Exelon Nuclear

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10 CFR 50.90

September 14, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 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: Response to Request for Additional Information
Technical Specifications Change Request -Type A Test Extension

Reference: Letter from P. B. Cowan (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, dated February 20, 2007

In the referenced letter, Exelon Generation Company, LLC (EGC) requested an amendment to Appendix A, Technical Specifications, of Facility Operating License Nos. NPF-39 and NPF-85. The proposed change modifies Technical Specifications (TS) 6.8.4.g, "Primary Containment Leakage Rate Testing Program." Specifically, the proposed change will revise TS 6.8.4.g to reflect a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) from 10 to 15 years. This one-time extension will require the Type A ILRT to be performed no later than May 15, 2013 (Unit 1) and May 21, 2014 (Unit 2).

Attached is our response to a request for additional information concerning this issue as discussed with the U. S. Nuclear Regulatory Commission staff on August 27, 2007.

There are no commitments contained in this letter.

If any additional information is needed, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 14th of September, 2007.

Respectfully,

Pamela B. Cowan

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1) Response to Request for Additional Information
2) Response to Questions 1 and 2 from the RAI for Limerick Generating Station, Units 1 and 2 ILRT Interval Extension Request

cc: R. R. Janati, Commonwealth of Pennsylvania
S. J. Collins, Administrator, Region 1, USNRC
S. Hansell, USNRC Senior Resident Inspector
P. Bamford, Project Manager, USNRC

ATTACHMENT 1

Technical Specifications Change Request -Type A Test Extension
Response to Request for Additional Information

ATTACHMENT 1
Technical Specifications Change Request -Type A Test Extension
Response to Request for Additional Information

QUESTION:

1. The approach used to assess the risk impact of the integrated leak rate test (ILRT) extension considered only internal events risk. As stated in Section 2.2.4 of Regulatory Guide 1.174, the risk-acceptance guidelines (in this case, for large early release frequency or LERF) are intended for comparison with a full-scope assessment risk assessment, including internal and external events. Consistent with this guidance, and to the extent supportable by the available risk models for Limerick Generating Station, provide an assessment of the impact of the requested change on ΔLERF and total LERF (based on the Nuclear Energy Institute Interim Guidance Methodology) when external events are included within the assessment.

RESPONSE:

See response contained in Attachment 2.

QUESTION:

2. In addition to extending the ILRT test interval from 10 to 15 years, the license amendment request would extend the drywell-to-suppression chamber bypass test (DWBT) interval to 15 years. The U.S. Nuclear Regulatory Commission has issued similar amendments to the operating licenses for Clinton, Susquehanna, and several other boiling-water reactor (BWR) plants. These interval extensions were based in part on a determination that the combined effect of both test interval extensions on risk was small. To provide insights into cumulative risk impacts, provide an assessment of the combined effect of the ILRT and DWBT interval extensions on risk (i.e., population dose, LERF, and conditional containment failure probability) similar to that provided for these other BWRs.

RESPONSE:

See response contained in Attachment 2.

QUESTION:

3. In Section 1.2, Background, last paragraph on page five of the risk impact assessment of your application you state, "Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment three times every 10 years. These requirements will not be changed as a result of the extended ILRT interval."

   Explain the relevance of this information, given that in your Attachment 1, Evaluation of Proposed Change, Reference (4) - U. S. Nuclear Regulatory Commission letter dated January 24, 2007, "Limerick Generation Station, Units 1 and 2 - Relief Requests I3R-01 For Alignment of Inservice Inspection and Containment Inservice Inspection (TAC NOS. MD2727 AND MD 2728)", you were approved to synchronize containment inspection programs in going from the ASME Code, Section XI, 1992 Edition through the 1992 Addenda to the 2001 Edition through the 2003 Addenda. Will LGS be applying Subsection IWE 1998 or later Edition throughout the requested ILRT interval? 10 CFR 50.55a(b)(2)(ix) reads:

(ix) *"Examination of metal containments and the liners of concrete containments.* Licensees applying Subsection IWE, 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda, shall satisfy the requirements of paragraphs (b)(2)(ix)(A) through (b)(2)(ix)(E) of this section. Licensees applying Subsection IWE, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, shall satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I) of this section."

RESPONSE:

10 CFR 50.55a(b)(2)(ix)**(E)** was the incorrect paragraph to reference. It is not applicable to the 1998 and later Editions of ASME Section XI. LGS will be applying Subsection IWE 1998 or later Editions throughout the requested ILRT interval. The current Containment Inservice Inspection (CISI) program utilizes the 2001 Edition through the 2003 Addenda of ASME Section XI. This Edition and Addenda will remain in effect for the current 10-year CISI interval, which is scheduled to end on January 31, 2017.

QUESTION:

4. In your LAR Attachment 1, Evaluation of Proposed Change, Section 4.2 Integrated Leak Rate History, on page three of 11 you provide historical ILRT results for each unit. Were there any changes in test or calculation methodology that might affect comparison of one test result to another for any of the tests for which results are listed? What were the corresponding combined Type B and Type C Test leakage rate totals, either in weight percent per day or volume/time with conversion factor to weight percent per day?

RESPONSE:

A review of the previous ILRT's was performed, and no changes were identified in the test or calculation methodology. The leakage rate was calculated using the Absolute Method as defined in ANSI/ANS 56.8. The following table identifies the applicable code year, and the combined Type B and Type C Test leakage rate totals in weight percent per day. The Type B and Type C leakage rate totals are based on Max Pathway Leakage.

LGS, Unit 1:

		LLRT Total	
ILRT Year	Calculation Method	Standard Cubic Centimeters per Minute (sccm)	Weight Percent Per day (%/day)
1984	Absolute Method Mass Point per 56.8-1981	23582	0.0752
1987	Absolute Method Mass Point per 56.8-1981	74245	0.2345
1990	Absolute Method Mass Point per 56.8-1981	55255	0.1795
1998	Absolute Method Mass Point per 56.8-1994	42047	0.1350

LGS, Unit 2:

ILRT Year	Calculation Method	LLRT Total	
		Standard Cubic Centimeters per Minute (sccm)	Weight Percent Per day (%/day)
1989	Absolute Method Mass Point per 56.8-1987	28928	0.0907
1993	Absolute Method Mass Point per 56.8-1987	41883	0.1304
1999	Absolute Method Mass Point per 56.8-1994	33278	0.1082

QUESTION:

5. Since the ILRT, the local leak rate test (LLRT), and containment inservice inspection (CISI) program collectively ensure leak-tight integrity and structural integrity of the containment, the NRC staff requests the following information to complete the review of the LAR.

   a. With reference to the fourth paragraph in Section 4.1 of Attachment 1 of the LAR, please indicate the current test intervals under Option B for the Type B and Type C LLRT. Please provide a schedule for the Type B and Type C tests on containment pressure-retaining boundaries that are or will be scheduled to be performed prior to and during the requested 5-year extension period.

RESPONSE:

The following tables identify the current Type B and C penetration test frequency, and those penetrations with non-metallic seals. The test frequencies are established based on performance utilizing the requirements of Option B. The test frequencies are re-evaluated after each refueling outage for potential changes.

Unit 1

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-200A	ST-4-LLR-B01-1	SUPPRESSION POOL ACCESS HATCH	24	X
X-200B	ST-4-LLR-B02-1	SUPPRESSION POOL ACCESS HATCH	24	X
X-002	ST-4-LLR-B03-1	EQUIPMENT ACCESS HATCH	24	X
X-001	ST-4-LLR-B04-1	EQUIPMENT ACCESS DOOR	24	X
X-004	ST-4-LLR-B05-1	HEAD ACCESS MANHOLE	24	X
X-006	ST-4-LLR-B06-1	CRD REMOVAL HATCH	24	X
X-002	ST-4-LLR-B07-1	PERSONNEL LOCK DOOR SEALS	24	X
N/A	ST-4-LLR-B08-1	DRYWELL HEAD SEALS	24	X
X-025	ST-4-LLR-B09-1	VALVE O-RINGS/PACKING - DRYWELL PURGE SUPPLY	60	X
X-025	ST-4-LLR-B10-1	VALVE O-RINGS - DRYWELL PURGE SUPPLY	60	X
X-025	ST-4-LLR-B11-1	VALVE O-RINGS/ PACKING - DRYWELL PURGE SUPPLY	60	X
X-026	ST-4-LLR-B12-1	VALVE O-RINGS - DRYWELL PURGE EXHAUST	60	X
X-026	ST-4-LLR-B13-1	VALVE O-RINGS - "A" POST LOCA RECOMBINER INLET	60	X

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-201A	ST-4-LLR-B14-1	VALVE O-RINGS - SUPPRESSION POOL PURGE SUPPLY	60	X
X-201A	ST-4-LLR-B15-1	VALVE O-RINGS - SUPPRESSION POOL PURGE SUPPLY	60	X
X-201A	ST-4-LLR-B16-1	VALVE O-RINGS -"B" POST LOCA RECOMBINER EXHAUST	60	X
X-202	ST-4-LLR-B17-1	VALVE O-RINGS - SUPPRESSION POOL PURGE EXHAUST	60	X
X-202	ST-4-LLR-B18-1	VALVE O-RINGS - "A" POST LOCA RECOMBINER EXHAUST	60	X
X-002	ST-4-LLR-B19-1	PERSONNEL AIR-LOCK FLANGE	24	X
JX-100A	ST-4-LLR-E01-1	ELECTRICAL PENETRATION	60	
JX-100B	ST-4-LLR-E02-1	ELECTRICAL PENETRATION	60	
JX-100C	ST-4-LLR-E03-1	ELECTRICAL PENETRATION	60	
JX-100D	ST-4-LLR-E04-1	ELECTRICAL PENETRATION	60	
JX-101A	ST-4-LLR-E05-1	ELECTRICAL PENETRATION	60	
JX-101B	ST-4-LLR-E06-1	ELECTRICAL PENETRATION	60	
JX-101C	ST-4-LLR-E07-1	ELECTRICAL PENETRATION	60	
JX-101D	ST-4-LLR-E08-1	ELECTRICAL PENETRATION	60	
JX-103A	ST-4-LLR-E09-1	ELECTRICAL PENETRATION	60	
JX-103B	ST-4-LLR-E10-1	ELECTRICAL PENETRATION	60	
JX-104A	ST-4-LLR-E11-1	ELECTRICAL PENETRATION	24	
JX-104B	ST-4-LLR-E12-1	ELECTRICAL PENETRATION	60	
JX-104C	ST-4-LLR-E13-1	ELECTRICAL PENETRATION	60	
JX-104D	ST-4-LLR-E14-1	ELECTRICAL PENETRATION	60	
JX-105A	ST-4-LLR-E15-1	ELECTRICAL PENETRATION	60	
JX-105B	ST-4-LLR-E16-1	ELECTRICAL PENETRATION	60	
JX-105C	ST-4-LLR-E17-1	ELECTRICAL PENETRATION	60	
JX-105D	ST-4-LLR-E18-1	ELECTRICAL PENETRATION	60	
JX-105E	ST-4-LLR-E19-1	ELECTRICAL PENETRATION	60	
JX-106A	ST-4-LLR-E20-1	ELECTRICAL PENETRATION	60	
JX-106B	ST-4-LLR-E21-1	ELECTRICAL PENETRATION	60	
JX-106C	ST-4-LLR-E22-1	ELECTRICAL PENETRATION	60	
JX-222	ST-4-LLR-E23-1	ELECTRICAL PENETRATION	60	
JX-230A	ST-4-LLR-E24-1	ELECTRICAL PENETRATION	60	
X-026,X-202	ST-4-LLR-003-1	"A" HYDROGEN RECOMBINER	24	
X-025,X-201A	ST-4-LLR-004-1	"B" HYDROGEN RECOMBINER	24	
X-003B	ST-4-LLR-011-1	INSTRUMENT GAS SUPPLY	24	
X-003D	ST-4-LLR-021-1	INSTRUMENT GAS SUPPLY	60	
X-007A	ST-4-LLR-031-1	MAIN STEAM LINE "A"	24	
X-007B	ST-4-LLR-041-1	MAIN STEAM LINE "B"	24	
X-007C	ST-4-LLR-051-1	MAIN STEAM LINE "C"	24	
X-007D	ST-4-LLR-061-1	MAIN STEAM LINE "D"	24	
X-008	ST-4-LLR-071-1	MAIN STEAM LINE DRAIN	24	
X-009A	ST-4-LLR-082-1	FEEDWATER VALVE HV-44-1F039	24	
X-009A	ST-4-LLR-084-1	"A" FEEDWATER	24	
X-009B	ST-4-LLR-092-1	"B" FEEDWATER	24	
X-010	ST-4-LLR-101-1	STEAM TO RCIC TURBINE	24	
X-011	ST-4-LLR-110-1	STEAM TO HPCI TURBINE	24	
X-012	ST-4-LLR-121-1	RHR SHUTDOWN COOLING SUPPLY	24	

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-013A	ST-4-LLR-131-1	"A" RHR S/D CLG RETURN	24	
X-013B	ST-4-LLR-141-1	"B" RHR S/D CLG RETURN	24	
X-014	ST-4-LLR-151-1	REACTOR WATER CLEANUP SUPPLY	24	
X-016A	ST-4-LLR-161-1	"A" CORE SPRAY PUMP DISCHARGE	24	
X-016B	ST-4-LLR-171-1	"B" CORE SPRAY PUMP DISCHARGE	24	
X-021	ST-4-LLR-191-1	SERVICE AIR SYSTEM	60	
X-023	ST-4-LLR-201-1	REACTOR ENCLOSURE COOLING WATER SUPPLY	60	
X-024	ST-4-LLR-211-1	REACTOR ENCLOSURE COOLING WATER RETURN	60	
X-025	ST-4-LLR-222-1	DRYWELL PURGE SUPPLY	24	
X-026	ST-4-LLR-231-1	DRYWELL PURGE EXHAUST	24	
X-026	ST-4-LLR-232-1	DRYWELL PURGE EXHAUST	24	
X-027A	ST-4-LLR-241-1	INSTRUMENT GAS SUPPLY	60	
X-028A	ST-4-LLR-261-1	RECIRC LOOP SAMPLE	60	
X-028A	ST-4-LLR-262-1	DRYWELL H2/O2 SAMPLE	24	
X-028B	ST-4-LLR-271-1	DRYWELL H2/O2 SAMPLE	24	
X-035B	ST-4-LLR-281-1	INSTRUMENT GAS TIP INDEX MECHANISMS	60	
X-035C	ST-4-LLR-291-1	TIP DRIVERS	60	X
X-035D	ST-4-LLR-301-1	TIP DRIVERS	60	X
X-035E	ST-4-LLR-311-1	TIP DRIVERS	60	X
X-035F	ST-4-LLR-321-1	TIP DRIVERS	60	X
X-035G	ST-4-LLR-331-1	TIP DRIVERS	60	X
X-038A-D	ST-4-LLR-361-1	SCRAM DISCHARGE VOLUME VENT AND DRAIN	24	
X-037A-B	ST-4-LLR-362-1	CRD CHARGING WATER HEADER	60	
X-037C-D	ST-4-LLR-363-1	CRD COOLING WATER HEADER	60	
X-038A-B	ST-4-LLR-364-1	CRD DRIVE WATER HEADER	60	
X-038C-D	ST-4-LLR-365-1	CRD EXHAUST CHARGING WATER HEADER	60	
X-039A	ST-4-LLR-371-1	DRYWELL SPRAY	24	
X-039B	ST-4-LLR-381-1	DRYWELL SPRAY	24	
X-040F	ST-4-LLR-391-1	INSTRUMENT GAS SUCTION	24	
X-040G	ST-4-LLR-401-1	ILRT DATA ACQUISITION SYSTEM	60	
X-040H	ST-4-LLR-411-1	INSTRUMENT GAS SUPPLY	24	
X-042,X-116	ST-4-LLR-421-1	STANDBY LIQUID CONTROL	24	
X-043B	ST-4-LLR-431-1	MAIN STEAM SAMPLE	60	
X-044	ST-4-LLR-441-1	RWCU ALTERNATE RETURN	24	
X-045A	ST-4-LLR-451-1	"A" RHR LPCI	24	
X-045B	ST-4-LLR-461-1	"B" RHR LPCI	24	
X-045C	ST-4-LLR-471-1	"C" RHR LPCI	24	
X-045D	ST-4-LLR-481-1	"D" RHR LPCI	24	
X-053	ST-4-LLR-491-1	DRYWELL CHILLED WATER SUPPLY	24	
X-054	ST-4-LLR-501-1	DRYWELL CHILLED WATER RETURN	24	
X-055	ST-4-LLR-511-1	DRYWELL CHILLED WATER SUPPLY	24	
X-056	ST-4-LLR-521-1	DRYWELL CHILLED WATER RETURN	24	
X-061A	ST-4-LLR-531-1	RECIRC PUMP A SEAL PURGE	24	
X-061A	ST-4-LLR-532-1	RECIRC PUMP B SEAL PURGE	24	
X-062	ST-4-LLR-541-1	H2/O2 SAMPLE RETURN	24	

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-117B	ST-4-LLR-561-1	D/W RAD MONITOR SUPPLY AND RETURN	60	
X-201A	ST-4-LLR-571-1	S/P PURGE SUPPLY	24	
X-025,X-201A	ST-4-LLR-572-1	S/P PURGE SUPPLY	24	
X-202	ST-4-LLR-581-1	S/P POOL PURGE EXHAUST	24	
X-202	ST-4-LLR-582-1	S/P PURGE EXHAUST	24	
X-205A	ST-4-LLR-651-1	"A" RHR S/P SPRAY	60	
X-205B	ST-4-LLR-661-1	"B" RHR S/P SPRAY	60	
X-217	ST-4-LLR-801-1	RCIC VACUUM PUMP DISCHARGE	60	
X-218	ST-4-LLR-811-1	INST GAS TO VACUUM RELIEF VLV	60	
X-221A	ST-4-LLR-831-1	WETWELL H2/02 SAMPLE	24	
X-221B	ST-4-LLR-841-1	WETWELL H2/02 SAMPLE	24	
X-227	ST-4-LLR-881-1	ILRT DATA ACQUISITION SYSTEM	60	
X-228D	ST-4-LLR-891-1	HPCI VACUUM RELIEF	60	
X-231A	ST-4-LLR-901-1	D/W SUMP DRAIN	24	
X-231B	ST-4-LLR-911-1	D/W SUMP DRAIN	60	
X-237-1	ST-4-LLR-941-1	S/P CLEAN-UP PUMP SUCTION	24	
X-238	ST-4-LLR-961-1	"B" RHR HEAT X SHELL VENT	24	X
X-239	ST-4-LLR-971-1	"A" RHR HEAT X SHELL VENT	24	X
X-240	ST-4-LLR-981-1	FLANGE O-RING SUPPRESSION POOL	24	X
X-241	ST-4-LLR-991-1	RCIC VACUUM RELIEF	24	

Unit 2

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-200A	ST-4-LLR-B01-2	SUPPRESSION POOL ACCESS HATCH	24	X
X-200B	ST-4-LLR-B02-2	SUPPRESSION POOL ACCESS HATCH	24	X
X-002	ST-4-LLR-B03-2	EQUIPMENT ACCESS HATCH	24	X
X-001	ST-4-LLR-B04-2	EQUIPMENT ACCESS DOOR	24	X
X-004	ST-4-LLR-B05-2	HEAD ACCESS MANHOLE	24	X
X-006	ST-4-LLR-B06-2	CRD REMOVAL HATCH	24	X
X-002	ST-4-LLR-B07-2	PERSONNEL LOCK DOOR SEALS	24	X
N/A	ST-4-LLR-B08-2	DRYWELL HEAD SEALS	24	X
X-025	ST-4-LLR-B09-2	VALVE O-RINGS/PACKING - DRYWELL PURGE SUPPLY	60	X
X-025	ST-4-LLR-B10-2	VALVE O-RINGS - DRYWELL PURGE SUPPLY	60	X
X-025	ST-4-LLR-B11-2	VALVE O-RINGS/ PACKING - DRYWELL PURGE SUPPLY	60	X
X-026	ST-4-LLR-B12-2	VALVE O-RINGS - DRYWELL PURGE EXHAUST	60	X
X-026	ST-4-LLR-B13-2	VALVE O-RINGS - "A" POST LOCA RECOMBINER INLET	60	X
X-201A	ST-4-LLR-B14-2	VALVE O-RINGS - SUPPRESSION POOL PURGE SUPPLY	60	X
X-201A	ST-4-LLR-B15-2	VALVE O-RINGS - SUPPRESSION POOL PURGE SUPPLY	60	X
X-201A	ST-4-LLR-B16-2	VALVE O-RINGS -"B" POST LOCA RECOMBINER EXHAUST	60	X
X-202	ST-4-LLR-B17-2	VALVE O-RINGS - SUPPRESSION POOL PURGE EXHAUST	60	X
X-202	ST-4-LLR-B18-2	VALVE O-RINGS - "A" POST LOCA RECOMBINER EXHAUST	60	X

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-002	ST-4-LLR-B19-2	PERSONNEL AIR-LOCK FLANGE	24	X
X-201A	ST-4-LLR-B20-2	TEST TAPS	24	
X-202	ST-4-LLR-B21-2	TEST TAPS	24	
X-026	ST-4-LLR-B22-2	TEST TAPS	24	
X-240	ST-4-LLR-B23-2	TEST TAPS	24	
JX-100A	ST-4-LLR-E01-2	ELECTRICAL PENETRATION	60	
JX-100B	ST-4-LLR-E02-2	ELECTRICAL PENETRATION	60	
JX-100C	ST-4-LLR-E03-2	ELECTRICAL PENETRATION	60	
JX-100D	ST-4-LLR-E04-2	ELECTRICAL PENETRATION	60	
JX-101A	ST-4-LLR-E05-2	ELECTRICAL PENETRATION	60	
JX-101B	ST-4-LLR-E06-2	ELECTRICAL PENETRATION	60	
JX-101C	ST-4-LLR-E07-2	ELECTRICAL PENETRATION	60	
JX-101D	ST-4-LLR-E08-2	ELECTRICAL PENETRATION	60	
JX-103A	ST-4-LLR-E09-2	ELECTRICAL PENETRATION	60	
JX-103B	ST-4-LLR-E10-2	ELECTRICAL PENETRATION	60	
JX-104A	ST-4-LLR-E11-2	ELECTRICAL PENETRATION	60	
JX-104B	ST-4-LLR-E12-2	ELECTRICAL PENETRATION	60	
JX-104C	ST-4-LLR-E13-2	ELECTRICAL PENETRATION	60	
JX-104D	ST-4-LLR-E14-2	ELECTRICAL PENETRATION	60	
JX-105A	ST-4-LLR-E15-2	ELECTRICAL PENETRATION	60	
JX-105B	ST-4-LLR-E16-2	ELECTRICAL PENETRATION	60	
JX-105C	ST-4-LLR-E17-2	ELECTRICAL PENETRATION	60	
JX-105D	ST-4-LLR-E18-2	ELECTRICAL PENETRATION	60	
JX-105E	ST-4-LLR-E19-2	ELECTRICAL PENETRATION	60	
JX-106A	ST-4-LLR-E20-2	ELECTRICAL PENETRATION	60	
JX-106B	ST-4-LLR-E21-2	ELECTRICAL PENETRATION	60	
JX-106C	ST-4-LLR-E22-2	ELECTRICAL PENETRATION	60	
JX-222	ST-4-LLR-E23-2	ELECTRICAL PENETRATION	60	
JX-230A	ST-4-LLR-E24-2	ELECTRICAL PENETRATION	60	
X-026,X-202	ST-4-LLR-003-2	"A" HYDROGEN RECOMBINER	24	
X-025,X-201A	ST-4-LLR-004-2	"B" HYDROGEN RECOMBINER	24	
X-003B	ST-4-LLR-011-2	INSTRUMENT GAS SUPPLY	60	
X-003D	ST-4-LLR-021-2	INSTRUMENT GAS SUPPLY	24	
X-007A	ST-4-LLR-031-2	MAIN STEAM LINE "A"	24	
X-007B	ST-4-LLR-041-2	MAIN STEAM LINE "B"	24	
X-007C	ST-4-LLR-051-2	MAIN STEAM LINE "C"	24	
X-007D	ST-4-LLR-061-2	MAIN STEAM LINE "D"	24	
X-008	ST-4-LLR-071-2	MAIN STEAM LINE DRAIN	60	
X-009A	ST-4-LLR-082-2	FEEDWATER VALVE HV-44-2F039	24	
X-009A	ST-4-LLR-084-2	"A" FEEDWATER	24	
X-009B	ST-4-LLR-092-2	"B" FEEDWATER	24	
X-010	ST-4-LLR-101-2	STEAM TO RCIC TURBINE	24	
X-011	ST-4-LLR-110-2	STEAM TO HPCI TURBINE	24	
X-012	ST-4-LLR-121-2	RHR SHUTDOWN COOLING SUPPLY	24	
X-013A	ST-4-LLR-131-2	"A" RHR S/D CLG RETURN	24	

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-013B	ST-4-LLR-141-2	"B" RHR S/D CLG RETURN	24	
X-014	ST-4-LLR-151-2	REACTOR WATER CLEANUP SUPPLY	60	
X-016A	ST-4-LLR-161-2	"A" CORE SPRAY PUMP DISCHARGE	24	
X-016B	ST-4-LLR-171-2	"B" CORE SPRAY PUMP DISCHARGE	24	
X-021	ST-4-LLR-191-2	SERVICE AIR SYSTEM	24	
X-023	ST-4-LLR-201-2	REACTOR ENCLOSURE COOLING WATER SUPPLY	60	
X-024	ST-4-LLR-211-2	REACTOR ENCLOSURE COOLING WATER RETURN	60	
X-025	ST-4-LLR-222-2	DRYWELL PURGE SUPPLY	24	
X-026	ST-4-LLR-231-2	DRYWELL PURGE EXHAUST	24	
X-026	ST-4-LLR-232-2	DRYWELL PURGE EXHAUST	24	
X-027A	ST-4-LLR-241-2	INSTRUMENT GAS SUPPLY	24	
X-028A	ST-4-LLR-261-2	RECIRC LOOP SAMPLE	60	
X-028A	ST-4-LLR-262-2	DRYWELL H2/O2 SAMPLE	60	
X-028B	ST-4-LLR-271-2	DRYWELL H2/O2 SAMPLE	60	
X-035B	ST-4-LLR-281-2	INSTRUMENT GAS TIP INDEX MECHANISMS	60	
X-035C	ST-4-LLR-291-2	TIP DRIVERS	60	X
X-035D	ST-4-LLR-301-2	TIP DRIVERS	60	X
X-035E	ST-4-LLR-311-2	TIP DRIVERS	60	X
X-035F	ST-4-LLR-321-2	TIP DRIVERS	60	X
X-035G	ST-4-LLR-331-2	TIP DRIVERS	60	X
X-038A-D	ST-4-LLR-361-2	SCRAM DISCHARGE VOLUME VENT AND DRAIN	24	
X-037A-B	ST-4-LLR-362-2	CRD CHARGING WATER HEADER	60	
X-037C-D	ST-4-LLR-363-2	CRD COOLING WATER HEADER	60	
X-038A-B	ST-4-LLR-364-2	CRD DRIVE WATER HEADER	60	
X-038C-D	ST-4-LLR-365-2	CRD EXHAUST CHARGING WATER HEADER	60	
X-039A	ST-4-LLR-371-2	DRYWELL SPRAY	24	
X-039B	ST-4-LLR-381-2	DRYWELL SPRAY	24	
X-040F	ST-4-LLR-391-2	INSTRUMENT GAS SUCTION	60	
X-040G	ST-4-LLR-401-2	ILRT DATA ACQUISITION SYSTEM	60	
X-040H	ST-4-LLR-411-2	INSTRUMENT GAS SUPPLY	24	
X-042,X-116	ST-4-LLR-421-2	STANDBY LIQUID CONTROL	24	
X-043B	ST-4-LLR-431-2	MAIN STEAM SAMPLE	60	
X-044	ST-4-LLR-441-2	RWCU ALTERNATE RETURN	24	
X-045A	ST-4-LLR-451-2	"A" RHR LPCI	24	
X-045B	ST-4-LLR-461-2	"B" RHR LPCI	24	
X-045C	ST-4-LLR-471-2	"C" RHR LPCI	24	
X-045D	ST-4-LLR-481-2	"D" RHR LPCI	24	
X-053	ST-4-LLR-491-2	DRYWELL CHILLED WATER SUPPLY	24	
X-054	ST-4-LLR-501-2	DRYWELL CHILLED WATER RETURN	24	
X-055	ST-4-LLR-511-2	DRYWELL CHILLED WATER SUPPLY	24	
X-056	ST-4-LLR-521-2	DRYWELL CHILLED WATER RETURN	24	
X-061A	ST-4-LLR-531-2	RECIRC PUMP A SEAL PURGE	24	
X-061A	ST-4-LLR-532-2	RECIRC PUMP B SEAL PURGE	24	
X-062	ST-4-LLR-541-2	H2/O2 SAMPLE RETURN	60	
X-117B	ST-4-LLR-561-2	D/W RAD MONITOR SUPPLY AND RETURN	60	

Penetration	Procedure	Description	Frequency (Months)	Non Metallic Seal
X-201A	ST-4-LLR-571-2	S/P PURGE SUPPLY	24	
X-025,X-201A	ST-4-LLR-572-2	S/P PURGE SUPPLY	24	
X-202	ST-4-LLR-581-2	S/P POOL PURGE EXHAUST	24	
X-202	ST-4-LLR-582-2	S/P PURGE EXHAUST	24	
X-205A	ST-4-LLR-651-2	"A" RHR S/P SPRAY	24	
X-205B	ST-4-LLR-661-2	"B" RHR S/P SPRAY	60	
X-217	ST-4-LLR-801-2	RCIC VACUUM PUMP DISCHARGE	24	
X-218	ST-4-LLR-811-2	INST GAS TO VACUUM RELIEF VLV	60	
X-221A	ST-4-LLR-831-2	WETWELL H2/02 SAMPLE	60	
X-221B	ST-4-LLR-841-2	WETWELL H2/02 SAMPLE	60	
X-225	ST-4-LLR-851-2	PENT X-240 O-RING	120	X
X-227	ST-4-LLR-881-2	ILRT DATA ACQUISITION SYSTEM	60	
X-228D	ST-4-LLR-891-2	HPCI VACUUM RELIEF	24	
X-231A	ST-4-LLR-901-2	D/W SUMP DRAIN	60	
X-231B	ST-4-LLR-911-2	D/W SUMP DRAIN	60	
X-237-1	ST-4-LLR-941-2	S/P CLEAN-UP PUMP SUCTION	24	
X-238	ST-4-LLR-961-2	"B" RHR HEAT X SHELL VENT	24	X
X-239	ST-4-LLR-971-2	"A" RHR HEAT X SHELL VENT	24	X
X-240	ST-4-LLR-981-2	FLANGE O-RING SUPPRESSION POOL	24	X
X-241	ST-4-LLR-991-2	RCIC VACUUM RELIEF	24	

QUESTION:

b. Section 4.4, Containment Inspections, in Attachment 1 of the LAR, do not provide any explicit discussion regarding the implementation of the Appendix J Option B general visual inspection requirements. In the LGS Technical Specifications, the Containment Leakage Rate Testing Program is based on RG 1.163. Regulatory Position C.3 of RG 1.163 specifies that visual examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test based on a 10-year ILRT interval. For LGS Units 1 & 2, please discuss your program for visual inspections (with schedule and methods) that meets this requirement. Please indicate with schedule how you would supplement this 10-year interval-based visual inspections requirement for the requested 15-year ILRT interval to ensure a continuing means of early uncovering of evidence of containment structural deterioration.

RESPONSE:

Regulatory Guide 1.163, Regulatory Position C.3 states,

> "Section 9.2.1, "Pretest Inspection and Test Methodology," of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration."

NEI 94-01, Section 9.2.1, "Pretest Inspection and Test Methodology", states,

> "Prior to initiating a Type A test, a visual examination shall be conducted of accessible interior and exterior surfaces of the containment system for structural problems which may affect either the containment structure leakage integrity or the performance of the Type A test. This inspection should be a general visual inspection of accessible interior and exterior surfaces of the primary containment and components."

Neither the RG 1.163 nor the NEI 94-01 document give additional details as to the definition of a general visual inspection. There is no methodology description given for a general visual inspection and no requirements listed that identify, for example, the acceptance criteria, inspector certification, evaluation, repair, replacement, or reporting criteria. Therefore, LGS has chosen to utilize the required examinations of the ASME Section XI CISI program, as modified by 10 CFR 50.55a, as the "visual inspection program". The ASME Section XI requirements can be considered equal to or superior than the requirements of the RG 1.163 Position C.3 and NEI 94-01 Section 9.2.1 on the basis that ASME Section XI defines requirements for performing containment visual inspections. For example, ASME Section XI defines the acceptance criteria, inspector certification, evaluation, repair, replacement, and reporting criteria.

The containment liner portion of the ASME Section XI CISI program (e.g., Exam Category E-A, Item No. E1.11) requires that a general visual examination be performed on 100% of the accessible surface areas during each inspection period. This requirement will not change as a result of this request. This includes mostly internal containment areas (e.g., the liner) but also includes external metallic containment (MC) components as well (e.g., portions of the MC classified flued heads that protrude through the concrete to the outside surface of containment). Therefore, the examination for MC components is performed from the inside and outside of containment, three times during the 10-year CISI interval. Since the previous CISI interval also contained the same exam frequency requirement, this examination will be performed more than three times in the proposed 15-year Type A test interval.

The concrete portion of the ASME Section XI CISI program (e.g., Exam Category L-A, Item No. L1.11) requires that a general visual examination be performed on all accessible surface areas once per 5 years. The examination of containment concrete (CC) components would be performed twice during the current 10-year CISI interval, and since the previous CISI interval also contained the same exam frequency requirement, this examination will be performed at least three times in the proposed 15-year Type A test interval.

With this ASME Section XI CISI Program, LGS has defined a "visual examination program" with methodology requirements, acceptance criteria, evaluation, repair, replacement, and reporting criteria that was not well defined by RG 1.163 and NEI 94-01. The three-time exam requirement (i.e., prior to initiating a Type A test, and during two other refueling outages before the next Type A test) will still be maintained by the ASME Section XI CISI program frequencies. Therefore, by utilizing the ASME Section XI requirements for containment inspections as modified by 10 CFR 50.55a, LGS will ensure that any evidence of structural deterioration is identified and/or corrected prior to the conduct of the Type A test.

QUESTION:

c. With reference to Section 4.4 (page 5) of Attachment 1 of the LAR, please describe with schedule and methods the IWE/IWL CISI program examinations that are or will be scheduled to be performed on containment pressure-retaining structures, systems and components prior to and during the requested 5-year extension period. This should also include your schedule and methods for examination and testing of seals, gaskets, moisture barriers and bolted connections associated with containment pressure boundary. Please provide this information for LGS Units 1 and 2. Also, indicate the dates when the most recent IWE examinations were completed for Unit 1.

RESPONSE:

The LGS CISI Program examinations are scheduled to be performed in accordance with the requirements of ASME Section XI IWE (Class MC components) and IWL (Class CC components). For example, Exam Category E-A, Item No. E1.11, for the MC components, require a general visual examination be performed on 100% of the accessible surface areas during each inspection period, and Exam Category L-A, Item No. L1.11, for the CC components, require that a general visual examination be performed on all accessible surface areas once per 5 years.

The 2001 Edition with the 2003 Addenda of ASME Section XI no longer requires that seals and gaskets be inspected. However, the Appendix J program still requires that leak rate testing be conducted on the applicable containment penetrations. The tables included in the response to question 5.a identify those penetrations with non-metallic seals.

The LGS containment design does not have a moisture barrier, and therefore no inspection is performed. Containment bolted connection examinations will be performed for the Second CISI Interval in accordance with ASME Section XI, Article IWE, as modified by 10 CFR 50.55a(b)(2)(ix)(H). Also, the most recent IWE examinations for Unit 1 were completed during the 1R11 refuel outage in March 2006.

QUESTION:

d. Please provide information of instances, if any, during implementation of the IWE/IWL CISI program at LGS Units 1 & 2 where existence of or potential for degradation conditions in inaccessible areas of the primary containment structure and metallic liners were identified and evaluated based on conditions found in accessible areas as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). If there were any instances of such conditions, please discuss the findings and actions taken.

RESPONSE:

No conditions have been found on either Unit 1 or Unit 2 that required an evaluation of the condition of the inaccessible areas in accordance with 10 CFR 50.55a(b)(2)(viii)(E) or 10 CFR 50.55a(b)(2)(ix)(A).

QUESTION:

e. Are bellows used on penetrations through containment pressure-retaining boundaries at LGS Units 1 & 2? If so, please provide information on their location, inspection, testing and operating experience with regard to detection of leakage through penetration bellows.

RESPONSE:

No bellows are used on penetrations through containment pressure-retaining boundaries.

ATTACHMENT 2

Response to Questions 1 and 2 from the RAI for Limerick Generating Station, Units 1 and 2 ILRT Interval Extension Request

RM DOCUMENTATION NO.	LG-2007-LAR-01	REV: 1	PAGE NO. 1

STATION: LIMERICK

UNIT(S) AFFECTED: 1 and 2

TITLE: Responses to RAIs 1 and 2 for the Limerick Unit 1 and Unit 2 ILRT Interval Extension Request

SUMMARY (Include UREs incorporated): The purpose of this analysis is to provide a response to requests for additional information from the NRC related to the risk assessment portion associated with implementing a one-time extension of the Limerick Unit 1 and Limerick Unit 2 containment integrated leak rate test (ILRT) and drywell to suppression chamber bypass test (DWBT) interval to 15 years. Revision 1 was necessary to accommodate minor editorial changes as a result of the Exelon review process.

Internal RM Documentation

Electronic Calculation Data Files:
(Program Name, Version, File Name extension/size/date/hour/min)

Prepared by: Donald E. Vanover / Donald E. Vanover / 9/10/07
Print / Sign / Date

Reviewed by: Robert J. Wolfgang / Robert J. Wolfgang / SEP 10 2007
Print / Sign / Date

Method of Review: [X] Detailed [ ] Alternate

This RM documentation supersedes: Rev. 0 **in its entirety.**

Approved by: Jeff R. Gabor / Jeff R. Gabor / 9/10/07
Print / Sign / Date

External RM Documentation

Reviewed by: N/A / /
Print / Sign / Date

Approved by: N/A / /
Print / Sign / Date

Do any ASSUMPTIONS / ENGINEERING JUDGEMENTS require later verification? **[ ] Yes [X] No**

Tracked By: AT#, URE# etc.)

RAI 1) The approach used to assess the risk impact of the integrated leak rate test (ILRT) extension considered only internal events risk. As stated in Section 2.2.4 of Regulatory Guide 1.174, the risk-acceptance guidelines (in this case, for large early release frequency or LERF) are intended for comparison with a full-scope assessment risk assessment, including internal and external events. Consistent with this guidance, and to the extent supportable by the available risk models for Limerick Generating Station, provide an assessment of the impact of the requested change on ΔLERF and total LERF (based on the Nuclear Energy Institute Interim Guidance Methodology) when external events are included within the assessment.

RAI 1 Response

External hazards were evaluated in the Limerick Generating Station (LGS) Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the LGS IPEEE study are documented in the LGS IPEEE Main Report [1]. The primary areas of external event evaluation at LGS were internal fire and seismic. The internal fire events were addressed by using the Fire Induced Vulnerability Evaluation (FIVE) methodology [2] and the seismic evaluations were performed in accordance with the EPRI Seismic Margins Analysis (SMA) methodology [6]. As such, there are no comprehensive CDF and LERF values available from the IPEEE to support the ILRT risk assessment.

In addition to internal fires and seismic events, the Limerick IPEEE analysis of high winds, external floods, transportation and nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that Limerick meets the applicable Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards. As such, these hazards were determined in the Limerick IPEEE to be negligible contributors to overall plant risk.

Since the performance of the IPEEE, an LGS fire PRA was completed in 2004. The EPRI FIVE Methodology [2] and Fire PRA Implementation Guide (FPRAIG) [3] screening approaches, EPRI Fire Events Database [4] and plant specific data were used to develop the LGS Fire PRA [5]. Based on the 2004 LGS fire PRA update, the LGS CDF contribution due to internal fires in the unscreened fire areas is calculated at 1.08E-5/yr for Unit 1 and 1.13E-5/yr for Unit 2. The fire PRA does not quantify the LERF risk measure, however, review of NUREG-1742 [7], indicates that the fire CDF for BWRs is primarily determined by plant transient type of events such that the LERF distribution from the fire CDF can be assumed to be similar to that from the internal events model.

The reported fire PRA CDF values are approximately a factor of three higher than the internal events CDF values. The fire CDF values are judged to be conservative given the methods employed in developing the fire PRA for Limerick when compared to the best estimate CDF and LERF values obtained from the internal events models. Given this, it is reasonable to assume that the total impact from external events risk is bounded by assuming a factor of three on the internal events evaluation. This assumption is used to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

Using the relationship described in the LAR submittal for LGS for the impact on 3b frequency due to increases in the ILRT surveillance interval, the EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT intervals are shown in Table 5.6-1 of the PRA analysis portion of the submittal as 1.01E-8/yr, 3.45E-8/yr, and 5.33E-8/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-per-15 years, including both internal and external hazard risk, is estimated as shown below:

	3b Frequency (3-per-10 yr ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase[(1)]
Internal Events Contribution	1.01E-8/yr	3.45E-8/yr	5.33E-8/yr	4.32E-8/yr
External Events Contribution	3.03E-8/yr	1.04E-7/yr	1.60E-7/yr	1.30E-7/yr
Combined (Internal + External)	4.04E-8/yr	1.38E-7/yr	2.13E-7/yr	1.73E-7/yr

Thus, the increase in LERF (ΔLERF) due to the combined internal and external events contribution is estimated as 1.73E-7/yr.

(1) Associated with the change from the current 3-per-10 year frequency to the proposed 1-per-15 year frequency.

NRC Regulatory Guide 1.174 [8], "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. As discussed in Section 2 of the PRA analysis in the submittal, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 1.7E-7/yr increase in LERF due to the combined internal and external events from extending the Limerick ILRT frequency from 3-per-10 years to 1-per-15 years falls into Region II between 1E-7 to 1E-6 per reactor year ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "Small Change" range, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above (i.e. a factor of three compared to the internal events results) is used for the total LERF estimate.

Per Table 2-2 of the PRA analysis portion of the submittal, the Limerick LERF due to internal event accidents is 6.8E-8/yr. With this information, the LERF due to external events is estimated based on the discussion above:

Internal Events LERF	6.8E-8/yr
External Events LERF	2.0E-7/yr
Internal Events LERF due to ILRT (at 15 years)	5.3E-8/yr
External Events LERF due to ILRT (at 15 years)	1.6E-7/yr
Total:	4.8E-7/yr

As can be seen, the total LERF for Limerick is estimated at 4.8E-7/yr, which is less than the RG 1.174 requirement to demonstrate that the total LERF of internal events and external events is less than 1E-5/yr.

RAI 2) In addition to extending the ILRT test interval from 10 to 15 years, the license amendment request would extend the drywell-to-suppression chamber bypass test (DWBT) interval to 15 years. The U.S. Nuclear Regulatory Commission has issued similar amendments to the operating licenses for Clinton, Susquehanna, and several other boiling-water reactor (BWR) plants. These interval extensions were based in part on a determination that the combined effect of both test interval extensions on risk was small. To provide insights into cumulative risk impacts, provide an assessment of the combined effect of the ILRT and DWBT interval extensions on risk (i.e., population dose, LERF, and conditional containment failure probability) similar to that provided for these other BWRs.

RAI 2 Response

The following steps are used to perform the analysis for the DWBT interval extension:

- Discuss design basis
- Provide historical test results
- Provide qualitative technical justification of change
- Perform deterministic calculations
- Perform risk assessment of change

LGS Mark II Pressure Suppression Containment Design

LGS incorporates a Mark II containment with the drywell located over the suppression chamber and separated by a diaphragm slab. The suppression chamber contains a pool of water having a depth that varies between 22' and 24'-3" during normal operation. Eighty-seven downcomers and 14 main steam safety/relief valve (SRV) discharge lines penetrate the diaphragm slab and terminate at a pre-designed submergence within the pool. During a loss of coolant accident (LOCA) inside containment, the containment design directs steam from the drywell to the suppression pool via the downcomers through the pool of water to limit the maximum containment pressure response to less than the design pressure of 55 psig. The effectiveness of the LGS pressure suppression containment requires that the leak path from the drywell to the suppression chamber airspace be minimized. Steam that enters the suppression pool airspace through the leak paths will bypass the suppression pool and can result in a rapid post-LOCA increase in containment pressure depending on the size of the bypass flow area.

The design value for leakage area is determined by analyzing a spectrum of LOCA break sizes. For each break size there is a limiting leakage area. In determining the limiting leakage area, credit is taken for the capability of operators to initiate drywell and suppression pool sprays after a period of time sufficient for them to realize that there is a significant bypass leakage flow. The effect of suppression pool bypass on containment pressure response is greatest with small breaks. The design value of 0.0500 ft^2 for LGS represents the maximum leakage area that can be tolerated for that break size that is most limiting with respect to suppression pool bypass.

Limerick Tech Spec (TS) requirements conservatively specify a maximum allowable bypass area of 10% of the design value of 0.0500 ft^2. The TS limit provides an additional factor of 10 safety margin above the conservatisms taken in the steam bypass analysis. The drywell-to-suppression chamber bypass test verifies that the actual bypass flow area is less than or equal to the TS limit.

Historical Test Results

A review of the past test history for the drywell-to-suppression chamber bypass leakage test has identified no failures. The following are the test results:

Unit 1 (Acceptance - 0.005 sq. ft.)	Unit 2 (Acceptance - 0.005 sq. ft.)
1984 - 0.00026	1989 - 0.000069
1987 - 0.000051	1993 - 0.000076
1990 - 0.000278	1999 - 0.000012
1998 - 0.000075	

The history of test results indicates that the typical leakage is about two orders of magnitude or more below the acceptance criteria (which is set at an order of magnitude below the design basis limit). This excellent history combined with the conservatism included in the allowable leakage rate helps to support the qualitative justification provided below, and also helps support the low likelihood of a large undetected bypass leakage in the risk assessment.

Qualitative Justification for DWBT Interval Extension

Several potential bypass leakage pathways exist:

- Leakage through the diaphragm floor penetrations (SRV discharge line downcomers),
- Cracks in the diaphragm floor/liner plate,
- Cracks in the downcomers that pass through the suppression pool airspace,

- Valve seat leakage in the four sets of drywell-to-suppression chamber containment vacuum breakers, and
- Seat leakage of isolation valves in piping connecting the drywell and the suppression chamber air space.

A previous assessment [10] demonstrated that the most likely source of potential bypass leakage is the four sets of drywell-to-suppression chamber vacuum breakers. Each set consists of two vacuum breakers in series, flange mounted to a tee off the downcomers in the suppression chamber airspace. The drywell-to-suppression chamber bypass leak test is currently performed on a schedule consistent with the ILRT. However, TS 4.6.2.1.e requires that the vacuum breaker leakage tests on all four sets of vacuum breakers be performed on all non-ILRT outages. Therefore, the most likely largest contributor to the bypass leakage will still be monitored each refueling outage and therefore will continue to be managed and controlled to assure Tech Spec leakage is maintained.

The vacuum breaker leakage test and stringent acceptance criteria, combined with the historical negligible non-vacuum breaker leakage, and thorough periodic visual inspection provide an equivalent level of assurance as the DWBT that the drywell to suppression chamber bypass leakage can be measured and any adverse condition detected prior to a Loss of Coolant Accident (LOCA).

Deterministic Calculations

As part of the risk assessment of the DWBT interval extension, a set of deterministic thermal hydraulic analyses have been performed to identify the impact of increased drywell to suppression chamber leakage on the risk spectrum.

Tables 2-1 through 2-4 summarize the results of the deterministic thermal hydraulic analyses using the Limerick specific plant model. The first three sets of results in Tables 2-1 through 2-3 focus on the response of containment pressurization to LOCA events as a function of the drywell to suppression chamber bypass leakage. Table 2-4 includes results related to scenarios that proceed to vessel failure and the potential impact on containment pressures following vessel failure as a function of the drywell to suppression chamber bypass leakage. The LOCA events will be discussed first followed by a discussion of the vessel breach scenarios.

Tables 2-1 through 2-3 display the following key results from this analysis and the impact of increased drywell to suppression chamber bypass leakage:

- Medium and large LOCA events create the earliest peak in the containment pressurization. However, these challenges are well within the design basis containment pressure capability even when 35 times the allowable drywell to suppression chamber leakage is assumed to exist. Even for these more severe challenges of a stuck open vacuum breaker, the ultimate containment pressure (~140 psig) is not approached. Therefore, the drywell to suppression chamber bypass leakage does not influence the CDF because adequate vapor suppression is present over the range of bypass leakage considered.
- The small LOCA event presents the most significant challenge to the containment when there is drywell to suppression chamber bypass leakage because the RPV remains at pressure and continues to add energy to the drywell.

However, it should be noted that there are simple crew actions that can successfully mitigate the containment pressurization observed in the small LOCA cases:

- Use of drywell sprays
- Emergency depressurization

Both actions are called for by the LGS TRIPs and neither system is adversely impacted by the small LOCA initiating event.

The time frame for the mitigation system actions to prevent containment overpressure is derived directly from the deterministic calculations and is greater than the "early" phase (i.e., 4 hrs). This time means that the TSC is operational and actions according to the EOPs will be taken with a high degree of certainty, comparable to the certainty applied to the initiation of RHR.

In conclusion, for medium and large LOCAs, variations in the drywell to suppression chamber bypass leakage, from zero to many times Tech Spec leakage, do not impact the vapor suppression capability of the LGS containment and therefore do not significantly(1) impact the calculated CDF or radionuclide release frequency for these accident scenarios. For small LOCAs, the results indicate that the containment pressure exceeds the design pressure during

(1) Defined consistent with NRC definition in RG 1.200. [9]

the 24 hour mission time of the PRA (for the 35x allowable leakage case and for the stuck open vacuum breaker case), but does not exceed the ultimate containment pressure in any case. For simplicity, an operator action to initiate containment sprays or perform an emergency depressurization is assumed to be required to prevent containment overpressure failure for a leakage of this magnitude. These conclusions regarding the impact of the potential for increased drywell to suppression chamber leakage are factored into the risk assessment.

Additional deterministic calculations were run to determine the potential impact of increased drywell to suppression chamber bypass leakage on the likelihood that early containment failure occurs. In this case, a bounding SBO scenario with molten core debris allowed to transport to the suppression pool near the time of vessel failure with various bypass leakage rates was explored. The transport of molten debris to the pool was set to maximize the amount of steam generation shortly after vessel failure to determine the maximum deleterious impact of the bypass leakage. The results are summarized in Table 2-4. In this case, it is clear that the peak pressure immediately following vessel failure is not significantly impacted by variations in the bypass leakage up to 35x the allowable leakage, and a peak pressure of just 96 psig is obtained even when the bypass leakage is equivalent to the vacuum breaker flow area. The calculated peak pressures in all cases are well below the ultimate containment failure pressure of 140 psig. This supports the assessment that any reasonable amount of undetectable drywell bypass leakage will not have a measurable impact on the potential for increased occurrence of early containment failure due to vapor suppression failure at the time of vessel failure that could lead to an increase in LERF. This assessment is also consistent with the Limerick Level 2 PRA model success criterion that requires two vacuum breaker lines failed open to represent a vapor suppression failure at the time of vessel failure. As such, the risk assessment assumes that there is no increase in LERF from this potential accident scenario (i.e. LERF due to early containment failure from vapor suppression failure at the time of vessel failure) due to changing the DWBT interval.

Table 2-1

CONTAINMENT PRESSURE RESPONSE FOR SMALL LOCA INITIATORS AS A FUNCTION OF DRYWELL TO WETWELL BYPASS LEAKAGE

Case	Description	Drywell Pressure (psig)		
		Initial Peak	At 5 Hrs	At 24 Hrs
SLOCA-0L	Small LOCA with ECCS available and no DW to WW bypass. • RHR is operating in pool cooling. • No sprays actuated.	NA	33	39
SLOCA-10L	Small LOCA with ECCS available and 10x Tech Spec leakage from DW to WW. • RHR is operating in pool cooling. • No sprays actuated.	NA	42	52
SLOCA-35L	Small LOCA with ECCS available and 35x Tech Spec leakage from DW to WW. • RHR is operating in pool cooling. • No sprays actuated.	NA	49	72
SLOCA-600L	Small LOCA with ECCS available and stuck open vacuum breaker (~600x Tech Spec leakage from DW to WW). • RHR is operating in pool cooling. • No sprays actuated.	NA	48	70

Table 2-2

CONTAINMENT PRESSURE RESPONSE FOR MEDIUM LOCA INITIATORS AS A FUNCTION OF DRYWELL TO WETWELL BYPASS LEAKAGE

Case	Description	Drywell Pressure (psig)		
		Initial Peak	At 5 Hrs	At 24 Hrs
MLOCA-0L	Medium LOCA[(1)] with ECCS available and no DW to WW bypass. • RHR is operating in pool cooling. • No sprays actuated.	29.6	13.4	12.5
MLOCA-10L	Medium LOCA[(1)] with ECCS available and 10x Tech Spec leakage from DW to WW. • RHR is operating in pool cooling. • No sprays actuated.	31.1	13.9	12.4
MLOCA-35L	Medium LOCA[(1)] with ECCS available and 35x Tech Spec leakage from DW to WW. • RHR is operating in pool cooling. • No sprays actuated.	35.3	13.8	12.3
MLOCA-600L	Medium LOCA[(1)] with ECCS available and stuck open vacuum breaker (~600x Tech Spec leakage from DW to WW). • RHR is operating in pool cooling. • No sprays actuated.	75.3	14.0	11.2

(1) 5" water break

Table 2-3

CONTAINMENT PRESSURE RESPONSE FOR LARGE LOCA INITIATORS AS A FUNCTION OF DRYWELL TO WETWELL BYPASS LEAKAGE

Case	Description	Drywell Pressure (psig)		
		Initial Peak	At 5 Hrs	At 24 Hrs
LLOCA-0L	Large LOCA[1] with ECCS available and no DW to WW bypass. • RHR is operating in pool cooling. • No sprays actuated.	28.8	10.0	6.7
LLOCA-10L	Large LOCA[1] with ECCS available and 10x Tech Spec leakage from DW to WW. • RHR is operating in pool cooling. • No sprays actuated.	29.4	10.2	6.8
LLOCA-35L	Large LOCA[1] with ECCS available and 35x Tech Spec leakage from DW to WW. • RHR is operating in pool cooling. • No sprays actuated.	32.2	8.7	5.9
LLOCA-600L	Large LOCA[1] with ECCS available and stuck open vacuum breaker (~600x Tech Spec leakage from DW to WW). • RHR is operating in pool cooling. • No sprays actuated.	86.8	14.8	7.9

(1) 13.5" water break

Table 2-4

CONTAINMENT PRESSURE RESPONSE STATION BLACKOUT SCENARIOS AS A FUNCTION OF DRYWELL TO WETWELL BYPASS LEAKAGE

Case	Description	Drywell Pressure (psig) Peak After Vessel Failure
SBO-0L	Station Blackout with no ECCS available and no DW to WW bypass. • No RHR in pool cooling. • No sprays actuated.	60
SBO-10L	Station Blackout with no ECCS available and 10x Tech Spec leakage from DW to WW. • No RHR in pool cooling. • No sprays actuated.	65
SBO-35L	Station Blackout with no ECCS available and 35x Tech Spec leakage from DW to WW. • No RHR in pool cooling. • No sprays actuated.	74
SBO-600L	Station Blackout with no ECCS available and stuck open vacuum breaker (~600x Tech Spec leakage from DW to WW). • No RHR in pool cooling. • No sprays actuated.	96

Risk Assessment

The Drywell to Suppression Chamber leakage can lead to the following perturbations on risk metrics:

- The increase in leakage could result in an increase in the failure probability of the vapor suppression function and consequential failure of containment. This could lead to pool bypass and core damage.
- The bypass leakage would result in an increase in the radionuclides in the suppression chamber airspace following an RPV breach if drywell sprays were unavailable. This could result in increased radionuclide release for suppression chamber breach cases or suppression chamber (wetwell) vent cases with core damage and no drywell failure or other pool bypass mechanisms.

The following steps are used for the risk assessment:

1. Determine sequences that are impacted by changes in bypass area.
2. Calculate probability of large bypass area.
3. Calculate risk metrics for original bypass test interval.
4. Calculate risk metrics for 10 year bypass test interval.
5. Calculate risk metrics for 15 year bypass test interval.
6. Summarize the changes in the calculated risk metrics.

Step 1: Determine Sequences Impacted by Changes in Bypass Area

As shown in the deterministic calculations, the only accident sequences that are impacted by the DWBT interval extension are those severe accidents induced by a loss of containment integrity due to overpressure failure. Additionally, it was shown that the only potential contributors to this situation are those small LOCAs that have sufficiently high bypass leakage to allow continual containment pressurization coupled with no mitigating actions. These small LOCA scenarios represent a change in CDF, but not LERF because the late failures would result in radionuclide releases at >6 hours after a general emergency is declared.

Separate deterministic calculations also showed that the peak containment pressure following vessel failure was not significantly impacted by the assumed bypass leakage. Therefore, there is no measurable increase in LERF from this potential impact.

On the other hand, however, it is acknowledged that some accident scenarios that are currently classified as early wetwell region failures have the potential to be re-categorized as LERF due to the presence of a large bypass area that would render the fission product scrubbing capabilities of the suppression pool ineffective in reducing the source term below LERF threshold values.

(The current LGS PRA does not include the DW to WW bypass leakage term as a potential failure mode. Therefore, the current baseline risk metric calculations need to be adjusted to incorporate the probability that the bypass leakage is unacceptably high.)

Step 2: Calculate Probability of Large Bypass Area

Industry and LGS experience with the results of the DWBT has been quite good. However, for simplicity and for consistency with the original ILRT analysis for Limerick, it will be assumed that the base case potential for a large drywell to suppression chamber bypass leak (35La) is the same as was utilized for the ILRT analysis (i.e. 0.0027).

Additionally, consistent with the NEI Guidance [11], the change in the probability of a large undetected bypass increases by a factor of 3.33 for a ten-year interval and an extension to a 15 year interval can be estimated to lead to a factor increase of 5.0 in the non-detection probability of a leak.

Step 3: Calculate the Risk for the Original Bypass Leak Rate Test Interval

The LGS base case did not include DW to WW bypass failure. Therefore the frequency of the Base Case model is adjusted to incorporate the severe accident frequency.

As described in Step 2, the probability of a "large" bypass given the original DWBT interval and excellent historical test experience is assumed to be 0.0027. Thus, the CDF to be added to the base model is:

ΔCDF = SLOCA $*$ Large Bypass Leak Probability $*$ DW Spray Failure Probability $*$ Emergency Depressurization Failure Probability

Where the SLOCA, DW spray failure probability and emergency depressurization human error failure probability values are taken from the current LGS PRA model. Thus:

$$\Delta CDF = SLOCA * 0.0027 * 4E\text{-}3 * 3E\text{-}4$$

$$\Delta CDF = 8.6E\text{-}3/yr * 0.0027 * 4E\text{-}3 * 3E\text{-}4 = 2.8E\text{-}11/yr$$

Adjustments are made to EPRI Category 7 to address this "new" contributor to core damage. However, as can be easily seen, this "new" contributor is negligible compared with the previously assessed base case, and will not have any measurable impact on the results.

The potential change in LERF is limited to those accident scenarios that were previously classified as early wetwell region failures in Category 7 (i.e. APB numbers 1 and 2 from Table 5.1-2 of the PRA portion of the Limerick submittal). That portion of APBs #1 and #2 impacted by the bypass leakage will be re-assigned to APB #3 (EPRI Class 2) and #4 (EPRI Class 8), respectively for the corresponding DW failure modes, and the measured change will be assumed to represent a change in LERF.

$$\Delta APB\#3 = APB\#1_{Original} * \text{Large Bypass Leak Probability}$$

$$= 7.40E\text{-}09/yr * 0.0027 = 2.00E\text{-}11/yr$$

$$\Delta APB\#4 = APB\#2_{Original} * \text{Large Bypass Leak Probability}$$

$$= 2.39E\text{-}07/yr * 0.0027 = 6.45E\text{-}10/yr$$

$$\Delta APB\#1 = -\Delta APB\#3$$

$$\Delta APB\#2 = -\Delta APB\#4$$

For the purposes of this assessment, the changes to EPRI Classes 3a and 3b from the ILRT interval extension will be ignored so as to isolate the potential impact of the changes on the DWBT interval extension. With the population dose information derived for LGS as shown in Table 5.2-2 of the PRA portion of the Limerick submittal, with the initial EPRI Class 2, 7, and 8 frequency information obtained from the detailed information that was used to support the development of that table, and with EPRI Class 1 assigned the remaining CDF from the total, the revised base case results showing the adjustments to Class 2, 7 and 8 as described above are shown in Table 2-5.

Table 2-5

QUANTITATIVE RESULTS AS A FUNCTION OF
ORIGINAL DWBT INTERVAL

EPRI Class	Dose (Person-Rem Within 50 miles)	Original DWBT Interval	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	1.32E+04	1.324E-6	1.747E-2
2	7.93E+06	2.182E-8 + 2.00E-11 = 2.184E-8	1.732E-1
3a	1.32E+05	-	-
3b	1.32E+06	-	-
4	N/A	N/A	N/A
5	N/A	N/A	N/A
6	N/A	N/A	N/A
7	3.97E+06	2.306E-6 + 2.80E-11 - 2.00E-11 - 6.45E-10 = 2.305E-6	9.153
8	6.01E+06	4.648E-8 + 6.45E-10 4.712E-8	2.832E-1
TOTALS CDF: LERF (Class 2 + Class 8): CCFP[1]:		 3.698E-6 6.896E-8 64.20%	9.627

[1] Determined from (Class 2 + Class 7 + Class 8) / (Total CDF)

Step 4: Calculate the Risk for 10 Year Bypass Leak Rate Test Interval

The risk metrics for the 10 year DWBT interval are the same as the base case from Step 3, except the impact of the bypass leakage is increased by a factor of 3.33 consistent with the ILRT assessment. The revised results are shown in Table 2-6.

Table 2-6

QUANTITATIVE RESULTS AS A FUNCTION OF
10 YEAR DWBT INTERVAL

EPRI Class	Dose (Person-Rem Within 50 miles)	10 Year DWBT Interval: Accident Frequency (per year)	10 Year DWBT Interval: Population Dose Rate (Person-Rem/Year Within 50 miles)
1	1.32E+04	1.324E-6	1.747E-2
2	7.93E+06	2.182E-8 + 3.33 * 2.00E-11 = 2.189E-8	1.736E-1
3a	1.32E+05	-	-
3b	1.32E+06	-	-
4	N/A	N/A	N/A
5	N/A	N/A	N/A
6	N/A	N/A	N/A
7	3.97E+06	2.306E-6 + 3.33 * 2.80E-11 - 3.33 * 2.00E-11 - 3.33 * 6.45E-10 = 2.304E-6	9.147
8	6.01E+06	4.648E-8 + 3.33 * 6.45E-10 4.863E-8	2.922E-1
TOTALS			9.630
	CDF:	3.698E-6	
	LERF (Class 2 + Class 8):	7.051E-8	
	CCFP:	64.21%	

Step 5: Calculate the Risk for 15 Year Bypass Leak Rate Test Interval

The risk metrics for the 15 year DWBT interval are the same as the base case from Step 3, except the impact of the bypass leakage is increased by a factor of 5.0 consistent with the ILRT assessment. The revised results are shown in Table 2-7.

Table 2-7

QUANTITATIVE RESULTS AS A FUNCTION OF
15 YEAR DWBT INTERVAL

EPRI Class	Dose (Person-Rem Within 50 miles)	15 Year DWBT Interval	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	1.32E+04	1.324E-6	1.747E-2
2	7.93E+06	2.182E-8 + 5.0 * 2.00E-11 = 2.192E-8	1.738E-1
3a	1.32E+05	-	-
3b	1.32E+06	-	-
4	N/A	N/A	N/A
5	N/A	N/A	N/A
6	N/A	N/A	N/A
7	3.97E+06	2.306E-6 + 5.0 * 2.80E-11 - 5.0 * 2.00E-11 - 5.0 * 6.45E-10 = 2.303E-6	9.143
8	6.01E+06	4.648E-8 + 5.0 * 6.45E-10 4.970E-8	2.987E-1
TOTALS CDF: LERF (Class 2 + Class 8): CCFP:		 3.698E-6 7.162E-8 64.21%	9.633

Step 6: Summarize the Changes in the Calculated Risk Metrics

Consistent with the ILRT assessment, the relevant figures of merit are change in LERF, population dose, and conditional containment failure probability (CCFP). Additionally, the DWBT extension will also lead to a potential slight change in CDF as described above. The results for these figures of merit from the DWBT interval extension are shown below in Table 2-8.

Table 2-8

SUMMARY OF QUANTITATIVE RESULTS FOR DWBT INTERVAL EXTENSION REQUEST

Figure of Merit	Original DWBT Interval	10 Year DWBT Interval	15 Year DWBT Interval
CDF (/yr)	3.698E-06	3.698E-06	3.698E-06
LERF (/yr)	6.896E-08	7.051E-08	7.162E-08
Dose (person-rem/yr)	9.627	9.630	9.633
CCFP (%)	64.20%	64.21%	64.21%
Changes from original interval			
Increase in CDF (/yr):		6.52E-11	1.12E-10
Increase in LERF (/yr):		1.55E-09	2.66E-09
Increase in Dose (person-rem/yr):		0.003	0.006
Increase in CCFP (%):		0.01%	0.01%

Based on the results of the deterministic studies and their probabilistic risk assessment implications, the following can be defined:

- Increasing the DWBT interval is assumed to increase the probability of increased bypass leakage.
- There is a very small change in core damage frequency (CDF) associated with the possibility that a small LOCA occurs with the increased DW to WW bypass leakage and the containment pressurization is not mitigated. This is conservatively assumed to lead to containment failure and consequential loss of RPV makeup and results in core damage.
- There is also a very small change in the large early release frequency (LERF) associated with the possibility that previous early WW region failures that were not considered LERF due to the fission product scrubbing effects of the suppression pool would be LERF if sufficient bypass leakage area exists.
- The change in population dose associated with the other changes above is noted in Table 2-8. The overall change in population dose is also negligible (<<1%).
- There is also a very small change in the conditional containment failure probability (CCFP) since the increase in CDF is negligible and the increase in LERF is only from cases that were already containment failure cases (albeit shifted to a LERF release).

The risk metric changes to be compared are then:

Δ CDF	= 1.12E-10/yr
Δ LERF	= 2.66E-09/yr
Δ Person-rem dose rate	= 0.006 person-rem/year
Δ CCFP	= 0.01%

The changes in CDF and LERF meet the Regulatory Guide 1.174 [8] acceptance guidelines for very small risk change. While no acceptance guideline is available for the change in population dose rate or CCFP, the minimal change in dose rate and in CCFP indicate that there is very negligible change in ex-plant consequences associated with the DWBT interval extension.

This very small change in the risk metrics would also not significantly change the revised assessment which factors in the potential impact from external events in the RAI 1 response above.

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[9] U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Regulatory Guide 1.200, Revision 1, January 2007.

[10] Letter from G.A. Hunger, Jr. (Philadelphia Electric Company) to U.S. Nuclear Regulatory Commission, "Limerick Generating Station, Units 1 and 2 Technical Specifications Change Request, Dockets No. 50-352 and 50-353", November 30, 1993.

[11] Letter from A. Pietrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," November 13, 2001.