doses
Unit Affected: 1&2
Year Implemented: 2015

Brief Description:
The activity is the implementation of GNF2 as a new fuel type at Limerick Generating Station Units 1 & 2 (LGS). The introduction of a new fuel type affects the source term used in the design basis analyses that determine Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and Control Room (CR) doses for accident conditions evaluated in UFSAR Chapter 15.

The evaluation for the activity used the post-Loss of Coolant Accident (LOCA) containment, Emergency Core Cooling System (ECCS), and Main Steam Isolation Valve (MSIV) leakage information and applicable leak rates specified in the LGS Technical Specifications, the Alternative Source Term (AST) methodology, and the Total Effective Dose Equivalent (TEDE) dose criteria approved in License Amendment Nos. 185 for Unit 1 and 146 for Unit 2 to the LGS Operating License for the LOCA, Fuel Handling Accident (FHA), Control Rod Drop Accident (CRDA) and Main Steam Line Break Accident (MSLBA).

Summary of Evaluation:
The change to radiological dose consequences does not result in operation of equipment outside the design functions as currently described in the Updated Final Safety Analysis Report (UFSAR). The new GNF2 fuel type will perform the same functions within the same operational limits as the current fuel types in use at LGS. The malfunctions and non-radiological accidents currently analyzed in the UFSAR are not affected by the GNF2 fuel. There are no new system interfaces created by the activity and no physical changes are made to the environment or release paths evaluated in the design analyses. As such, the activity does not increase the likelihood of a malfunction of equipment important to safety, does not create the possibility for an accident or malfunction of equipment important to safety of a different type than previously analyzed in the UFSAR and does not increase the frequency of accidents previously evaluated in the UFSAR.

The proposed activity does establish new offsite and control room radiological consequences for the LOCA, FHA, and CRDA; however, the revised dose consequences do not result in more than a minimal dose increase to comply with the provisions of 10 CFR 50.59. The Current Licensing Basis (CLB) dose consequences for the MSLBA are unchanged for the GNF2 fuel. In addition, the proposed activity will not result in a design basis limit for a fission product barrier being altered or exceeded. The evaluation methodology is consistent with the AST methodology and TEDE dose criteria approved in Units 1 and 2 Amendments 185 & 146 to the LGS Operating License. Based on the results of this review, the activity can be implemented without prior NRC review and approval.

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Title: On-Line NobleChem Injection Process Procedure Changes
Unit Affected: 1&2
Year Implemented: 2015

Brief Description:
This activity involves evaluating the previous revisions made to Chemistry procedures associated with the application of the On-Line NobleChem (OLNC) injection process and the required 10 CFR 50.59 Evaluation supporting the procedure revisions. The procedures were revised to incorporate the following changes associated with the OLNC injection process utilized at LGS Units 1&2:

1. Inject during normal plant operation (Mode 1) – previous procedure revision limited injection only during Hot Shutdown (Mode 3).
2. Injection through the Feedwater System – previous procedure revision injected through the Reactor Recirculation System.
3. Injection solution contains no Rhodium – previous procedure revision allowed the injection solution to contain Rhodium.
4. Injection frequency annually and at a lower concentration – previous procedure revision limited the injection frequency to once a cycle.

The OLNC injection process involves injecting a predetermined quantity of platinum solution into the reactor via the feedwater system. The injection is performed during normal plant operation (Mode 1, above 70% power and core flow above 85%), unlike the previous Noble Metals Chemical injection process which was only done during reactor hot shutdown (Mode 3). The subsequent deposition of platinum on wetted reactor internals, including existing cracks, is intended to help mitigate Intergranular Stress Corrosion Cracking (IGSCC).

These procedure revisions are consistent with the GE Hitachi Nuclear Energy's On-Line NobleChem Application Technical Safety Evaluation for LGS Unit 1 (NEDC-33786P, Rev. 000) and for LGS Unit 2 (NEDC-33680P, Rev. 000).

Summary of Evaluation:
Performance of OLNC injection during normal plant operation (Mode 1) provides multiple benefits as compared to during hot shutdown (Mode 3). The primary benefit is the ability to provide a more effective deposition of the solution into any cracks and crevices within the wetted reactor internals that will be opened further due to the higher operating temperature during Mode 1. Additionally, the OLNC injection during full power will result in dose savings in the drywell due to the transformation of oxygen rich oxide films into low oxygen oxide films on the reactor coolant wetted surfaces. Low oxygen oxide films are more tenacious and thinner than oxygen rich oxides films and they hold less Cobalt 60 and have lower dose rates than oxygen rich oxides. This is beneficial because Cobalt 60 is the primary source for the dose within the drywell during refueling outages. The performance of the OLNC injection process will also result in a savings of critical path time during a refueling outage when compared to during hot shutdown.

This activity will improve the effectiveness of the noble metals injection process while also reducing dose during refueling outages. Injection through the Feedwater System during normal plant operation will not adversely impact any of the UFSAR described Feedwater System design or operating functions. This activity will also have no adverse impact on the operation of any plant structure, system or component, (SSC) and is consistent with the LGS Units 1&2 design and licensing bases.

Additionally, per Electric Power Research Institute (EPRI), "BWR Vessel Internal Project On-Line Noble Metal Chemical Application Generic Technical Safety Evaluation" (BWRVIP-143, June 2005), injecting noble metal compounds into the reactor vessel and the process of injection such as OLNC does not affect the safety operations or the health and safety of the public.

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Title: Post Accident Monitor recorders upgrade
Unit Affected: 1&2
Year Implemented: 2016

Brief Description:
The activity evaluates Yokogawa model DX1004N series digital, paperless, Liquid-Crystal Display (LCD) recorder as replacement for existing obsolete Post Accident Monitoring (PAM) reactor water level and pressure recorders (Component ID's XR-042-1(2)R623A(B)). The existing PAM reactor water level and pressure recorder is Yokogawa Micro R1000 series digital strip chart recorder. The PAM recorders are safety related and designated as Reg. Guide 1.97, Category 1 recorders.

The existing digital recorder, Yokogawa Micro R1000 series, for XR-042-1(2)R623A(B) is no longer manufactured and spare parts are not readily available for its continued maintenance. Yokogawa model DX1004N is found to be a suitable and alternate replacement recorder.

Summary of Evaluation:
The replacement recorder has no adverse impact on plant operations, design bases, or safety analyses described in the UFSAR. The replacement recorder meets and/or exceeds all performance, design, and qualification requirements of the existing recorders. The replacement recorder meets the same design bases and requirements of Reg. Guide 1.97 as described in the UFSAR.

The replacement recorders perform the same design function as the existing recorders and do not affect any UFSAR described SSC design functions or controls. The activity does not involve any alternative evaluation methodology, revise or replace an element of a UFSAR described evaluation methodology. The activity does not involve any new tests or experiments not described in the UFSAR. The activity does not require a change to the Technical Specifications or affect a design basis limit for fission product barriers. However, the activity involves the use of two identical replacement model recorders in two redundant channels for monitoring and recording of reactor water level and pressure. Due to the potential for software common cause failures of this configuration, screening question number 1 has been conservatively answered "Yes", and the change has been evaluated further in the 50.59 evaluation process to ensure that the replacement components do not increase the likelihood of occurrence of a malfunction important to safety. Since the new recorder is determined to be rugged and reliable, and the change in failure rate is found to be negligible, the 50.59 evaluation concluded that there was less than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. Based on the screening and evaluation results, the proposed modification has no impact to the station licensing requirements/commitments.

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Title: Electro-Hydraulic Control System Upgrades
Unit Affected: 1&2
Year Implemented: Unit 1: 2014; Unit 2: 2015

This modification was reported complete on Unit 1 in the 2014 24-Month 10 CFR 50.59 Evaluation Summary Report and is being updated in this report to reflect implementation on Unit 2 in 2015.

Brief Description:

The existing Electro-Hydraulic Control (EHC) system, which has little redundancy and fault tolerance, is obsolete and no longer supported by the manufacturer. The EHC system has been a significant contributor to past turbine trips and plant transients. The implementation of a Digital EHC (DEHC) system will provide redundant control elements in the new control system, and are configured to allow operation with a single failure along with facilitating on-line replacement of a failed component. This will effectively eliminate many of the numerous single failure points that are part of the existing EHC system. The DEHC system has continuous self-diagnostics that will issue an alarm if a problem has been detected.

The activity is a configuration change for each Unit that implements an upgrade to the Pressure Regulator and Turbine-Generator Control System described in UFSAR Section 7.7.1.5. It was implemented on Unit 1 in 2014 (and included in the 2014 50.59 report) and the summary is being updated in this report to reflect implementation on Unit 2 in 2015. Refer to the 2014 24-Month 10 CFR 50.59 Evaluation Summary Report for additional information. This upgrade replaced the GE Mark I analog Turbine Control System (referred to as the EHC system) with a Westinghouse digital Turbine Control and Protection System (referred to as the DEHC System). The DEHC system utilizes an Ovation Based Distributed Control System that includes a Turbine Control System (TCS) and an Emergency Trip System (ETS), each consisting of redundant controllers, power supplies, I/O and testable dump assemblies. The DEHC System TCS performs the reactor pressure control, turbine speed and load control, system test functions and provides backup overspeed protection. The DEHC System ETS performs the primary turbine overspeed protection and all other turbine protection related functions.

Summary of Evaluation:
The following changes associated with the activity were judged to fundamentally alter the existing means of performing or controlling design functions and are reviewed in the 50.59 Evaluation:

- Analog to digital control since the digital controls contain different failure modes than the existing analog system.
- Conversion from hard controls to soft controls as it involves more than minimal differences in the Human Machine Interface.

- Change from functionally diverse turbine trip mechanisms to redundant electrically diverse trip mechanisms.

The 50.59 Evaluation determined that the proposed activity does not result in operation of equipment outside the design functions as currently described in the UFSAR. The turbine and steam bypass pressure control system will perform the same functions within the same operational limits with the DEHC system as previously required for the EHC system. The malfunctions and accidents currently analyzed in the UFSAR for the EHC system are bounding for the DEHC system. In addition, the proposed change does NOT create the possibility for an accident or malfunction of equipment important to safety of a different type than previously analyzed in the UFSAR. With increased redundancy and improved reliability, the DEHC system will NOT increase the frequency of accidents previously evaluated in the UFSAR and will NOT increase the likelihood of a malfunction of equipment important to safety. There are no new system interfaces created by the proposed control system upgrade and no physical changes to the steam path, turbine-generator or steam bypass system. The design does not alter or affect any ECCS system or barrier credited in mitigating the consequences of an accident. As such, the proposed activity does NOT increase the consequences of an accident or malfunction of equipment important to safety as previously analyzed in the UFSAR and will NOT result in a design basis limit for a fission product barrier being altered or exceeded.

It has been determined that the equipment / hardware changes, operating / maintenance procedure changes and modification / operability testing being implemented in conjunction with the proposed activity do not affect or alter the performance requirements or design function of the turbine and steam bypass pressure control system, Reactor Protection System, Turbine Generator or any other SSC as described in the UFSAR. These changes have no adverse effect on how any UFSAR described design function is performed or controlled, and no adverse impact on plant procedures or system operating parameters. There are no changes to any UFSAR described evaluation methodology or the use of an alternative methodology in establishing the design basis or safety analyses. The activity does not involve a test or experiment that would operate any SSC outside of its UFSAR described design function. A change to the TRM description of the overspeed trip system will be required but no changes to the Technical Specifications or Operating License are required.