Exelon Generation
200 Exelon Way
Kennett Square, PA 19348
www.exeloncorp.com

10 CFR 50.90

November 15, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

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

Subject: License Amendment Request – Main Steam Line Flow-High Isolation Response Time Change from ≤ 0.5 seconds to ≤ 1.0 seconds

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (Exelon) requests the following amendments to the Technical Specifications, Appendix A, of Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2.

The proposed amendments would revise the Technical Specification (TS) Table 3.3.2-3, "Isolation System Instrumentation Response Time," for the Main Steam Line Flow-High from ≤ 0.5 seconds to ≤ 1.0 seconds.

Attachment 1 provides the Evaluation of Proposed Changes. Attachment 2 provides the Proposed Technical Specification Marked-Up Pages. Attachment 3 provides the current, unrevised Updated Final Safety Analysis Report (UFSAR) Tables referenced in Attachment 1. Attachment 4 provides the General Electric Hitachi Nuclear Energy (GEH) Limerick Generating Station Main Steam Isolation Valve (MSIV) Response Time Testing Analysis.

The proposed changes have been reviewed by the LGS, Units 1 and 2, Plant Operations Review Committee and approved in accordance with Nuclear Safety Review Board procedures.

Exelon requests approval of the proposed amendments by November 15, 2014. Once approved, the amendments shall be implemented within 60 days.

There are no regulatory commitments contained in this request.

Using the standards in 10 CFR 50.92, "Issuance of amendment," Exelon has concluded that these proposed changes do not constitute a significant hazards consideration as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1).

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," Exelon is notifying the Commonwealth of Pennsylvania of this application for changes to the TS and Operating Licenses by transmitting a copy of this letter and its attachments to the designated state official.

Should you have any questions concerning this submittal, please contact Frank Mascitelli at (610) 765-5512.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of November 2013.

Respectfully,

James Barstow

James Barstow
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1) Evaluation of Proposed Technical Specification Changes
2) Proposed Technical Specification Marked-Up Pages
3) LGS UFSAR Tables 3.6-6, 3.6-7, 15.6-8, 15.6-9, 15.6-11 and UFSAR Figure 3.6-10 (For information only)
4) General Electric Hitachi Nuclear Energy (GEH) Limerick Generating Station Main Steam Isolation Valve (MSIV) Response Time Testing Analysis

cc: USNRC Regional Administrator, Region I
USNRC Project Manager, LGS
USNRC Senior Resident Inspector, LGS
Director, Bureau of Radiation Protection – PA Department of Environmental Resources

ATTACHMENT 1

EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

SUBJECT: Main Steam Line Flow-High Isolation Response Time Change from ≤ 0.5 seconds to ≤ 1.0 seconds

CONTENTS

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating Licenses NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2.

The proposed changes will revise Technical Specification (TS) Table 3.3.2-3, "Isolation System Instrumentation Response Time, Trip Function 1.d, Main Steam Line (MSL) Flow-High Isolation Response Time from ≤ 0.5 seconds to ≤ 1.0 seconds.

The purpose of this request is to address a decrease in response time testing margin identified during surveillance testing of the relay logic that initiates the MSL Flow-High isolation by using the available design basis margin within the Main Steam Line Break Accident Analysis.

Exelon Generation Company, LLC (Exelon) requests approval of the proposed changes. Once approved, the amendments shall be implemented within 60 days.

2.0 DETAILED DESCRIPTION

Background

LGS TS Table 3.3.2-3 provides response time requirements for the Isolation System Instrumentation. Surveillance Tests, ST-2-041-911 (908, 909, 910)-1/2 (Reference 1), are performed every 24 months to demonstrate the response times of the instrument channels. Each instrument channel consists of a flow transmitter, trip unit and associated DC and AC isolation and implementation relays. The Surveillance Tests (STs) ensure that the total instrument channel response time, as provided in TS Table 3.3.2-3, Item 1.d, Main Steam Line Flow-High Isolation, is satisfied.

The total response time for a main steam line isolation on high steam flow is 5.5 seconds. The Main Steam Isolation Valves (MSIVs) start to close at 0.5 seconds on a high flow signal and are fully closed at 5.5 seconds. The TS-required instrument response time for the MSIV closure initiation instrument channel on high steam line flow is 500 milliseconds (0.5 seconds). The TS Table 3.3.2-3 had been previously amended, in license amendments 132 and 93 (Reference 2) in December 1998 to eliminate response time testing of the flow transmitter in each main steam line flow instrument channel. Prior to the license amendment, the allotted allowable response time limit for the trip unit and the output, isolation and implementation relays was ≤ 184 milliseconds (msec), allowing 316 msec for the transmitter response time. Per the amendment, 355 msec has been allotted for the transmitter and, as such, the remaining 145 msec has become the TS required limit for the trip unit and the three relays of the instrument channel. An administrative limit (allotted time) of 135 msec is used to proactively identify for drift, and hence alert if the TS limit is being approached.

At the time of the December 1998 license amendment, the ST as-found instrument response time was 90-135 msec. In the mid-2000s, the ST as-found instrument response times increased to 110-140 msec. Due to the increasing trend in the response time, an investigation was initiated (Reference 3). The investigation concluded that the upper end of the vendor specified relay response time was not properly evaluated against the TS requirement when the 1998 TS amendment was implemented. This was verified through failure-analysis of the relays removed following each ST failure, and review of the logic circuitry for external contributors. Upon sensing a trip condition, the cumulative response time accounting for each component on the upper end of the vendor specification equals 199 msec. Therefore, for an instrument channel to meet the TS criteria of 145 msec, it is necessary to install only the fastest relays.

A review of the performance of the STs following the 1998 license amendment also indicates that the response times have drifted slightly in the slower direction with each performance of the response time ST while relays are still being supplied within the design specification. This is due, in part, to after each successive round of testing of, and selection from, the pooled inventory of relays, the remaining group of relays tend to be within the lower band of their response specification (i.e., slowest relays are selected last). There is not a performance problem with the relay supplier. The last ST performance resulted in several instrument channels exceeding the TS limits for more than one division of the trip system. The reduced margin in response time for the trip-unit and the relays that resulted from the 1998 license amendment contributed to the inability of the instrument channel to meet its TS Response Time Test (RTT) criteria. This resulted in LGS issuing LER 2012-008 (Reference 4).

Current Status

The initiative to pre-test relays dedicated for this particular application continues to select only those remaining relays (and newer relays ordered to meet inventory levels) with the fastest response times. However, since the obtained margin remains low, this was determined to be a temporary solution. For a long-term solution, the following three alternatives were considered:

- Continuing with the existing program of installing only the fastest relays by improving the process for selecting the fastest relays.

- Modifying the relay logic design with less number of relays, or a different type of relay, or replacing with a solid state logic system.

- Submitting a license amendment request to use the available margin in the Main Steam Line Break Accident Analysis by increasing the instrument channel response time from ≤ 0.5 seconds to ≤ 1.0 seconds.

Continuing the existing program of installing only the fastest relays has the risk for additional repeat failures of the MSL high flow isolation logic response times during surveillance testing while the unit is on-line, creating additional Limiting Conditions for Operation (LCO) entries. Recent bench testing of the relays indicate a decreasing trend of the fastest Agastat relays available. This option was judged as high risk and could result in possible unit shutdowns if multichannel relays failed their response times during 24-month surveillance testing. Modifying the existing components with different relays or a solid-state system will require identifying new locations that can house them in a controlled environment. This option was judged to be more complex and impactful to existing interfacing systems and would require the longest time frame to implement. Increasing the instrument channel response time from ≤ 0.5 seconds to ≤ 1.0 seconds was judged to be the safest option. It would be the least impactful to the station's existing systems and offered the timeliest solution to implement. This option could be implemented quickly in a year's time frame as compared to the modification option being implemented within a five-year time frame. Maintaining the existing program continues to be the highest human performance risk option, as replacing logic relays frequently with one of the inoperable instrument channels placed in the trip condition, while the unit is on-line, increases the risk for a spurious main steam line isolation at full power.

In addition, the alternative of submitting a license amendment request to eliminate RTT for the relay logic of the instrument channel under the NRC-approved Boiling Water Reactor (BWR) Owners Group Licensing Topical Report NEDO-32291-A Supplement 1 (Reference 5) was considered. The Topical Report reviewed utility-supplied information, in conjunction with industry failure experience and component specifications, requirements, and performance test results, to establish a bounding response time (BRT). Based on the relay logic LGS utilizes and considering the BRT, per Table 6-2 of the Topical Report, the instrument loop would require a response time of 444 msec with the sensor excluded. Since this would cause the cumulative response time for the instrument channel to exceed the TS-required response time limit of ≤ 0.5 seconds, the option was not pursued.

3.0 TECHNICAL EVALUATION

Main Steam Line Break Analysis

This license amendment request involves the Steam System Piping Break Outside Primary Containment (Main Steam Line Break (MSLB) Accident) as described in LGS Updated Final Safety Analysis Report (UFSAR) Section 15.6.4 (Reference 6). The MSLB Accident involves the postulation of a large steam line pipe break outside primary containment. It is assumed that the largest steam line instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features. This postulated accident represents the envelope evaluation of steam line failures outside primary containment. The MSLB Accident is evaluated for impact on the reactor fuel (Loss of Coolant Accident Peak Cladding Temperature (LOCA PCT)) and radiological consequences based on an assumed bounding reactor coolant fission product inventory.

The current analysis of record for the LGS MSLB LOCA PCT response is documented in G.E. Nuclear Energy, Limerick Generating Station, Units 1 and 2, SAFER/GESTR – LOCA, Loss-of-Coolant Accident Analysis, NEDC-32170P, Rev. 2, May 1995. The MSLB LOCA PCT response is not affected by the proposed change as the MSLB LOCA event sequence involves initiation of the Automatic Depressurization System (ADS) which occurs well after closure of the MSIVs. The MSLB LOCA PCT occurs after ADS initiation. No fuel damage is predicted as a result of this event.

The sequence of events and approximate time required to respond to the MSLB Accident are provided in UFSAR Table 15.6-8 (Attachment 3). The MSLB Accident Radiological Consequence Analysis Results are provided in UFSAR Table 15.6-11 (Attachment 3) and are unchanged. The existing analysis assumes the main steam isolation valves start to close at 0.5 seconds on high flow signal and are fully closed at 5.5 seconds. Initially, only steam will issue from the broken end of the steam line. The flow in each line is limited by critical flow at the flow limiter to a maximum of 200% of rated flow for each line. Rapid depressurization of the reactor pressure vessel causes the water level to rise, resulting in a steam/water mixture flowing from the break until the valves are closed. The total integrated mass leaving the reactor pressure vessel through the steam line break is 108,785 pounds mass (lbm) of which 88,333 lbm is liquid and 20,452 lbm is steam. For the radiological consequence evaluation, a total mass of 140,000 lbm is assumed.

A new MSLB safety analysis was performed by GE Hitachi Nuclear Energy (GEH) to evaluate the increase in the main steam line mass flow resulting from increasing the signal trip processing time from ≤ 0.5 seconds to ≤ 1.0 seconds. GEH performed the Main Steam Line Break Outside of Containment (STMO) analysis using the SAFER04A Engineering Computer Program (SAFER). SAFER is approved for use in STMO analyses examining fuel integrity and mass release. The SAFER simulation of STMO will provide not only fuel heat-up and long-term cooling system response, but also adequate simulation of break mass flow rate and integrated release for the purpose of dose consequence evaluation. There are no special events analyses (Anticipated Transient Without Scram, Fire Safe Shutdown, or Station Blackout) that consider main steam line breaks.

The calculated release from the new MSLB break flow analysis is based on a first principles thermo-dynamics transient methodology which computes the time varying break flow and reactor pressure and level response. This methodology is more accurate than the current method which is based on simplifying assumptions as described in UFSAR Section 15.6.4. The following assumptions and conditions were originally used in determining the mass loss from the primary system from the inception of the break to full closure of the MSIVs:

1. The reactor is operating at the power level associated with maximum mass release.
2. Nuclear system pressure is 1060 pounds per square inch absolute (psia) and remains constant during closure.
3. An instantaneous circumferential break of the main steam line occurs.
4. Isolation valves start to close at 0.5 seconds on high flow signal and are fully closed at 5.5 seconds.
5. The Moody critical flow model is applicable.
6. Level rise time is conservatively assumed to be 1 second. Mixture quality is conservatively taken to be a constant 7% (steam weight percentage) during mixture flow.

The following assumptions and conditions were used in the new MSLB analysis:

1. The reactor is operating at the lowest point on the Power/Flow diagram where a stable and converged solution is assured (Hot Standby condition -- 4% power and 35% core flow).
2. The initial nuclear system pressure is 1060 psia and as energy is removed from the system, SAFER calculates the pressure drop.
3. An instantaneous circumferential break of the main steam line occurs.
4. Isolation valves start to close at 1.0 seconds on high flow signal and are fully closed at 6.0 seconds.
5. The Appendix K break flow model (i.e., the Moody Slip Flow model) is used to maximize the break flow water mass.
6. High initial water level (Level 8) is conservatively assumed and results in higher liquid water mass release. SAFER is a systems code and calculates the time varying two phase flow during the event.

For the current instrument response time of 0.5 seconds, the new MSLB analysis predicts slightly less (~4.5%) total coolant mass release, although slightly more (~2.7%) water mass release.

A summary of the GEH analysis, "Limerick Main Steam Isolation Valve Response Time Testing Analysis," is provided in Attachment 4. The analysis concludes that the total coolant mass release leaving the reactor pressure vessel through the steam line break, under the most limiting scenario, is bounded by the basis documented in the UFSAR Section 15.6.4 (i.e., 140,000 lbm). The most limiting reactor conditions that produce the maximum total coolant mass release through the main steam line break is Startup/Hot-Standby condition (4% Power & 35% Flow). Table 1 below provides mass release values assuming a 0.5 second and a 1.0 second response time calculated utilizing SAFER, and can be compared to the original analysis of record mass flow values in Table 2 below based on UFSAR Section 15.6.4. These values had been established from Calculation LM-0644 (Reference 7).

Table 1, GEH STMO Results at Hot-Standby condition (4% Power & 35% Flow)

Level 8 0.5 Second Delay 5.0 Second Stroke	Water Mass Release:	90721 lbm
	Steam Mass Release:	13179 lbm
	Total Coolant Mass Release:	103900 lbm
Level 8 1.0 Second Delay 5.0 Second Stroke	Water Mass Release:	101562 lbm
	Steam Mass Release:	14138 lbm
	Total Coolant Mass Release:	115700 lbm

Table 2, LGS UFSAR 15.6.4

Water Mass Release:	88,333 lbm
Steam Mass Release:	20,452 lbm
Total Coolant Mass Release:	108,785 lbm

The change in the total coolant mass release of 6,915 lbm (115,700 lbm – 108,785 lbm) is well within the current available margin (~31,200 lbm) to the 140,000 lbm of reactor coolant bounding value (UFSAR Table 15.6-9, Attachment 3) used for the radiological consequence evaluation.

<u>High Energy Line Break (HELB) Evaluation</u>

In addition to the GEH MSLB Analysis discussed above, the impact of the increase in the main steam line break mass flow due to the increase in the instrument channel trip response time from ≤ 0.5 seconds to ≤ 1.0 seconds was evaluated for potential impacts on plant environmental conditions (i.e., compartment pressures and temperatures). The Limerick High Energy Line Break (HELB) Calculation -2006 (Reference 8) was revised to increase the MSLB HELB mass blowdown total duration by 0.5 seconds, from the current 6.5 seconds shown in UFSAR Table 3.6-6 to 7.0 seconds (Attachment 3).

The MSL mass blowdown used in the original MSL HELB calculation conservatively assumed that the MSIVs remained fully open for 5.0 seconds, as shown in UFSAR Figure 3.6-10 (Attachment 3). With the current 0.5-second instrument response time, the MSIVs will be fully closed at 5.5 seconds. However, an additional 1.0-second mass blowdown duration after MSIV closure is included in the analysis to conservatively account for the discharge of the "residual" steam/water mixture remaining in the broken pipes downstream of the MSIVs and results in a total blowdown duration of 6.5 seconds. As shown on UFSAR Figure 3.6-10 and Table 3.6-6 the mass blowdown flow is assumed to decrease linearly from 19,398 lbm/second at 5.0 seconds to 0.0 lbm/second at 6.5 seconds.

Based on the HELB calculation, conservatively assuming that the MSIVs do not begin to close after the current 0.5-second instrument response time but remain fully open for 5.0 seconds, there is no impact due to increasing the instrument response time to 1.0 seconds on the critical peak compartment temperatures and pressures determined in the calculation. As shown in UFSAR Table 3.6-7 (Attachment 3), the peak temperatures and pressures following a MSLB occur at approximately 1.0 seconds, except as discussed below. At approximately 1.0 seconds, the blowdown flow transitions from 100% steam with an enthalpy of 1192.4 BTU/lbm to a water/steam mix with an enthalpy of 595 BTU/lbm per UFSAR Table 3.6-6. The reduction in enthalpy at this time results in the compartment pressures and temperatures starting to fall from the previously obtained peak values.

The only compartments not reaching their peak temperatures at approximately one second are the main condenser area and steam venting plenum that contains the blowout panels that vents the main condenser area. The main condenser area is the largest compartment in the HELB model and, therefore, its temperature only reaches its peak temperature near the end of the blowdown flow through the area. The smaller steam venting plenum is downstream of the main condenser area and connected to the outside atmosphere so its temperature rise is driven by the temperature of the main condenser area and its temperature peaks near the end of the blowdown.

The proposed increase in the instrument response time does not impact the calculated peak pressures and temperatures that occur at approximately 1.0 seconds since the blowdown flow is not impacted until the MSIVs are assumed to start closing at 5.0 seconds. However, the increase in response time could have an impact on the overall duration of the blowdown. Therefore, Calculation -2006 was revised to include computer study runs, using CFLUD (Reference 9) conservatively assuming that the MSIVs remain fully open for 6.0 seconds (5.0 seconds + 1.0 seconds) and the total blowdown duration was increased from 6.5 seconds to 7.0 seconds. These runs confirmed that the critical peak temperatures and pressures did not change and that the only impact was a less than 4.0 degree Fahrenheit increase in the main condenser area and steam venting plenum peak temperatures. These two areas do not contain safety-related, environmentally qualified equipment and, therefore, this minimal increase in peak temperature has no adverse impact on the plant.

<u>Radiological Consequences Evaluation</u>

The current Analysis of Record for the MSLB radiological evaluation is documented in LGS Calculation LM-0644, Rev. 1, "Re-analysis of Main Steam Line Break Accident (MSLB) Using Alternative Source Terms," and UFSAR Section 15.6.4.

Regulation 10 CFR 50.67, "Accident Source Term," provides a mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design-basis accident analyses with an "Alternative Source Term" (AST). The methodology of approach to this replacement is given in NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and its associated NUREG 0800 Standard Review Plan (SRP) 15.0.1, "Radiological Consequences using Alternative Source Terms." LGS was approved for AST methodology via License Amendments 185 and 146 on August 23, 2006. (Reference 10)

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. However, the break mass released is taken for the dose calculations as a bounding maximized value for all current Boiling Water Reactor plants of 140,000 lbm of water, as provided in SRP 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)," for a GESSAR-251 plant. LGS is a GESSAR-251 plant. This value bounds for dose calculation purposes the historic UFSAR values, ensuring that the dose consequences are maximized and that the releases bound any other credible pipe break. Two activity release cases, corresponding to the pre-accident spike and maximum equilibrium concentration allowed by TSs of 4.0 microcuries/gram (μCi/gm) and 0.2 μCi/gm dose equivalent I-131 respectively, were assumed with inhalation Committed Effective Dose Equivalent (CEDE) dose conversion factors from Federal Guidance Report 11 conservatively used for normalized Dose Equivalent I-131 determination. The released activity assumptions are consistent with the guidance provided in Appendix D of Regulatory Guide 1.183.

The analysis assumes an instantaneous ground level release. For the control room dose calculations, the released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature (consistent with an assumption of no turbine building credit). This hemisphere is then assumed to move at a speed of one meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored. For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified methodology of Regulatory Guide 1.5, "Assumptions used for Evaluating the Potential Radiological Consequences for a Steam Line Break Accident for Boiling Water Reactors."

As discussed above, the MSLB dose calculation LM-0644 uses the SRP bounding mass blowdown value of 140,000 lbm. This input parameter is not impacted by the increase in the instrument response time as shown by the results of the GEH analysis provided in Table 1 above. The total coolant mass release of 115,700 lbm for the 1.0-second delay case is still well bounded by the 140,000 lbm input. Therefore, the MSLB accident doses for the Control Room, Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) determined in calculation LM-0644 and shown in UFSAR Table 15.6-11 remain unchanged. However, calculation LM-0644 is being revised to update the discussion of the basis for using 140,000 lbm as the bounding input to reflect the increase in the mass blowdown from 108,785 lbm to 115,700 lbm based on the GEH analysis.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Exelon Generation Company, LLC (Exelon) has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TSs. The following applicable regulations and regulatory requirements were reviewed in making this determination:

Codes:

10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants

> 50.49(b)(1)(i)(C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.
>
> 50.49(d)(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

10 CFR 50.67, Accident source term

> 10 CFR 50.67(b)(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:
>
> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

10 CFR 50 Appendix A General Design Criteria

Criterion 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 13 - Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 15 - Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 19 - Control room. 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 condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

Criterion 20 - Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 29 - Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Criterion 46 - Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

> 10 CFR 50.46(b)(1) Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.

Criterion 60 - Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Relevant Guidance:

Regulatory (Safety) Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," March 10, 1971.

U.S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982.

U.S. Nuclear Regulatory Commission Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

NUREG 0800 Section 15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR).

4.2 PRECEDENT

None

4.3 NO SIGNIFICANT HAZARDS CONSIDERATION

Exelon 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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No. The proposed increase in Main Steam Line (MSL) High Flow Isolation System Instrumentation Response Time from ≤ 0.5 seconds to ≤ 1.0 seconds does not involve a significant increase in the probability or consequences of an accident previously evaluated (i.e., Main Steam Line Break (MSLB)). GE Hitachi Nuclear Energy, using the SAFER04A Engineering Computer Program (SAFER), has performed an analysis of the impact to existing MSLB analysis using 1.0 seconds as the new response time input for the instrument channel high flow trip signal. The analysis concluded that for the worst case conditions, which is the Hot Standby initial operating condition, by increasing the instrument delay for Main Steam Line Isolation Valve (MSIV) actuation from 0.5 seconds to 1.0 seconds, the water mass release is increased by about 12%, the steam mass release is increased by about 8%, and the total coolant mass release increased by about 12% to 115,700 pounds mass (lbm). The major source of coolant activity which contributes to the released dose is contained in the coolant that is initially released in the liquid water phase. The enveloping total coolant mass release for radiological consequence evaluation is 140,000 lbm liquid; therefore, the MSLB total coolant mass release values calculated in this analysis remain bounded and the original MSLB Accident Dose Evaluation remains unchanged.

In regards to Peak Cladding Temperatures (PCT), the MSLB Accident is considered in evaluating a plant's response for fuel integrity and barrier protection to Loss of Coolant Accidents (LOCAs). Specifically, the MSLB Accident breaks either inside containment or outside containment are considered for fuel heat-up and neither scenario is limiting for Peak Cladding Temperature. The MSLB LOCA PCT response is not affected by the proposed amendment.

There are no special events analyses (Anticipated Transient Without Scram (ATWS), Fire Safe Shutdown, or Station Blackout) that consider main steam line breaks.

For building compartments that contain safety related equipment, the proposed increase in the instrument response time does not impact the calculated peak pressures and temperatures that occur at approximately 1.0 seconds since the blowdown flow is not impacted until the MSIVs are assumed to start closing at 5.0 seconds. However, the increase in response time could have an impact on the overall duration of the blowdown. The MSL High Energy Line Break (HELB) Analysis was revised to conservatively assume that the MSIVs remain fully open for 6.0 seconds (5.0 seconds + 1.0 seconds) and the total blowdown duration was increased from 6.5 seconds to 7.0 seconds. The revised HELB analysis confirmed that the critical peak temperatures and pressures did not change in building compartments containing safety related equipment and that the only impact was a less than 4.0-degree Fahrenheit increase in the main condenser area compartment and steam venting plenum compartment peak temperatures. These two compartments do not contain safety-related environmentally qualified equipment. Therefore, this minimal increase in peak temperature has no adverse impact on the plant.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No. The proposed increase in MSL High Flow Isolation System Instrumentation Response Time from ≤ 0.5 seconds to ≤ 1.0 seconds does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change only affects the primary containment isolation system response time, which is a mitigating system, for which the effects have been specifically evaluated for impact to the MSLB Accident and found to be acceptable. There are no special events analyses (ATWS, Fire Safe Shutdown, or Station Blackout) that consider main steam line breaks. The pressure and temperature of affected compartments do not affect the environmental qualification or performance of safety related equipment.

   The instrument channel logic delay time associated with this proposal was not postulated as an initiator of any previously analyzed accident, and is not expected to create any new system interactions, transient precursors, or failure modes of any structures, systems and components (SSCs). Thus, equipment important to safety will continue to operate as designed, and the proposed change will not result in any adverse conditions or any increase in challenges to safety systems.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No. The proposed increase in MSL High Flow Isolation System Instrumentation Response Time from ≤ 0.5 seconds to ≤ 1.0 seconds does not involve a significant reduction in a margin of safety. The proposed change will increase the total calculated total coolant mass release from 108,785 lbm to 115,700 lbm. The change in the total coolant mass release of 6,915 lbm is well within the current available margin (~31,200 lbm) to the 140,000 lbm bounding value used for the radiological consequence evaluation.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

4.4 CONCLUSIONS

In conclusion, based on the considerations discussed 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 be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. 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. Surveillance Tests (ST)-2-041-911-1/2 [(908, 909, 910)-1/2, for similar instrument channels], "NSSSS - Main Steam Line Flow - High Division IIB, Channel D Response Time Test (PDIS-41-2N686D, PDIS-41-2N687D, PDIS-41-2N688D, PDIS-41-2N689D)"

2. License Amendment No. 132 to Facility Operating License No. NPF-39 and Amendment No. 93 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2, "Limerick Generating Station, Units 1 and 2 (TAC NOS M96392 and M96393)", dated December 14, 1998 (ML011560614)

3. Equipment Apparent Cause Evaluation (EACE) 1412841-05, "Agastat relays in Main Steam Line (MSL) Flow-HI failed the Technical Specifications response time requirement," dated October 26, 2012

4. LER 2012-008-00 "Condition Prohibited by Technical Specifications due to Three Inoperable Isolation Instrumentation Channels," dated November 12, 2012

5. NEDO-32291-A Supplement 1, "BWR Owners' Group Licensing Topical Report, System Analyses for The Elimination of Selected Response Time Testing Requirements," dated October 1999

6. Limerick Generating Station Updated Final Safety Analysis Report, Rev 16, Section 15.6.4, "Steam System Piping Break Outside Primary Containment," dated September 2012

7. Calculation LM-0644, Rev. 1, "Re-analysis of Main Steam Line Break (MSLB) Accident Using Alternative Source Terms," dated September 20, 2005

8. MSL HELB Calculation -2006, Rev. 08, "Compartment Pressure & Temperature Analysis due to Pipe Break Outside Primary Containment"

9. CONCOIL-FLUD (CFLUD), Version 1.0, "Thermofluid Dynamics for a System of Interconnected Compartments," (Exelon DTSQA Log No. EX0001199)

10. License Amendment No. 185 to Facility Operating License No. NPF-39 and Amendment No. 146 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2 - Issuance of Amendments Re: Application of Alternate Source Term Methodology (TAC NOS. MC2295 and MC2296), dated August 23, 2006 (ML062210214)

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION MARKED-UP PAGES

(Units 1 and 2)

Page 3/4 3-23
(Marked-up)

Page 3/4 3-26
(For information only)

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	N.A.
2) Low, Low, Low - Level 1	≤1.0###*
b. DELETED	DELETED
c. Main Steam Line Pressure - Low	≤1.0###*
d. Main Steam Line Flow - High	≤1.0 ~~≤0.5~~###*
e. Condenser Vacuum - Low	N.A.
f. Outboard MSIV Room Temperature - High	N.A.
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
h. Manual Initiation	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low - Level 3	N.A.
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
c. Manual Initiation	N.A.
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCS Δ Flow - High	N.A.##
b. RWCS Area Temperature - High	N.A.
c. RWCS Area Ventilation Δ Temperature - High	N.A.
d. SLCS Initiation	N.A.
e. Reactor Vessel Water Level - Low, Low - Level 2	N.A.
f. Manual Initiation	N.A.

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TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
f. Deleted	
g. Reactor Enclosure Manual Initiation	N.A.
h. Refueling Area Manual Initiation	N.A.

TABLE NOTATIONS

(a) DELETED

(b) DELETED

* Isolation system instrumentation response time for MSIV only. No diesel generator delays assumed for MSIVs.

** DELETED

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to the isolation time for the valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\## With 45 second time delay.

\### Sensor is eliminated from response time testing for the MSIV actuation logic circuits. Response time testing and conformance to the administrative limits for the remaining channel including trip unit and relay logic are required.

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TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	N.A.
2) Low, Low, Low - Level 1	≤1.0###*
b. DELETED	DELETED
c. Main Steam Line Pressure - Low	≤1.0###*
d. Main Steam Line Flow - High	≤1.0 ~~≤0.5~~###*
e. Condenser Vacuum - Low	N.A.
f. Outboard MSIV Room Temperature - High	N.A.
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
h. Manual Initiation	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low - Level 3	N.A.
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
c. Manual Initiation	N.A.
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCS Δ Flow - High	N.A.##
b. RWCS Area Temperature - High	N.A.
c. RWCS Area Ventilation Δ Temperature - High	N.A.
d. SLCS Initiation	N.A.
e. Reactor Vessel Water Level - Low, Low - Level 2	N.A.
f. Manual Initiation	N.A.

JAN 07 1999

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
f. Deleted	
g. Reactor Enclosure Manual Initiation	N.A.
h. Refueling Area Manual Initiation	N.A.

TABLE NOTATIONS

(a) DELETED

(b) DELETED

* Isolation system instrumentation response time for MSIV only. No diesel generator delays assumed for MSIVs.

** DELETED

Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to the isolation time for the valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

With 45 second time delay.

Sensor is eliminated from response time testing for the MSIV actuation logic circuits. Response time testing and conformance to the administrative limits for the remaining channel including trip unit and relay logic are required.

ATTACHMENT 3

LGS UFSAR Tables 3.6-6, 3.6-7, 15.6-8, 15.6-9, 15.6-11

and UFSAR Figure 3.6-10

(For information only)

Table 3.6-6

BLOWDOWN DATA FOR HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

HIGH ENERGY LINE	BLOWDOWN: TIME AFTER BREAK (sec)	BLOWDOWN: MASS FLOW RATE (lb/sec)	BLOWDOWN: ENTHALPY (Btu/lb)[3]	ISOLATION VALVES	ISOLATION VALVE CLOSURE: VALVE CLOSING TIME (sec)	ISOLATION VALVE CLOSURE: SIGNAL DELAY TIME (sec)	ISOLATION VALVE CLOSURE: TOTAL INTERVAL (sec)
Main steam line	0.00	13,356	1192.4	HV41-F022A,B,C&D	5.0	0.5	5.5
(26 inch EBB-101, EBB-102,	0.076	8,543	1192.4	HV41-F028A,B,C&D	5.0	0.5	5.5
EBB-103, or EBB-104)	0.16	9,177	1192.4				
	1.0	15,972	1192.4				
	1.001	19,398	595.0				
	5.0	19,398	595.0				
	6.5	0	595.0				
RWCU suction line	0.0	2,820	525.3	HV44-F001	10.0 max.	2.0	12.0 max.
(6 inch DCC-103)	0.1	2,319	524.7	HV44-F004	10.0 max.	2.0	12.0 max.
	0.2	1,818	513.0	HV44-F039[1] -	-	-	
	0.3	1,486	509.8				
	0.4	1,200	506.3				
	0.5	1,027	503.7				
	0.6	905	501.3				
	0.7	791	497.4				
	0.8	723	495.0				
	0.9	701	491.0				
	1.0	638	485.1				
	1.26	563	485.0				
	6.0	563	485.0				
	9.0	0	485.0				
RWCU pump discharge line *	0.00	1410 *	526.4	HV44-F001	10.0 max.	2.0	12.0 max.
(4 inch DCC-101)	0.02	912 *	526.4	HV44-F004	10.0 max.	2.0	12.0 max.
	0.03	623 *	526.4	HV44-F039[1] -	-	-	
	0.11	473 *	526.4				
	0.25	688 *	526.4				
	0.37	583 *	526.4				
	2.00	531 *	526.4				
	15.00	273 *	526.4				

* Mass flow rate scale from values for 3" break to 4" break by using the ratio of the break areas as a scaling factor (A4"/A3") = 1.74.

Table 3.6-6 (Cont'd)

HIGH ENERGY LINE	BLOWDOWN TIME AFTER BREAK (sec)	MASS FLOW RATE (lb/sec)	ENTHALPY (Btu/lb)[3]	ISOLATION VALVES	ISOLATION VALVE CLOSURE VALVE CLOSING TIME (sec)	SIGNAL DELAY TIME (sec)	TOTAL INTERVAL (sec)
RWCU pump discharge line	0.0	1,164	526.4	HV44-F001	10.0 max.	2.0	12.0 max.
at inlet to regenerative	0.2	947	526.4	HV44-F004	10.0 max.	2.0	12.0 max.
heat exchanger	0.3	904	526.4	HV44-F039[1]	-	-	-
(4 inch DCC-101)	0.35	1,051	526.4				
	1.0	682	526.4				
	1.5	503	526.4				
	2.0	471	526.4				
	16.0	223	526.4				
RWCU pump discharge line	0.00	1,164	203.9	HV44-F001	10.0 max.	2.0	12.0 max.
at inlet to nonregenerative	0.34	1,056	203.9	HV44-F004	10.0 max.	2.0	12.0 max.
heat exchanger(4 inch DCC-102)	0.59	868	203.9	HV44-F039[1]	-	-	-
	0.59	868	125.2				
	1.00	677	131.4				
	3.72	464	149.4				
	3.72	464	316.2				
	16.00	464	316.2				
	16.00	224	91.0				
	20.00	224	91.0				
HPCI steam supply line	0.0	1,470	1192.4	HV55-F002	12.0	1.0	13.0
at turbine inlet valve	0.24	1,045	1192.4	HV55-F003	12.0	1.0	13.0
(12 inch EBB-108)	0.36	280	1192.4				
	13.0	280	1192.4				
	15.0	0	1192.4				
HPCI steam supply line	0.0	2,940	1192.4	HV55-F002	12.0	1.0	13.0
in piping area	0.11	1,958	1192.4	HV55-F003	12.0	1.0	13.0
(12 inch EBB-108)	0.14	1,594	1192.4				
	0.22	266	1192.4				
	13.0	266	1192.4				
	14.0	0	1192.4				

Table 3.6-6 (Cont'd)

HIGH ENERGY LINE	BLOWDOWN [2] TIME AFTER BREAK (sec)	MASS FLOW RATE (lb/sec)	ENTHALPY (Btu/lb)[3]	VALVES	ISOLATION VALVE CLOSURE VALVE ISOLATION (sec)	SIGNAL CLOSING TIME (sec)	TOTAL DELAY TIME INTERVAL (sec)
HPCI steam supply line	0.0	2,940	1192.4	HV55-F002	12.0	1.0	13.0
in isolation valve compartment	0.135	1,272	1192.4	HV55-F003	12.0	1.0	13.0
(12 inch EBB-108)	0.23	902	1192.4				
	0.475	328	1192.4				
	13.0	328	1192.4				
	14.0	0	1192.4				
RCIC steam supply line	0.0	380	1192.4	HV49-F007	7.2	1.0	8.2
at turbine inlet valve	0.311	168	1192.4	HV49-F008	7.2	1.0	8.2
(6 inch EBB-109)	0.43	40	1192.4				
	7.2	40	1192.4				
	8.2	0	1192.4				
RCIC steam supply line	0.0	760	1192.4	HV49-F007	7.2	1.0	8.2
in upper pipe tunnel	0.13	382	1192.4	HV49-F008	7.2	1.0	8.2
(6 inch EBB-109)	0.26	85.4	1192.4				
	0.302	42	1192.4				
	7.2	42	1192.4				
	8.2	0	1192.4				

(1) Valve closure time is not applicable for HV44-F039 since it is a check valve. This valve prevents backflow of water from the feedwater lines into the RWCU equipment compartments in the event of a break.

(2) The blowdown table is based on original power level. Environmental effects from blowdown are addressed based 3527 MWt conditions in Table 3.6-7 and 3.6-9, which bounds the operation at MUR power level of 3515 MWt.

(3) Enthalpy of system is conservatively assumed for saturated steam at 1000 psig (0% moisture carryover) based on original design conditions.

Table 3.6-7

PRESSURE-TEMPERATURE TRANSIENT ANALYSIS RESULTS FOR [6]
HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME [5] AFTER BREAK (sec)
A. Main Steam Line Break in Main Steam Tunnel				
1. Main steam tunnel	10.18	1.00	321	1.00
2. Main steam tunnel venting stack (lower region)	8.18	1.00	324	1.00
3. Main steam tunnel venting stack (mid-region)	5.32	1.01	325	1.00
4. Main steam tunnel venting stack (upper region)	2.93	0.14	319	1.02
5. Main condenser area	0.56[4]	0.21	179[4]	6.42
6. Steam venting plenum	0.61	0.18	182	6.38
B. Main Steam Line Break in Main Condenser Area				
4. Main condenser area	2.33	1.00	208	5.46
C. RWCU Suction Line Break in Penetration Room				
6. Nonregenerative heat exchanger room "A"	3.02	0.65	129[4]	0.51
7. Nonregenerative heat exchanger room "B"	3.02	0.64	128[4]	0.51
9. Regenerative heat exchanger room	2.92	0.65	127[4]	0.65
10. RWCU pump-room	2.91	0.38	105[4]	0.80
13. RWCU penetration room	2.92	0.40	202[4]	7.12

Table 3.6-7 (Cont'd)

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME[5] AFTER BREAK (sec)
D. RWCU Pump Discharge Line Break in Pump-Room				
6. Nonregenerative heat exchanger room "A"	0.44[4]	0.78	115[4]	0.78
7. Nonregenerative heat exchanger room "B"	0.45[4]	0.76	115[4]	0.76
9. Regenerative heat exchanger room	0.44[4]	0.74	112[4]	0.29
10. RWCU pump-room (B, C only)	1.96[4]	0.33	222	3.07
10A.RWCU pump room (A only)	6.2	0.33	229	3.07
13. RWCU penetration room	0.44[4]	0.87	206[4]	15.00
E. RWCU Pump Discharge Line Break in Regenerative Heat Exchanger Room				
6. Nonregenerative heat exchanger room "A"	2.45[4]	0.92	113[4]	0.11
7. Nonregenerative heat exchanger room "B"	2.45[4]	0.92	112[4]	0.11
9. Regenerative heat exchanger room	2.42[4]	1.01	221	6.65
10. RWCU pump-room	1.78[4]	1.13	113[4]	0.43
13. RWCU penetration room	1.77[4]	1.16	215	15.95
F. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "A"				
6. Nonregenerative heat exchanger room "A"	1.56[4]	10.18	221	16.00
7. Nonregenerative heat exchanger room "B"	1.06[4]	10.82	111[4]	0.10
9. Regenerative heat exchanger room	0.50[4]	15.44	208[4]	16.13
10. RWCU pump-room	0.36[4]	16.23	107[4]	0.38
13. RWCU penetration room	0.36[4]	16.20	183[4]	16.43

Table 3.6-7 (Cont'd)

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME[5] AFTER BREAK (sec)
G. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "B"				
6. Nonregenerative heat exchanger room "A"	1.07[4]	10.73	111[4]	0.10
7. Nonregenerative heat exchanger room "B"	2.03[4]	10.03	217[4]	16.00
9. Regenerative heat exchanger room	0.50[4]	15.55	208[4]	16.17
10. RWCU pump-room	0.36[4]	16.20	107[4]	0.40
13. RWCU penetration room	0.36[4]	16.23	183[4]	16.49
H. HPCI Steam Supply Line Break in HPCI Pump-Room				
17. HPCI pump-room	2.94	0.10	307	13.13
18. HPCI piping area	2.22[4]	0.07	308	13.08
21. Isolation valve compartment	1.03[4]	0.33	233[4]	14.00
I. HPCI Steam Supply Line Break in HPCI Piping Area				
17. HPCI pump-room	2.54[4]	0.11	151[4]	0.11
18. HPCI piping area	6.64	0.02	295[4]	5.67
21. Isolation valve compartment	1.52	0.21	205[4]	7.00
J. HPCI Steam Supply Line Break in Isolation Valve Compartment				
21. Isolation valve compartment	1.39[4]	0.19	273	13.62
22. Steam venting tunnel	0.84[4]	0.19	260[4]	13.18

Table 3.6-7 (Cont'd)

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME[5] AFTER BREAK (sec)
K. RCIC Steam Supply Line Break in RCIC Pump-Room				
19. RCIC pump-room	2.94	0.18	229	8.39
20. RCIC upper pipe tunnel	2.56[4]	0.16	213[4]	8.13
21. Isolation valve compartment	0.29[4]	0.34	126[4]	8.42
L. RCIC Steam Supply Line Break in RCIC Upper Pipe Tunnel				
19. RCIC pump-room	2.68[4]	0.14	153[4]	0.39
20. RCIC upper pipe tunnel	5.77	0.05	306	5.97
21. Isolation valve compartment	0.32[4]	0.21	139[4]	8.33

Table 3.6-7 (Cont'd)

PRESSURE-TEMPERATURE TRANSIENT ANALYSIS RESULTS FOR
HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

(Unit 2)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[3]	TIME[5] AFTER BREAK (sec)
A. Main Steam Line Break in Main Steam Tunnel				
1. Main steam tunnel	11.39	1.00	320	1.00
2. Main steam tunnel venting stack (lower region)	9.78	1.00	325	1.00
3. Venting stack security plenum	7.58	1.00	325	1.00
4. Main steam tunnel venting stack (upper region)	2.72	1.00	319	1.00
5. Main steam tunnel security plenum	10.30	1.00	320	1.00
6. Main condenser area	0.5[4]	0.20	182[5]	38.00
7. Steam venting plenum	0.54	0.20	187	40.00
B. Main Steam Line Break in Main Condenser Area				
4. Main condenser area	2.33	1.00	208	5.46
C. RWCU Suction Line Break in Penetration Room				
6. Nonregenerative heat exchanger room "A"	3.02	0.65	129[4]	0.51
7. Nonregenerative heat exchanger room "B"	3.02	0.64	128[4]	0.51
9. Regenerative heat exchanger room	2.92	0.65	127[4]	0.65
10. RWCU pump-room	2.91	0.38	105[4]	0.80
13. RWCU penetration room	2.92	0.40	202[4]	7.12

Table 3.6-7 (Cont'd)

(Unit 2)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[3]	TIME[5] AFTER BREAK (sec)
D. RWCU Pump Discharge Line Break in Pump-Room				
6. Nonregenerative heat exchanger room "A"	0.44[4]	0.78	115[4]	0.78
7. Nonregenerative heat exchanger room "B"	0.45[4]	0.76	115[4]	0.76
9. Regenerative heat exchanger room	0.44[4]	0.74	112[4]	0.29
10. RWCU pump-room(B, C only)	1.96[4]	0.33	222	3.07
10A.RWCU pump-room (A only)	6.2	0.33	229	3.07
13. RWCU penetration room	0.44[4]	0.87	206[4]	15.00
E. RWCU Pump Discharge Line Break in Regenerative Heat Exchanger Room				
6. Nonregenerative heat exchanger room "A"	2.4[4]	0.92	113[4]	0.11
7. Nonregenerative heat exchanger room "B"	2.45[4]	0.92	11[4]	0.11
9. Regenerative heat exchanger room	2.42[4]	1.01	221	6.65
10. RWCU pump-room	1.78[4]	1.13	113[4]	0.43
13. RWCU penetration room	1.77[4]	1.16	215	15.95
F. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "A"				
6. Nonregenerative heat exchanger room "A"	1.56[4]	10.18	221	16.00
7. Nonregenerative heat exchanger room "B"	1.06[4]	10.82	111[4]	0.10
9. Regenerative heat exchanger room	0.50[4]	15.44	208[4]	16.13
10. RWCU pump-room	0.36[4]	16.23	107[4]	0.38
13. RWCU penetration room	0.36[4]	16.20	183[4]	16.43

Table 3.6-7 (Cont'd)

(Unit 2)

COMPARTMENT[(1)]	PEAK[(2)] PRESSURE (psig)	TIME[(5)] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[(3)]	TIME[(5)] AFTER BREAK (sec)
G. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "B"				
6. Nonregenerative heat exchanger room "A"	1.11[(4)]	10.73	111[(4)]	0.10
7. Nonregenerative heat exchanger room "B"	2.11[(4)]	10.03	214[(4)]	16.00
9. Regenerative heat exchanger room	0.50[(4)]	15.55	206[(4)]	16.17
10. RWCU pump-room	0.38[(4)]	16.20	107[(4)]	0.40
13. RWCU penetration room	0.38[(4)]	16.23	184[(4)]	16.49
H. HPCI Steam Supply Line Break in HPCI Pump-Room				
17. HPCI pump-room	2.13[(4)]	0.24	290	11.00
18. HPCI piping area	1.52[(4)]	0.27	265[(4)]	6.1
21. Isolation valve compartment	1.04[(4)]	0.36	210[(4)]	16.00
I. HPCI Steam Supply Line Break in HPCI Piping Area				
17. HPCI pump-room	2.33	0.14	243[(4)]	14.00
18. HPCI piping area	2.35	0.13	299	0.14
21. Isolation valve compartment	1.36[(4)]	0.20	271	15.00
J. HPCI Steam Supply Line Break in Isolation Valve Compartment				
21. Isolation valve compartment	1.60	0.18	245[(4)]	14.00
22. Steam venting tunnel	1.11[(4)]	0.10	261	14.00

Table 3.6-7 (Cont'd)

(Unit 2)

COMPARTMENT[(1)]	PEAK[(2)] PRESSURE (psig)	TIME[(5)] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[(3)]	TIME[(5)] AFTER BREAK (sec)
K. RCIC Steam Supply Line Break in RCIC Pump-Room				
19. RCIC pump-room	1.76[(4)]	0.26	228	10.00
20. RCIC upper pipe tunnel	1.21[(4)]	0.27	198[(4)]	9.00
21. Isolation valve compartment	0.01[(4)]	0.02	122[(4)]	11.00
L. RCIC Steam Supply Line Break in RCIC Upper Pipe Tunnel				
19. RCIC pump-room	1.99	0.16	152[(4)]	0.13
20. RCIC upper pipe tunnel	2.47	0.03	280	0.18
21. Isolation valve compartment	0.1[(4)]	0.24	131[(4)]	20.00

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(1) Compartment numbers used in this table correspond to the compartment numbers used in the flow models (Figures 3.6-11, 3.6-12, 3.6-19, 3.6-23, 3.6-24, and 3.6-27).

(2) The compartment design pressures and the pressure-temperature transient analysis results are in Table 3.6-9.

(3) For Unit 2, design bulk temperatures may be less. Note also that temperatures in this area due to breaks elsewhere may be bounding.

(4) The valve shown is based on original power level. It is bounded by another break in this compartment where the valve shown is for the 3527 MWt power level.

(5) Time shown is based on original power level and was not recalculated for rerate since it is not used for any design basis evaluations.

(6) Valves shown unless noted are based on power level of 3527 MWt. They were established based on the original valves and a multiplier. The multiplier was calculated based on a maximum pressure increase associated with a power level of 3527 MWt and its impact on the blowdown and subsequent impact to subcompartment pressures and temperatures.

Table 15.6-8

SEQUENCE OF EVENTS FOR STEAM LINE BREAK
OUTSIDE PRIMARY CONTAINMENT

TIME (sec)	EVENT
0	Guillotine break of one main steam line outside primary containment.
0.5 (approx)	High steam line flow signal initiates closure of MSIV.
< 1.0	Reactor begins scram.
≤ 5.5	MSIVs fully closed.
60.0 (approx)	RCIC and HPCI initiate on low water level (Level 2) (RCIC considered unavailable, HPCI assumed single failure and therefore may not be available).
60.0 (approx)	SRVs open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1170 psi.
1780 (approx)	Low water level (Level 1) reached. Low pressure ECCS receives signal to start. ADS logic is initiated.
1900 (approx)	High drywell pressure bypass timer and ADS timer "timed out". ADS starts. Vessel depressurizes.
2100 (approx)	Low pressure ECCS begin injection. Core partially uncovers.
2160 (approx)	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

Table 15.6-9

MSLB – RADIOLOGICAL CONSEQUENCES
KEY INPUTS AND ASSUMPTIONS

Key MSLB Accident Analysis Inputs and Assumptions

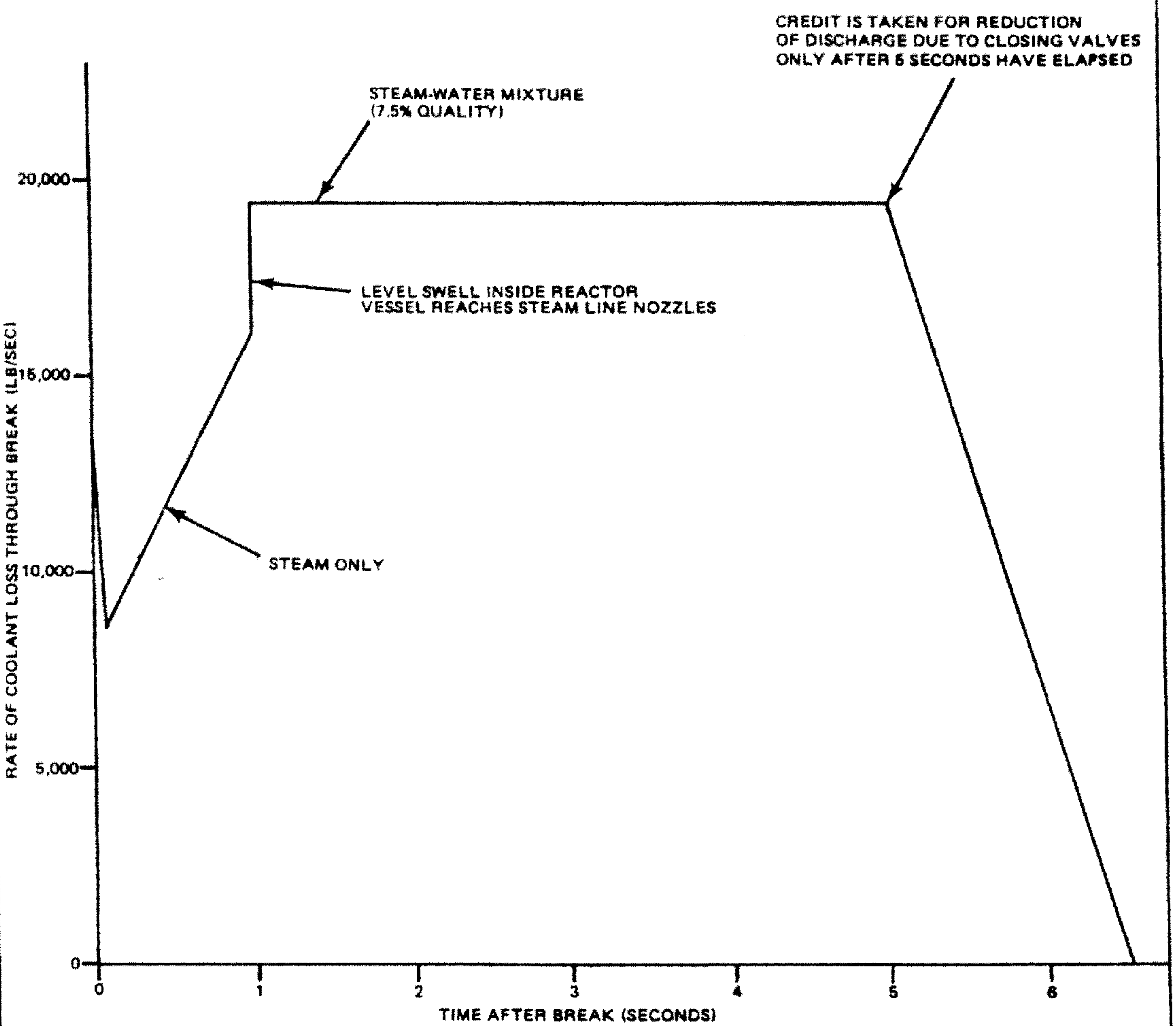
Input/Assumption	**Value**
Mass Release	140,000 lb_m of reactor coolant
Pre-Accident Spike Iodine Concentration	4 μCi/gm I-131 equivalent
Maximum Equilibrium Iodine Concentration	0.2 μCi/gm I-131 equivalent
Transport model for Control Room	Steam cloud moves past the Control Room intake at 1 m/sec
Control Room Filtration	No Credit Taken

Table 15.6-11

MSLB RADIOLOGICAL CONSEQUENCE RESULTS

MSLB Accident Radiological Consequence Analysis Results

		4 μCi/gm Dose Equivalent I-131 TEDE (rem)	**0.2 μCi/gm Dose Equivalent I-131 TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 day integrated dose	3.97	0.198	5
EAB	Worst 2-hour integrated dose	2.22	0.111	25 (4.0 μCi/gm) 2.5 (0.2 μCi/gm)
LPZ	30 day integrated dose	0.877	0.044	25 (4.0 μCi/gm) 2.5 (0.2 μCi/gm)



NOTE: This figure is based on the original power level. Environmental conditions resulting from the break based on a power level of 3527 MWt are listed in Tables 3.6-7 and 3.6-9.

LIMERICK GENERATING STATION
UNITS 1 AND 2
UPDATED FINAL SAFETY ANALYSIS REPORT

RATE OF COOLANT LOSS FOR MAIN STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT

FIGURE 3.6-10 Rev. 5, 4/96

ATTACHMENT 4

General Electric Hitachi Nuclear Energy (GEH) Limerick Generating Station

Main Steam Isolation Valve (MSIV) Response Time Testing Analysis

GE Hitachi Nuclear Energy

0000-0158-9651-NP
Revision 0
DRF Section 0000-0158-9651 R2
October 2013

Non-Proprietary Information – Class I (Public)

LIMERICK GENERATING STATION MAIN STEAM ISOLATION VALVE RESPONSE TIME TESTING ANALYSIS

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

Table of Contents

Tables

Acronyms and Abbreviations

Term	Definition
AOR	Analysis of Record
LOCA	Loss of Coolant Accident
LAR	License Amendment Request
LGS	Limerick Generating Station
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
RPV	Reactor Pressure Vessel
RTT	Response Time Test
STMO	Main Steam Line Break Outside of Containment
UFSAR	Updated Final Safety Analysis Report

1. Introduction

In support of a license amendment request (LAR) to revise the response time test (RTT) for the main steam line (MSL) high flow surveillance test, an evaluation is completed for the safety analyses affected by an increase in MSL high flow trip signal processing time from 0.5 second to 1.0 second. Several analyses incorporate a delay signal in main steam isolation valve (MSIV) closure initiation; however the principal affected design basis event for MSL high flow trip is the steam line piping break outside primary containment. An increase in the signal processing delay time that initiates MSIV closure may result in an increase of total released mass for the postulated event, thereby potentially leading to an increase in the calculated radiological exposures. This analysis evaluates the mass release and estimated exposure associated with a MSL flow trip signal processing time of 1.0 second and assures compliance with dose limitations provided by the Limerick Generating Station (LGS) Updated Final Safety Analysis Report (UFSAR).

2. Background

Of the many postulated scenarios considered in a boiling water reactor licensee's design basis and reported in a plant's Final Safety Analysis Report (FSAR, UFSAR, etc.), main steam line breaks (MSLBs) are considered in evaluating a plant's response for fuel integrity and barrier protection to loss of coolant accidents (LOCAs). Specifically, MSLBs either inside containment or outside containment are considered for fuel-heat up (neither scenarios are limiting for peak cladding temperature), MSLBs inside containment are considered for containment pressurization response and validating drywell equipment environmental qualification envelope, and MSLB outside of containment (STMO) discharge mass is used as an input value for radiological dose consequence evaluation. There are no special events analyses (anticipated transient without scram, fire safe shutdown, or station blackout) that consider MSLBs. The current analysis of record (AOR) for the STMO event is documented in Reference 1.

3. Method

The SAFER04A Engineering Computer Program, generally referred to as SAFER, is approved for use in STMO analyses (Reference 1) examining fuel integrity and mass release. The SAFER simulation of STMO will provide not only fuel heat up and long term cooling system response, but also provides adequate simulation of break mass flow rate and integrated release for the purpose of dose consequence evaluation.

The general approach taken in evaluating the mass release from a STMO break starts with an available SAFER basedeck and establishing a baseline for STMO analysis, then executing the STMO analysis as a baseline to the Reference 1 STMO analysis. The results are compared with the original analysis, and verified that the results are in good agreement with the AOR results.

The reference basedeck was then modified to update the analyzed fuel type to GNF2 fuel and various scenarios of MSIV closure time were analyzed based on the provided delay time information. Realistic and/or nominal input conditions for the analyses are applied. Analysis is performed with GNF2 fuel, however results for alternate GNF fuel types would be insignificantly different - affected by small changes in core coolant hydraulic parameters affecting core pressure drop. Core coolant thermodynamic behavior dominates during the duration of the STMO event for calculating integrated mass release, and the response is reasonably considered as being independent of fuel type.

The signal processing delay time for MSIV closure is changed from 0.5 second to 1.0 second in this analysis. In addition to different MSIV closure times, several sensitivities, including initial water level and operating condition, rated (100% power and 100% core flow) and hot standby condition (4% power and 35% core flow), sensitivities were performed to identify the bounding mass release condition.

4. Inputs and Assumptions

Generic Assumptions:

- Hot Standby: core power = 145 MWth, core flow = 35 Mlbm / hr.
- Rated: core power = 3,622 MWth, core flow = 100 Mlbm / hr.
- General Assumption A: The MSLB outside of containment is the limiting LOCA event for this analysis because mass released to outside the containment through the break is not terminated until closure of the main steam isolation valves occurs.
- General Assumption B: The Appendix K break flow model (i.e., the Moody Slip Flow model) is used to maximize the break flow water mass.

LGS Specific Assumptions (Assumptions derived from Reference 2, LGS UFSAR):

- LGS UFSAR 15.6.4.4.a: "The reactor is operating at the power level associated with maximum mass release". This statement seems to indicate the reactor is in RUN mode, operating at full power and rated flow. This analysis was performed at those conditions; however consideration may later be given to release from a break occurring while the reactor system is in STARTUP/HOT STANDBY mode. Evaluations for the dose response to steam line breaks outside containment have been analyzed for this hot standby condition, defined as the lowest point on the Power / Flow diagram where a stable and converged solution is assured.
- LGS UFSAR 15.6.4.4.b: "Nuclear system pressure is 1,060 psia and remains constant during the closure" This condition of constant pressure was necessary for calculating the mass flow rate and the total mass discharged in the direct approach calculation upon which the UFSAR analysis is based. The initial pressure for the SAFER analysis cases is 1060 psia and as energy is removed from the system the pressure drops. SAFER is a systems code and can calculate the time varying system pressure and mass release during the STMO event. The problem will be solved differently using the SAFER methodology therefore, it is not necessary to make the constant pressure assumption.
- LGS UFSAR 15.6.4.4.c: "An instantaneous circumferential break of the main steam line occurs." The Event is modeled in this way.
- LGS UFSAR 15.6.4.4.d: "Isolation valves start to close at 0.5 seconds on high flow signal and are fully closed at 5.5 seconds." This is the assumption that is being modified. It is the objective of this analysis to determine feasibility of beginning isolation at 1.0 seconds, close fully at 6.0 seconds, and remain bounded by the current UFSAR

analysis. For radiological consequence evaluation, a total integrated mass of 140,000 lbm liquid is assumed as specified in UFSAR 15.6.4.5.5."

- LGS UFSAR 15.6.4.4.e: "The Moody critical flow model is applicable." It is, as is stated in General Assumption B above.
- LGS UFSAR 15.6.4.4.f: "Level rise time is conservatively assumed to be 1 second. Mixture quality is conservatively taken to be a constant 7% (steam weight percentage) during mixture flow." This assumption is not necessary. SAFER is a systems code and can calculate the time varying two phase break flow during the STMO event.

5. Results

Tables 1 and 2 summarize the results for both rated power/flow and hot standby conditions, respectively. Results show that, as expected, high initial water level (Level 8) cases resulted in higher liquid water mass release and lower initial water level cases resulted in higher steam mass release. The total mass release is dominated by the liquid water mass release and hence the high initial water level cases are bounding.

6. Conclusions

As described in Section 5, the higher initial water level results in higher total mass release. It has also been shown in this study that the greatest water mass release and total mass release occur in hot standby, 4% power and 35 % flow, at Level 8.

In hot standby initial operating condition, there are fewer voids, therefore a greater initial water mass. Furthermore, the reactor is at a lower steaming rate due to lower power, therefore less feedwater is entering the downcomer. The downcomer and lower plenum are much closer to saturation. When the MSL ruptures, the reactor pressure vessel (RPV) is depressurized, and flashing occurs almost instantaneously throughout the entire RPV. This large two-phase inventory reaches the MSL considerably quicker than it would from a full power condition.

For hot standby initial operating condition, by increasing the instrument delay for MSIV actuation from 0.5 second to 1.0 second, the liquid water mass release increases by about 12%, whereas the steam mass release increases by about 8%. The major source of coolant activity which contributes to the released dose is contained in the coolant that is initially released in the liquid water phase. The enveloping total mass for radiological consequence evaluation is 140,000 lbm liquid, therefore the STMO total coolant mass discharge values calculated in this analysis remain bounded and do not depend on fuel type.

As discussed in Reference 3, GE Hitachi Nuclear Energy has now completed the evaluation of SC 12-18 R2 and has concluded that the total mass release calculated under the most limiting scenario is still bounded by the basis used in FSAR. Therefore, it is concluded that there is no effect from this issue on the LGS MSL high flow trip and hence this condition is not reportable under 10 CFR Part 21.

Table 1. STMO Results at Rated conditions (100% Power & 100% Flow)

Case	Mass Release	
Level 8	Water Mass Release:	33,896 lbm
0.5 Second Delay	Steam Mass Release:	26,064 lbm
5.0 Second Stroke	Total Coolant Mass Release:	59,960 lbm
Level 8	Water Mass Release:	42,025 lbm
1.0 Second Delay	Steam Mass Release:	27,545 lbm
5.0 Second Stroke	Total Coolant Mass Release:	69,570 lbm

Table 2. STMO Results at Hot-Standby condition (4% Power & 35% Flow)

Case	Mass Release	
Level 8	Water Mass Release:	90,721 lbm
0.5 Second Delay	Steam Mass Release:	13,179 lbm
5.0 Second Stroke	Total Coolant Mass Release:	103,900 lbm
Level 8	Water Mass Release:	101,562 lbm
1.0 Second Delay	Steam Mass Release:	14,138 lbm
5.0 Second Stroke	Total Coolant Mass Release:	115,700 lbm

7. References

1. GE Nuclear Energy, "Limerick Generating Station 1 & 2 SAFER/GESTR-LOCA Analysis," NEDC-32170P, Revision 2, May 1995.
2. Limerick Generating Station Updated Final Safety Analysis Report, Revision 16, September 2012.
3. GE Hitachi Nuclear Energy Safety Communication, "Error in Main Steam Line High Flow Calculation Methodology," SC 12-18 R2, February 2013.