NEXTera ENERGY DUANE ARNOLD

August 18, 2015

NG-15-0234
10 CFR 50.90

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

Duane Arnold Energy Center
Docket No. 50-331
Renewed Facility Operating License No. DPR-49

License Amendment Request (TSCR-143) to Extend Containment Leakage Test Frequency

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), NextEra Energy Duane Arnold, LLC (hereafter, NextEra Energy Duane Arnold) is submitting a request for an amendment to the Technical Specifications (TS) for Duane Arnold Energy Center (DAEC).

The proposed amendment revises Technical Specifications (TS) Section 5.5.12, "Primary Containment Leakage Rate Testing Program," by requiring compliance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," instead of Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," including exemption 1.

Attachment 1 provides an evaluation of the proposed changes. Attachment 2 provides marked-up pages of the existing TS to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides a plant specific risk analysis. Attachment 5 provides documentation of probabilistic risk assessment technical adequacy. There are no new Regulatory Commitments or revisions to existing Regulatory Commitments.

Approval is requested by September 1, 2016, to support Refueling Outage (RFO) 25, with the amendment being implemented within 60 days of its receipt.

In accordance with 10 CFR 50.91(b)(1), "Notice for Public Comment; State Consultation," a copy of this application, including attachments, is being provided to the designated State of Iowa official.

The DAEC Onsite Review Group has reviewed the proposed license amendment request.

If you have any questions or require additional information, please contact J. Michael Davis at 319-851-7032.

A017

Rec'd@ DPC on 9/2/15

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

Executed on August 18, 2015.

T. A. Vehec
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Attachments: As stated

cc: Regional Administrator, USNRC, Region III,
Project Manager, USNRC, Duane Arnold Energy Center
Resident Inspector, USNRC, Duane Arnold Energy Center
A. Leek (State of Iowa)

ATTACHMENT 1 to NG-15-0234

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-143)
EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY

EVALUATION OF PROPOSED CHANGES

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, NextEra Energy Duane Arnold, LLC (NextEra Energy Duane Arnold) hereby requests an amendment to Duane Arnold Energy Center (DAEC) Technical Specifications (TS). This proposed change will allow extension of the Type A test interval up to one test in 15 years and extension of the Type C test interval up to 75 months, based on acceptable performance history as defined in NEI 94-01, Revision 3-A.

The requested amendment would revise TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to follow guidance developed by Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," (Reference 7.1) that was found by the NRC to describe an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, as modified by the conditions and limitations in the Safety Evaluation (Reference 7.2).

The purpose of NEI 94-01, Revision 3-A guidance is to assist licensees in the implementation of Option B to 10 CFR 50, Appendix J, "Leakage Rate Testing of Containment of Light Water Cooled Nuclear Power Plants," (hereafter referred to as Appendix J, Option B). Revision 2-A of NEI 94-01 (Reference 7.3) added guidance for extending containment integrated leak rate test (ILRT or Type A test) surveillance intervals beyond ten years; and Revision 3-A of NEI 94-01 adds guidance for extending containment isolation valve (Type C test) local leakage-rate test (LLRT) surveillance intervals beyond sixty months.

The technical basis for the proposed license amendment utilizes risk-informed analysis augmented with non-risk related considerations. A risk impact evaluation performed by Westinghouse Electric Company (WEC) concluded that the increases in large early release frequency (LERF) are within the limits set forth by the applicable guidance contained in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.174 (Reference 7.4), NUREG-1493, "Performance-Based Containment Leak-Test Program," and EPRI Technical Report TR-1009325 (Reference 7.5).

In accordance with the guidance of NEI 94-01 Revision 3-A, DAEC proposes to extend the maximum surveillance interval for the ILRT to no longer than 15 years from the last ILRT based on satisfactory performance history. The current interval is no longer than 10 years and would require that the next ILRT for DAEC be performed during the Fall 2016 refueling outage. The proposed change would allow the DAEC ILRT to be performed in 2022. This will reduce the number of ILRTs performed over the licensed period of operation resulting in significant savings in radiation exposure to personnel, cost, and critical path time during refueling outages.

2.0 DETAILED DESCRIPTION

2.1 PROPOSED CHANGE

The proposed license amendment would revise TS Section 5.5.12.b by changing the wording to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A, instead of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.163.

Current TS Section 5.5.12.b states in part that:

> This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as

modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

1. The first Type A test after the September 1993 Type A test shall be performed no later than September 2008.

The proposed amendment would change this wording to indicate that the program shall be in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A. The proposed amendment would also delete the reference to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 and the listed Type A test exception.

A marked-up copy of the proposed changes to the TS is provided in Attachment 2. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides a plant specific risk analysis.

2.2 DESCRIPTION OF DAEC PRIMARY CONTAINMENT

The DAEC primary containment structure is a portion of the General Electric (GE) Mark I Primary Containment Pressure Suppression System. The complete pressure suppression system consists of the drywell which houses the reactor vessel and reactor coolant recirculation loops, the pressure suppression chamber, the connecting vent system between the drywell and pressure suppression chamber, isolation valves, vacuum relief system, and containment cooling systems.

The drywell is a steel pressure vessel (0.75 to 3.0 inches thick), with a spherical lower portion and cylinder upper portion. It is enclosed in reinforced concrete, 4 to 7 feet thick, for shielding, and to provide additional resistance to deformation and buckling over areas where the concrete backs up the steel shell. Above the foundation transition zone, and below the flange, the drywell is separated from the reinforced concrete by a gap of approximately 2 inches to allow for thermal expansion. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

The drywell vessel is provided with a removable head to facilitate refueling, one combination double door personnel access lock/equipment lock, one equipment hatch, one personnel access hatch, and one control rod drive removal hatch. The head and hatches are all bolted in place and have double seals and test taps for leak tests.

Special bellows seals are provided between the reactor vessel, the drywell vessel, and the reactor well to form a watertight seal and enable flooding of the upper portion of the drywell during refueling operations. To protect the outer circumference of the bellows, a backing plate is provided which has a test tap for leakage monitoring. During normal operation, six watertight hinged covers are opened permitting circulation of ventilation air in the region above the reactor well seal bulkhead plate via removable air supply and return ducts.

The pressure suppression chamber is a steel pressure vessel (0.50-0.534 inches thick) in the shape of a torus located below and encircling the drywell. The pressure suppression chamber contains the suppression pool and the gas space above the pool. The suppression chamber will transmit seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside the chamber for inspection. Access to the chamber is provided at two locations. There are two 4 foot diameter manhole entrances with double gasketed, leak testable, and bolted covers connected to the chamber by 4 foot diameter steel pipe inserts. These access ports will be closed when

Primary Containment is required and will be opened only when the primary coolant temperature is below 212 °F and the pressure suppression capability is no longer required.

The pressure suppression pool serves as a heat sink for postulated transient or accident conditions. Energy is transferred to the pool by either the discharge piping from the reactor pressure safety/relief valves or the drywell vent piping, which discharge below the water level. The pool condenses the steam portion of the flow and collects any water carryover while non-condensable gases (including any gaseous fission products) are released to the suppression chamber gas space. The pool also acts as a heat sink for High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System steam exhaust. Energy is removed from the suppression pool when the Residual Heat Removal (RHR) System is operating in the suppression pool cooling mode.

The suppression pool is also the primary source of water for the Core Spray System and the Low Pressure Coolant Injection (LPCI) mode of the RHR System and the secondary source of water for the RCIC and HPCI Systems. The quantity of water stored in the suppression pool is sufficient to condense the steam from a design basis accident and to provide adequate water for the emergency core cooling systems (ECCS). The suppression chamber is subject to the pressure associated with the storage of a minimum of 58,900 - 61,500 cubic feet of water distributed uniformly within the vessel during normal operation. Under accident conditions, the suppression chamber is designed for 61,500 cubic feet of water and a maximum containment pressure of 62 psig.

Eight 4'9" diameter vent pipes connect the drywell and the pressure suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces or projectiles that might accompany a pipe break in the drywell. The vent pipes are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber. These bellows have test connections that allow for leak testing and for determining that the passages between the two-ply bellows are not obstructed.

The drywell vents are connected to a 3'6" diameter vent header in the form of a torus, which is contained within the air space of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, 24 inches in diameter and terminating 3 feet below the water surface of the pool and approximately 7 feet above the bottom of the Torus.

Containment penetrations are designed for the same integrity as the primary containment structure itself. They will not limit the capabilities of the Primary Containment System to act as a radiological barrier before, during, or subsequent to any design basis accident.

One combination personnel access lock/equipment lock is provided for access to the drywell. The personnel lock has two gasketed doors in series, with each door designed and constructed to withstand the drywell design differential pressure. The doors are mechanically interlocked to ensure that at least one door is locked at times when primary containment is required. The locking mechanisms are designed so that a tight seal will be maintained when the doors are subjected to either internal or external pressure. The seals on this access opening are capable of being tested for leakage. The personnel access lock is bolted to an equipment insert barrel approximately 12 feet in diameter, which, in turn, provides double testable seals and is welded to the drywell shell. The personnel access lock can be completely removed by an overhead monorail to increase the size of the opening should a larger access be required.

A personnel access hatch is provided in the drywell head. There is a separate equipment access hatch that provides access for larger equipment to pass though the containment. These hatches are bolted in place and provide double testable seals.

Personnel and equipment hatches are sized and located with full consideration of service required, accessibility for maintenance, and periodic testing programs. A 2-inch minimum gap is maintained around the barrel of the personnel and equipment hatches as they pass through the concrete shield wall.

A control rod drive removal hatch with double, testable seals is provided. This hatch is bolted in place and permits removal of the drive mechanisms when required.

3.0 TECHNICAL EVALUATION

3.1 LEAK TEST HISTORY

3.1.1 Type A Testing

The historical results of the Type A tests for DAEC are included in the table provided below. The reported leak rate is at the 95 percent upper confidence level and includes any Type B and Type C penalties.

The last DAEC Type A test was completed on March 13, 2007. Previous Type A testing confirmed that the DAEC containment structure leakage is acceptable, with considerable margin, with respect to the TS acceptance criterion of 2.0 percent of primary containment air weight per day at the design basis loss of coolant accident pressure (P_a). Since the last two DAEC Type A test as-found results, as shown in the table provided below, were less than 1.0 L_a, a test frequency of at least once per 10 years was justified in accordance with NEI 94-01, Revision 0.

Repair or replacement activities (including any unplanned activities) performed on the pressure retaining boundary of the primary containment prior to the next scheduled Type A test would be subject to the leakage test requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Paragraph IWE-5221, "Leakage Test." There have been no pressure or temperature excursions in the containment that could have adversely affected containment integrity. There are no anticipated repairs or modifications of the containment that could affect leak-tightness that would not be measured by local leak rate testing as required in Section 9.2.4 of NEI 94-01, Revision 0.

Following the approval of this license amendment, the next DAEC Type A test must be performed on or before March 13, 2022.

DAEC Type A Test Historical Results Since 1985

Test Completion Date	As Found Leak Rate	As Found Acceptance Criteria	As Left Leak Rate	As Left Acceptance Criteria
1985 (RFO 7)	Not quantified	≤ 2.0 %wt/day	0.478 %wt/day	≤ 1.5 %wt/day
1987 (RFO 8)	Not quantified	≤ 2.0 %wt/day	0.503 %wt/day	≤ 1.5 %wt/day
1988 (RFO 9)	1.353 %wt/day	≤ 2.0 %wt/day	0.229 %wt/day	≤ 1.5 %wt/day
1990 (RFO 10)	1.633 %wt/day	≤ 2.0 %wt/day	1.146 %wt/day	≤ 1.5 %wt/day
9/20/1993 (RFO 12)	0.511 %wt/day	≤ 2.0 %wt/day	0.254 %wt/day	≤ 1.5 %wt/day
3/13/2007 (RFO 20)	0.355 %wt/day	≤ 2.0 %wt/day	0.342 %wt/day	≤ 1.5 %wt/day

%wt/day = Percent primary containment air weight per day

The following is a description of the results from the latest two ILRTs at DAEC.

The March 2007 periodic Type A test was performed using BN-TOP-1 calculated at the 95% upper confidence limit (UCL). The performance leak rate corresponding to the definition in NEI 94-01 was equal to the as-left ILRT results of 0.342 %wt/day since no leakage paths were isolated during the ILRT.

The September 1993 periodic Type A test was performed using BN-TOP-1 calculated at the 95% UCL, which resulted in a value of 0.15356 %wt/day. The performance leak rate corresponding to the definition in NEI 94-01 was 0.254 %wt/day with corrections.

As required by NEI 94-01, Revision 3-A Section 9.1.2, further extensions in test intervals are based upon two consecutive, periodic, successful Type A tests and requirements stated in Section 9.2.3 of this guideline. The results in the table show that there has been substantial margin to the maximum allowable leakage rate of 2.0 %wt/day.

3.1.2 Type B and C Testing

The Type B and Type C containment leakage rate testing program for DAEC requires pneumatic tests intended to detect or measure leakage across pressure-retaining or leakage limiting boundaries and containment isolation valves. As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of potential containment leakages.

As discussed in NUREG-1493 and NEI 94-01, Revision 3-A, Type B and Type C tests can identify the vast majority of all containment leakage paths. This amendment request adopts the guidance in NEI 94-01, Revision 3-A in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that containment leakage rates are maintained well within limits.

A review of the Type B and Type C test results from the spring of 2003 through the fall of 2014 has shown a large amount of margin between the actual as-found and as-left outage summations and the TS leakage rate acceptance criteria (that is, less than 0.6 L_a).

- The as-found minimum pathway leak rate for DAEC shows an average of 14.0 percent of 0.6 L_a.
- The as-left maximum pathway leak rate for DAEC shows an average of 23.8 percent of 0.6 L_a with a high of 35.5 percent or 0.22 L_a.

DAEC Type B and Type C Leak Rate Summation History Since 2003

Refueling Outage	As-Found Min Path	Percentage of 0.6 L_a	As-Left Max Path	Percentage of 0.6 L_a
RFO 18 Spring 2003	40,136 sccm	18.3%	55,184 sccm	25.1%
RFO 19 Spring 2005	37,522 sccm	17.1%	40,083 sccm	18.3%
RFO 20 Winter 2007	22,543 sccm	10.3%	77,918 sccm	35.5%
RFO 21 Winter 2009	18,212 sccm	8.3%	44,995 sccm	20.5%
RFO 22 Fall 2010	20,960 sccm	9.5%	53,525 sccm	24.4%
RFO 23 Fall 2012	53,212 sccm	24.2%	47,003 sccm	21.4%
RFO 24 Fall 2014	23,276 sccm	10.6%	47,346 sccm	21.6%

sccm = standard cubic centimeters per minute

There was one local leak rate test failure during RFO 24. The as-found test result for CV2211 was 2865 sccm, which exceeds the administrative limit of 1500 sccm. After running the HPCI system, an as-left test was successfully performed on CV2211. Therefore, the HPCI system flushed out debris on the valve disk and seat that caused the as-found test failure. The as-left LLRT was performed with a measured leakage of 540 sccm.

There were three test failures during RFO 23: CV5704B failed an as-found LLRT, CV4305 failed an as-found LLRT, and CV4300 failed an as-found LLRT.

An LLRT was attempted on CV5704B during RFO 23, but the required test pressure could not be obtained. A packing leak on V57-0076 was identified. After repairing the packing leak, testing was then performed again. Test pressure could not be maintained and flow was noticed at the vent point outside of the test boundary. A work order was then initiated to repair the valve. Repairs were performed and the valve returned for as-left testing. During the setup of the valve prior to testing, it was identified that there were air leaks at the diaphragm of PCV5704B, CV-5704B CONTROL AIR PRESSURE REGULATOR, and on the supply line. An as-left LLRT was performed which indicated leakage above the acceptance criteria. That afternoon it was discovered that there was a loose fitting on the test equipment. The fitting was tightened and the LLRT was successfully completed. PCV5704B was replaced after the test had been completed.

The inspection of CV5704B identified the following as-found degraded conditions: the valve body and piping contained dirt, indications of a packing leak on the stem, dirt on the stem, dirt on the seat, dirt on the disc, and excessive wear on the stem back seat. The as-found blue check on the valve seat to the disk seat was satisfactory as was the total indicated runout of the valve stem. The valve was cleaned and reassembled. The only valve components replaced were the bonnet gasket and the packing. The as-left LLRT was successfully performed at the end of RFO 23 with a measured leakage of 1,917 sccm.

Also during RFO 23, CV4305, TORUS VACUUM BREAKER V-43-168 ISOLATION, failed an LLRT. The administrative limit is 11,000 sccm, and the as-found result was unmeasurable. The test volume

could not be pressurized above 21.6 psig and air flow was felt coming from the vent path at V43-0037. A work order was initiated to repair the valve. The valve was disassembled and the T-seal and O-rings replaced. Information provided by the mechanics involved with this work indicates that nothing appeared to be abnormal with the T-seal or the O-rings. The valve was reassembled. The position stop was adjusted during valve reassembly. An LLRT was again performed and the test volume could not be pressurized above 14 psig. The vacuum breaker, V43-0168, was opened and it was observed that the disc was slightly off the seat. The actuator position stops were repositioned to allow the disc to fully close. Additionally, the set screw that actuates the spool valve to pressurize the T-seal was found loose. It is possible that the set screw would actuate the spool valve too early which would prematurely inflate the T-seal and could restrict the valve disc from seating fully. The set screw position was adjusted. The as-left LLRT was then performed with a measured leakage of 382 sccm.

The hex head screw is typically secured to the lever by a lock nut on CV4305. Following this failure, the set screw was able to be adjusted by hand indicating that the lock nut was not engaged. After reviewing the maintenance procedure, it was determined that there was no step to ensure that the lock nut is engaged. In order to prevent recurrence of this failure on CV4305, the maintenance procedure has been revised to include this step.

The final contributing failure during RFO 23 was due to a potential T-seal problem on CV4300, TORUS VENT LINE INBOARD ISOLATION. LLRT Test pressure (46-48 psig) between CV4300, CV4301 - TORUS VENT LINE OUTBOARD ISOLATION, and CV4357 - TORUS HARD PIPE VENT LINE ISOLATION - was initially achieved and then pressure dropped sharply to about 38 psig. Pressure to the T-seal for CV4300 was noted to be about 80 psig before pressurizing the volume between CV4300, CV4301, and CV4357 and 20 psig after the test volume depressurized to about 38 psig. T-seal pressure read about 20 psig at that point for CV4300. The administrative limit for this combined LLRT test is 22,000 sccm and the as-found result could not be determined.

Initially, an LLRT of CV4300 was attempted but a test pressure of greater than 38 psig could not be obtained. Troubleshooting was performed with the LLRT boundary pressurized to try to identify the cause of the leakage. This troubleshooting identified that the hex head screw was not fixed in place and would allow the plunger to reposition and deflate the seal. A work order was initiated to repair the valve by applying loctite to threads of the hex head screw. The as-left LLRT was then successfully performed with a measured leakage of 70 sccm.

To prevent recurrence of this failure on CV4300, a reduced height locknut was installed. This locknut is sufficient to secure the hex head screw in place and not allow the plunger of the spool valve to move and release the seal pressure.

3.2 CONTAINMENT INSPECTIONS

General visual examinations of the accessible surfaces of the primary containment are performed in accordance with the Primary Containment Inspection Program. These examinations are performed to assess the general condition of the primary containment surfaces and to satisfy the visual examination requirements of ASME Code Section XI, Subsection IWE. These examinations are performed in sufficient detail to detect signs of deterioration.

Detailed visual examinations are performed to determine the magnitude and extent of deterioration of suspect surfaces initially detected by general visual examinations. The conditions reported during the examinations are evaluated to determine acceptability. The conditions are acceptable if it is

determined that there is no evidence of damage or degradation sufficient to warrant further evaluation or performance of repair and replacement activities.

The primary containment is visually examined under two separate programs. The first is the Primary Containment Inspection Program discussed in Section 3.2.1. This program includes provisions to satisfy the visual examination requirements of ASME Code Section XI, Subsection IWE and 10 CFR 50, Appendix J, Option B. A visual examination is made of the accessible interior surfaces of containment in order to identify evidence of deterioration that may affect the containment structural integrity or leak tightness. If signs of corrosion are evident that exceed the acceptance standard (IWE-3500), they must be either corrected by a repair or replacement activity or deemed acceptable for continued service by an engineering evaluation. Both Regulatory Guide 1.163, September 1995, and the ASME Code require a general visual examination of the accessible liner surfaces three times in a ten year period.

The second program is the Containment Coatings Inspection and Assessment Program discussed in Section 3.2.8. This program mandates a visual inspection and assessment of the protective coatings on the containment structure and equipment in the readily accessible areas of the primary containment.

This program is implemented to ensure that the integrity of the coatings is maintained and was established in response to NRC Generic Letter 1998-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coatings Deficiencies and Foreign Material in Containment." The inspection frequency of the above programs ensures that when an area of concern is identified, it only affects a small localized area. Corrective action is taken following any signs of coating blistering, peeling, or corrosion.

3.2.1 Containment Inservice Inspection Program (IWE)

The ASME Code Section III, Class B, 1968 Edition with the Summer 1968 Addenda and Code Cases 1177, 1330, and 1413 were used for the design, fabrication, erection, and testing of the DAEC Primary Containment. The Primary Containment Inspection Program applies to the containment vessel (ASME Code Section XI, Subsection IWE).

ASME Code Section XI, Subsection IWE specifies that examinations will be performed on the pressure retaining boundary of the containment vessel, which includes the accessible surfaces of the liner plate, integral attachments and structures that are part of the reinforcing structure, surfaces of pressure retaining welds, pressure retaining bolted connections, and the moisture barrier, which prevents moisture intrusion at the concrete-to-metal interface at the basement floor. Also, the containment surfaces that may require augmented examination are included in this program.

In accordance with the NRC final rule amending 10 CFR 50.55a that was effective September 9, 1996, the IWE Program was developed with an initial interval start date of May 22, 1998. As required by the rulemaking, the 1992 Edition, 1992 Addenda of ASME Code Section XI was the basis for the programs. The required Subsection IWE examinations were completed for the first 10 year interval. The 2nd interval was scheduled to end with the end of the original operating license on February 21, 2014. In December 2010, DAEC received an extension of the operating license for 20 years. The inspection interval has been modified to be parallel to the 4th Ten Year ISI Program interval. The three inspection periods during the second inspection interval are as follows:

First Period: May 22, 2008 - May 21, 2010
Second Period: May 22, 2010 - October 31, 2013
Third Period: November 1, 2013 - May 21, 2017

The required Subsection IWE examinations are scheduled and tracked using a database. The current containment inspection interval is summarized in the table below:

Current IWE Interval

System Identification	Examination Description	Item Number	Exam Method	Period Scheduled		
Examination Category E-A				1	2	3
Drywell/Torus/ Downcomers	Accessible Surface Areas	E1.11	GV	1	1	1
Torus	Wetted Surfaces of Submerged Areas	E1.12	GV			1
Downcomers	BWR Vent System Accessible Surfaces	E1.20	GV	3	3	2
Drywell/Torus/ Downcomers	Moisture barrier	E1.30	GV	1	1	1

Examination Category E-C				1	2	3
Torus	Visible Surfaces	E4.11	VT-3	1		1
Torus	Surface Area Grid Minimum Wall Thickness Location	E4.12	UTT			

Item Number refers to item numbers listed in ASME Code Section XI, Table IWE-2500-1, titled "Examination Categories."
Exam Method GV - General Visual; UTT- Ultrasonic Thickness Test; and VT -3- examination method defined in ASME Code Section XI, Paragraph IWA-2213, "VT -3 Examination" Schedule.

Containment Surfaces Subject To Augmented Examinations

ASME Code Section XI paragraph IWE-1240 identifies containment surface areas requiring augmented examination as those surface areas likely to experience accelerated degradation and aging. Such areas include: (a) interior and exterior containment surface areas that are subject to accelerated corrosion with no or minimal corrosion allowance, or areas where the absence or repeated loss of protective coatings has resulted in substantial corrosion and pitting and (b) interior and exterior containment surface areas that are subject to excessive wear from abrasion or erosion that causes a loss of protective coatings, deformation, or material loss.

The submerged portion of the suppression pool at DAEC has been determined to be a surface area subject to augmented examination. In 2009, the general visual inspection frequency was increased to every outage. During the refueling outages in 2009 and 2010 only localized areas of corrosion (pitting) were observed and areas of significant depth were examined by UTT and determined to be acceptable. Significant areas of loss of protective coatings resulted in the recoating of the submerged areas during the 2012 refueling outage. The entire surface exposed after coating

removal was visually examined. Detailed evaluations were performed of the substrate to determine the acceptable thickness after coating removal. Nineteen localized areas were identified that required weld repair to restore the shell to an acceptable thickness. The entire submerged surface was recoated. These nineteen areas were inspected in the 2014 refueling outage and the examination was satisfactory and these nineteen areas no longer require successive inspections and have been removed from the augmented inspection requirements.

However, areas of coating degradation were identified in the 2014 refueling outage and the augmented detailed visual requirements remain in place for the submerged surface area in the suppression pool.

3.2.2 Containment Visual Inspections

Inspection Description

A Suppression Chamber and Drywell Visual Examination procedure for DAEC is utilized to perform general visual observations of the accessible interior and exterior surfaces of the primary containment in order to identify evidence of deterioration that may affect the containment integrity or leak tightness in accordance with the following.

- TS Surveillance Requirement 3.6.1.1.1 requires, in part, visual examinations in accordance with the Containment Leak Rate Testing Program.
- TS 5.5.12.b requires, in part, visual examinations in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. (Regulatory Position 3 requires that 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.)
- ASME Code Section XI, Subsection IWE requires visual examinations. General visual observations of the accessible interior and exterior surfaces of the primary containment are performed on a frequency that meets ASME Code Section XI, Subsection IWE, and 10 CFR 50 Appendix J, Option B. A Suppression Chamber Visual Examination of Submerged Areas procedure is used specifically to meet the ASME Code Section XI Subsection IWE submerged area examination requirements.

With the implementation of the proposed change, TS 5.5.12 will be revised by replacing the reference to Regulatory Guide 1.163 with reference to NEI 94-01, Revision 3-A. A general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity is required by NEI 94-01, Revision 3-A, prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years.

Recent Examination Results

The following is a summary of results of examinations that were recently performed.

An examination was successfully completed during the fall 2014 refueling outage (RFO24) for both the exterior surfaces and interior steel surfaces of the DAEC primary containment exposed to the atmosphere. The conditions identified were minor in nature and would not affect the integrity of the

primary containment. Identified conditions were documented in the inspection report and do not require action. The typical condition noted was localized light rusting, some minor paint chipping and flaking.

An examination was completed of the submerged surfaces areas of the suppression pool during the fall 2014 refueling outage (RFO24). This inspection identified delamination of the coating on the torus shell, structural steel and downcomers. There was no evidence of degradation of the torus shell itself or other steel components. The delaminated coating was removed. Additional detail on the remaining condition is provided in Section 3.2.8.

Conclusion

DAEC primary containment visual inspections were successfully completed during RFO24. Delaminated coating was discovered in the submerged portion of the suppression pool and removed to sound coating. Other identified deficiencies were accepted by an engineering evaluation or repaired in accordance with ASME Code Section XI, Subsection IWE. The DAEC primary containment continues to remain capable of performing its safety-related functions.

3.2.3 Containment Liner Test Channel Plugs

The U.S. Nuclear Regulatory Commission (NRC) issued information notice (IN) 2014-07 to inform addressees of issues identified by the NRC staff concerning degradation of floor weld leak-chase channel systems of steel containment shell and concrete containment metallic liner that could affect leak-tightness and aging management of containment structures.

The DAEC primary containment is a Mark I free standing steel containment vessel. The drawings of the DAEC containment were reviewed and no leak chase system was incorporated into the design of the DAEC containment.

3.2.4 Containment Corrosion

The NRC over the years has issued several information notices concerning containment corrosion. These notices have been reviewed as they were received by DAEC to determine the impact on the DAEC containment. Several instances have been cited where organic material that is left lodged between the containment and the surrounding has resulted in through-wall or significant corrosion. Other instances cited focus on the moisture barrier. These events have been evaluated and inspections have been conducted to determine the presence of these conditions at DAEC. Although containment visual inspections are required to be performed three times in a ten year period, DAEC has performed these visual inspections every refueling outage (five times in a ten year period). DAEC maintains the primary containment with an inerted internal atmosphere during plant operation. This has assisted DAEC in reducing the potential for corrosion..

DAEC has observed corrosion in the submerged area of the suppression pool. This has been monitored by the site staff for several years. The NRC issued IN 2011-15 and describes the corrosion condition as observed several years ago. Section 3.2.5 provides additional detail on this topic.

3.2.5 Suppression Chamber Corrosion

IN 2011-15 provides the following information on the corrosion of the submerged areas in the DAEC primary containment:

"During its review of the Duane Arnold Energy Center license renewal application (LRA), the NRC staff noted that since 1977, when the applicant performed the first inspection of the torus after the initial coating application, the applicant has found numerous instances of localized corrosion and depletion of the coating of the torus shell (see Enclosure Figures 1 and 2) of its boiling-water reactor (BWR) Mark I containment. The applicant performed repairs in 1980 and 1983 prior to a full recoat of the suppression chamber in 1985. Since 1988, the applicant has been repairing degraded coatings and managing and tracking the effects of aging of the torus in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, 'Rules for Inservice Inspection of Nuclear Power Plant Components,' Subsection IWE, 'Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants.' Since 1995, the torus coating has been repaired at more than 15,000 locations, all below the water line, which is equivalent to approximately 5 percent of the underwater coating surface inside the torus. The torus steel behind the degraded coating has corroded locally at some of these locations. The applicant evaluated this degradation of coating of the torus and determined that the degradation has not affected the structural integrity of the torus. In addition, the applicant performed detailed analysis and determined that localized corrosion of the torus shell was acceptable without repair. The NRC staff requested that the applicant address the possibility of localized galvanic corrosion (pitting) due to degraded coatings during the period of extended operation since the normal life of the underwater zinc coating is approximately 15 to 20 years and aging management of the torus steel minimizes the potential for pitting corrosion to extend through-wall. In response to NRC requests for additional information during the license renewal review, the applicant committed to recoating the suppression pool interior surfaces below the water line prior to startup from the first refueling outage during the period of extended operation (applicant letter, dated March 9, 2010, Agencywide Documents Access and Management System (ADAMS) Accession No. ML100700248). The applicant subsequently stated that the current project plan ensures that recoating will extend well above any fluctuations in water level, including the 2-foot wide splash band at the water level (applicant letter, dated April 2, 2010, ADAMS Accession No. ML100960277)."

The submerged areas in the DAEC primary containment were recoated in the fall of 2012. The sacrificial inorganic zinc coating was replaced with a 100% solids epoxy designed for the remaining plant life. This coating underwent qualification testing in accordance with approved industry standards to demonstrate its acceptability for the specified service as a safety related coating. The coating was intended to be installed in a single coat application but failures in the control of the application process resulted in the need to apply a 2nd coat in some areas to achieve the specified coating thickness. The initial coat and 2nd coat did not fully bond resulting in the delamination observed in the fall of 2014. This is discussed in greater detail in Section 3.2.8.

As part of the recoating process the existing inorganic zinc coating was removed and the substrate blasted to a near white condition. The entire surface exposed after coating removal was visually examined. Detailed evaluations were performed to determine the acceptable metal thickness after coating removal. Ultrasonic thickness readings were taken on all shell plates to determine the base thickness. Additionally, localized ultrasonic thickness readings were taken when significant pitting was observed. Nineteen localized areas were identified that required weld repair to restore the shell to an acceptable thickness. The entire submerged surface was recoated.

The nineteen areas that required repair in 2012 were inspected in the 2014 refueling outage and the examination was satisfactory and these nineteen areas no longer require specific inspections.

In summary, the corrosion condition described in IN 2011-15 has been corrected. The sacrificial coating has been replaced. The substrate has been examined and has an acceptable thickness in all locations.

3.2.6 Suppression Chamber Cracking

The U.S. Nuclear Regulatory Commission (NRC) issued IN 2006-01 to inform the owners of BWR Mark I containments about the occurrence and potential causes of the through-wall cracking of a torus in a BWR Mark I containment.

The FitzPatrick licensee performed a root cause investigation of the event, and after eliminating a number of possible causes (thermal fatigue, clearing load phenomena, metallurgical discontinuity, weld defects, corrosion, flow-induced phenomena, flow-accelerated corrosion, cavitation, and direct jet impingement), the licensee concluded that the most likely cause for the initiation and propagation of the crack was the hydrodynamic loads of the turbine exhaust pipe during HPCI operation coupled with the highly restrained condition of the torus shell at the torus column support.

The DAEC design has been reviewed and the turbine exhaust pipe configuration and hydrodynamic loading are sufficiently different at DAEC to eliminate the potential for this type of event.

3.2.7 Inaccessible Areas

For a Class MC application, DAEC must evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, DAEC will provide the following in the ISI Summary Report, as required by 10 CFR 50.55a(b)(2)(ix)(A):

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation, and;
- A description of necessary corrective actions.

DAEC has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, DAEC actively participates in various nuclear utility owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to DAEC. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

3.2.8 Containment Coatings Inspections

The site Protective Coatings Program defines the requirements and responsibilities for a program to implement inspections during refueling outages for the purpose of assessing the condition of the protective coatings on structures and equipment in the primary containment. These inspections assure compliance with the DAEC commitments in response to NRC Generic Letter 98-04. DAEC is not committed to following the requirements of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," but has developed a comparable program for monitoring and maintaining protective coatings inside primary containment. The DAEC program uses specific ASTM Standards that are acceptable to the NRC as stated in RG 1.54 Revision 1.

Containment coatings inspections are a scheduled activity conducted during refueling outages. The examination areas are selected such that painted surfaces are inspected every outage. This is done to comply with the recommendations of ASTM D5163-96, "Establishing Procedures to Monitor the Performance of Service Level 1 Coating Systems in an Operating Nuclear Power Plant."

Identified, degraded, or questionable coatings shall be remediated prior to the unit entering Mode 3 at the end of an outage. The remediation may include recoating the affected area with a qualified coating system, or removal of the degraded or questionable coatings to a sound and tightly adhered condition. Any coatings that are left as-is are evaluated, approved, and logged. DAEC maintains an unqualified coating log to track unqualified coatings materials, and degraded qualified coatings. This unqualified coating log is maintained to ensure the quantity of unqualified coating is less than the acceptable quantity of unqualified coating that could potentially reach the ECCS suction strainers in a design basis accident scenario.

Results of Recent Coatings Inspections - Fall 2014 Refueling Outage

The condition of the protective coatings in the primary containment air space inspected during fall 2014 refueling outage was typical and expected for the vintage of the coatings. In general the coatings were found to be performing well. Conditions identified were minor in nature and limited in their extent. The coatings were found to be performing acceptably and no negative trends were identified besides minor spots of surface corrosion.

A coating examination was completed of the submerged surfaces areas of the suppression pool during the fall 2014 refueling outage (RFO24). This inspection identified delamination of the coating on the torus shell, structural steel and downcomers. The delamination was the result of inadequate bonding between coats during the torus recoat in the fall of 2012.

The submerged coating originally installed was a sacrificial inorganic zinc coating. The submerged coating was reapplied in 1985 and the coating required replacement again in 2012. The replacement coating for the 2012 recoat was a single coat 100% solids epoxy. The application was intended as a single coat application but failures in the control of the application process resulted in the need to apply a 2nd coat in some areas to achieve the specified coating thickness. The initial coat and 2nd coat did not fully bond resulting in the delamination observed in the fall of 2014.

The delaminated coating was removed and adhesion testing was performed to determine that the remaining coating was adequately adhered. The coating thickness that was specified was reviewed and a minimum acceptable coating thickness was determined. Coating thicknesses of greater than 18 mils will supply required corrosion protection. The areas where the coating thickness was below the minimum acceptable were determined to be unqualified coating and added to the unqualified coating log.

Additionally, areas were identified on the ring girders, supports, and downcomers that were without coating. Areas of the shell were identified that required coating application. The shell areas were recoated in the fall of 2014 and the remaining areas will be recoated during the next refueling outage in the fall of 2016.

The operability of the primary containment and the ECCS were assessed prior to startup from the fall of 2014 refueling outage. The evaluation of ECCS suction strainer loading was reassessed and sufficient margin was present to determine the as-found degraded condition as operable and the as-left condition was also determined to be operable.

3.2.9 License Renewal Commitments

License renewal activities led to one commitment related to the DAEC containment. The following provides a status of the action that was identified and committed to by DAEC in the DAEC License Renewal Application.

DAEC Updated Final Safety Analysis Report (UFSAR) Commitment 50:

Perform recoating of suppression pool interior surfaces below the water line. Complete recoating prior to startup from the first refuel outage during the period of extended operation.

Status:

The suppression pool interior surfaces below the water line were recoated in refueling outage 23 (RFO23) in October 2012. This was the last outage prior to entering the period of extended operation and meets the scheduling requirement.

3.3 NRC INFORMATION NOTICE 92-20, "INADEQUATE LOCAL LEAK RATE TESTING"

NRC IN 92-20 was issued to alert licensees to problems involving local leak rate testing of containment penetrations under 10 CFR 50, Appendix J. Problems were identified with the testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

The bellows at DAEC are designed to ASME Section III – 1971 edition, and have two independent corrugated stainless steel elements – i.e., no section of straight pipe between them. The corrugated elements are constructed with two plies. Sandwiched between the two plies is a wire mesh, which assures that an annulus will be maintained throughout the entire bellows surface.

The flexible metallic bellows are tested during the ILRT. The ILRT pressurizes the area between the bellows and the guard pipe. If the inner ply of the corrugated elements is leak tight, the outer ply does not see the test pressure. The corrugated elements are tested by a LLRT. The LLRT pressurizes the area between the plies. At DAEC, the test is configured so that both corrugated elements are tested at the same time. Isolation valves have been provided so that each element can be tested independently. The test pressurizes both elements through a common supply and a common pressure gauge is installed at the other end of each element.

In 1992, in response to NRC IN 92-20, a test was performed for several penetrations to determine that restrictions were not present in the element to preclude a valid LLRT. This test measured the time to achieve pressure equalization in each element.

This test is similar to the test required by Specification BECH-MRS-M126 paragraph 6.2.1.b. Per that test stabilization time shall not exceed 3 minutes. The acceptance requirement for the test was no observable pressure drop in one minute. The stabilization time requirement ensures that there is no restriction in the plies and the pressure drop requirement ensures the test boundary is leak tight. This is accomplished by a simple pressure decay test.

The testing of two ply bellows was discussed with the bellows manufacturer. The time to achieve pressure equalization depends on the profile of the bellows. On a typical bellows of our size the pressurization only takes seconds. Manufacturer quality control personnel typically use a value of 10 minutes as a time of concern unless a time is specified by the customer. Based on this discussion, the previous methodology used by DAEC to verify an adequate gap was sufficient.

In 2005, DAEC repeated the testing on all penetrations with an established acceptance criteria of not to exceed 3 minutes to achieve an outlet pressure within 0.2 psig of the inlet pressure. All of the expansion bellows penetrations were tested in 2005 with satisfactory results since all of the stabilization times were less than or equal to 41 seconds.

The special testing performed in 1992 and in 2005 was similar to the manufacturing acceptance test performed per the specification. This test is appropriate to demonstrate that the penetration bellows has sufficient space between the plies to perform its design function.

3.4 NRC LIMITATIONS AND CONDITIONS

3.4.1 June 25, 2008 NRC Safety Evaluation

The limitations and conditions from the June 25, 2008 safety evaluation to NEI 94-01 Revision 2 are presented in the table below with the NextEra Energy Duane Arnold response for DAEC.

June 25, 2008 NRC Safety Evaluation (SE) Limitations and Conditions

Limitation/Condition (From Section 4.0 of Safety Evaluation)	Response for DAEC
1. For calculating the Type A leakage rate, the licensee should use the definition in NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1).	DAEC will utilize the definition in NEI 94-01, Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3)	Reference Section 3.2.1 and 3.2.2. General visual observations of the accessible interior and external surfaces of the containment structure shall continue to be performed in accordance with containment structural integrity test procedures to meet the requirements of the proposed revision to TS 5.5.12, the inspection requirements of ASME Code Section XI, subsection IWE and NEI 94-01, Revision 3.A, Sections 9.2.1 and 9.2.3.2.
3. The licensee addresses the areas of containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).	Reference Section 3.2.1 through 3.2.9. General visual observations of the accessible interior and external surfaces of the containment structure shall continue to be performed in accordance with containment structural integrity test procedures to meet the requirements of the proposed revision to TS 5.5.12, the inspection requirements of ASME Code Section XI, subsection IWE and NEI 94-01, Revision 3.A, Sections 9.2.1 and 9.2.3.2.

Limitation/Condition (From Section 4.0 of Safety Evaluation)	Response for DAEC
4. The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).	Engineering Change (EC) 281991 is to install a new Hardened Containment Vent System (HCVS). The design will remove the existing 8" containment isolation control valve CV-4357. The new cap installed on the remaining 8"-HBC-140 piping within the SE corner room will be the containment boundary. The modification adds two new 10" PCIVs and actuators and a new rupture disk. The two new PCIVs provide a containment isolation function. The rupture disk prevents the use of this system prior to the containment pressure exceeding 50 psig, unless the rupture disk is manually ruptured. The new pipe and valves are the containment penetration boundaries. The system is manually operated from the control room or remote location. Associated tests and inspections will confirm the leak tightness of the abandon penetration, the new PCIVs, and the piping from the containment to the new PCIVs. Testing procedures have yet to be developed.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions of section 9.1 or NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2).	DAEC will follow the requirements of NEI 94-01, Revision 3-A, Section 9.1. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. In accordance with section 3.1.1.2 of the NRC safety evaluation dated June 25, 2008 (ADAMS Accession No. ML 081140105), NextEra Energy Duane Arnold will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15 year interval is required. Justification for such an extension request will be in accordance with the staff position in Regulatory Issue Summary (RIS) 2008-27.
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design has been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and [Electric Power Research Institute] EPRI Topical Report No. TR-1009325, Revision 2, ["Risk-Impact Assessment of Extended Integrated Leak Rate Testing Intervals,"] including the use of past containment ILRT data.	Not applicable. DAEC was not licensed under 10 CFR Part 52.

3.4.2 June 8, 2012 NRC Safety Evaluation

The two conditions from Section 4.0 of the June 8, 2012 safety evaluation to NEI 94-01 Revision 3 are stated below with the NextEra Energy Duane Arnold response for DAEC.

Condition 1

NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs to be increased to 75 months with the requirement that a licensee's post-outage report include the margin between Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84 months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g., BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months. This is Topical Report Condition 1.

Response to Condition 1

Condition 1 presents three (3) separate issues that are addressed as follows:

ISSUE 1 – The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit.

Response to Condition 1, Issue 1

The post-outage report shall include the margin between the Type B and Type C minimum pathway leak rate summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 L_a.

ISSUE 2 – A corrective action plan shall be developed to restore the margin to an acceptable level.

Response to Condition 1, Issue 2

When the potential leakage understatement adjusted Type B and Type C minimum pathway leak rate total is greater than the DAEC administrative leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.60 L_a, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the DAEC administrative leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and the manner of timely corrective action (as deemed appropriate) that best focuses on the prevention of future component leakage performance issues.

ISSUE 3 – Use of the allowed 9 month extension for eligible Type C valves is only authorized for non-routine emergent conditions.

Response to Condition 1, Issue 3

DAEC will apply the 9 month grace period only to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

<u>Condition 2</u>

The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leak rates for the just tested penetrations are summed with the as-left minimum pathway leak rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves which, in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1. When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations. This is Topical Report Condition 2.

Response to Condition 2

Condition 2 presents two separate issues that are addressed as follows:

ISSUE 1 – Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1.

Response to Condition 2, Issue 1

The change in going from a 60 month extended test interval for Type C tested components to a 75 month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25 percent in the local leak rate test periodicity. As such, NextEra Energy Duane Arnold will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the as-left leakage total for each Type C component currently on the 75 month extended test interval. This will result in a combined conservative Type C total for all 75 month local leak rate tests being carried forward and included

whenever the total leakage summation is required to be updated (either while operating on-line or following an outage). When the potential leakage understatement adjusted leak rate total for those Type C components being tested on a 75 month extended interval is summed with the non-adjusted total of those Type C components being tested at less than the 75 month interval and the total of the Type B tested components, if the minimum pathway leak rate is greater than the DAEC administrative leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.60 L_a, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the administrative leakage limit. The corrective action plan shall focus on those components that have contributed the most to the increase in the leakage summation value and the manner of timely corrective action (as deemed appropriate) that best focuses on the prevention of future component leakage performance issues.

ISSUE 2 – When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Response to Condition 2, Issue 2

If the potential leakage understatement adjusted minimum pathway leak rate is less than the administrative leakage summation limit of 0.50 L_a, then the acceptability of the 75-month local leak rate test extension for all affected Type C components has been adequately demonstrated and the calculated local leak rate total represents the actual leakage potential of the penetrations.

In addition to Condition 1, Issues 1 and 2, which deal with the minimum pathway leak rate Type B and Type C summation margin, NEI 94-01, Revision 3-A, also has the following margin related requirement contained in Section 12.1, "Report Requirements."

A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and shall be available on-site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C leakage rate summation shall be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

In the event an adverse trend in the potential leakage understatement adjusted Type B and Type C summation is identified, an analysis and a corrective action plan shall be prepared to restore the margin to an acceptable level thereby eliminating the adverse trend. The corrective action plan shall focus on those components that have contributed the most to the adverse trend in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

An adverse trend is defined as three consecutive increases in the final pre-reactor coolant system Mode change Type B and Type C minimum pathway leak rate summation value adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of L_a.

3.5 PLANT-SPECIFIC CONFIRMATORY ANALYSIS

3.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the DAEC Type A test interval from the current 10 years to 15 years. A simplified bounding analysis consistent with the Electric Power Research Institute (EPRI) approach was used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in:

- Appendix H of Electric Power Research Institute, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325," EPRI Topical Report TR-1018243, dated October 2008,
- Electric Power Research Institute, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," EPRI Topical Report TR-104285, dated August 1994,
- Nuclear Regulatory Commission, "Performance-Based Containment Leak-Test Program," NUREG-1493, dated September 1995, and the
- Calvert Cliffs liner corrosion analysis described in a letter to the NRC dated March 27, 2002 (ADAMS Accession No. ML020920100).

The analysis uses results from a Level 2 analysis of core damage scenarios from the current DAEC probabilistic risk assessment models and subsequent containment responses resulting in various fission product release categories (including intact containment or negligible release) to determine large early release frequency (LERF).

In the safety evaluation issued by NRC letter dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their Technical Specifications to permanently extend the Type A surveillance test interval to 15 years, subject to the conditions noted in Section 4.2 of the safety evaluation. The following table addresses each of the four conditions for the use of EPRI TR-1 009325, Revision 2.

EPRI TR-1009325, Revision 2, Limitations and Conditions

Conditions (From Section 4.2 of Safety Evaluation)	Response for DAEC
1. The licensee submits documentation indication that the technical adequacy of their (probabilistic risk assessment) PRA is consistent with the requirements of [Regulatory Guide] RG 1.200 relevant to the [integrated leakage rate test] ILRT extension application.	DAEC PRA technical adequacy is addressed in Section 3.5.2.
2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this [safety evaluation] SE.	EPRI TR-1009325, Revision 2-A, incorporates these population dose and conditional containment failure probability acceptance guidelines, and these guidelines have been used for the DAEC plant specific assessments.

Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. *In addition, a small increase in [conditional* containment failure probability] CCFP should be defined as a value marginally greater than that accepted in a previous one-time 15 year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points.	The increase in population dose is discussed in Section 3.5.3. The increase in the conditional containment failure probability is discussed in Section 3.5.3.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation in the increase in expected population dose (per year of reactor operation). In order to make the methodology *acceptable, the average leak rate for the pre-* existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.	EPRI TR-1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident class 3b), and this value has been used in the DAEC plant *specific risk assessment.*
4. A [license amendment request] LAR is required in instances where containment over-pressure is relied upon for [emergency core cooling system] ECCS performance.	DAEC does not rely on containment overpressure.

3.5.2 Probabilistic Risk Assessment (PRA) Technical Adequacy

DAEC has Level 2 models that include both internal and external events. Severe accident sequences have been developed from internally and externally initiated events, including internal floods and internal fires. The sequences have been developed to determine the frequency for the radiological release end states to the environment. Information developed for Revision 6 of the PRA to support the Level 2 release categories is also used in this analysis.

The DAEC PRA models are highly detailed and include a wide variety of initiating events, modeled systems, operator actions, and common cause events. The DAEC model of record and supporting documentation has been maintained as a living program, with periodic updates to reflect the as-built, as-operated plant. The DAEC Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) PRA models underwent NRC reviews, and updates to these models have been the subject of several assessments to establish the technical adequacy of the PRA. Documentation of DAEC technical adequacy was previously submitted to the NRC in support of the license amendment request for adoption for NFPA-805 and Risk Informed TS Initiative 5b. This previously submitted documentation of PRA technical adequacy is deemed to be applicable for this Type A test interval extension submittal.

3.5.3 Conclusion of Plant-Specific Risk Assessment Results

The findings of the DAEC risk assessment confirm the general findings of previous studies that the impact associated with extending the Type A test interval from three in ten years to one in 15 years is small. The DAEC plant-specific results for extending the Type A test interval from the current 10 years to 15 years is summarized below.

Core damage frequency (CDF) is not significantly impacted by the proposed change. DAEC does not rely on containment overpressure to assure adequate net positive suction head is available for emergency core cooling system pumps taking suction from the containment sump following design basis accidents.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF less than 1.0×10^{-6} per reactor year and increases in LERF less than 1.0×10^{-7} per reactor year.

There was essentially no change in total (internal and external) CDF, which meets Regulatory Guide 1.174 acceptance guidelines for very small changes in CDF, and confirms that the impact on CDF from the Type A test extension is negligible. Thus, the relevant acceptance criterion is LERF.

The increase in LERF based on consideration of internal events only resulting from a change in the Type A test interval from three in ten years to one in fifteen years with corrosion included is conservatively estimated as 2.57×10^{-8} per year, which falls within the very small change region of the acceptance guidelines in Regulatory Guide 1.174. Regulatory Guide 1.174 also states that when the calculated increase in LERF is in the range of 1.0×10^{-6} to 1.0×10^{-7}per reactor year, applications will be considered only if can be reasonably shown that the total LERF is less than 1.0×10^{-5} per reactor year. When external events contribution is also considered, the total increase in LERF due to both internal and external events including corrosion goes to 8.14×10^{-8} per year, with associated total LERF of 9.79×10^{-6} per year. As such, the estimated change in total LERF is determined to be small using the acceptance guidelines of Regulatory Guide 1.174 acceptance criteria for total LERF of 1.0×10^{-5}. Sensitivity analysis using EPRI Expert Elicitation methodology, estimate the change in total LERF as 2.56×10^{-8} per year, which falls within the very small region.

The calculated increase in total 50-mile population dose risk for changing the Type A test frequency from three per 10 years to once per 15 years is measured as an increase to the total integrated dose risk for all accident sequences. The total 50-mile population dose risk increase (relative to the base case with corrosion) is 1.55×10^{-2} person-rem per year using the EPRI guidance. EPRI TR-1009325, Revision 2-A, states that a very small population dose is defined as an increase of less than or equal to 1.0 person-rem per year, or less than or equal to 1 percent of the total population dose, whichever is less restrictive. Thus, the calculated 50-mile population dose increase at DAEC is small using the guidelines of EPRI TR-1009325, Revision 2-A. Moreover, the risk impact when compared to other severe accident risks is negligible.

The increase in the conditional containment failure probability from the three per 10 years to once in 15 years interval including corrosion effects is 0.61 percent for DAEC. EPRI TR-1009325, Revision 2-A, states that increases in conditional containment failure probability of less than or equal to 1.5 percentage points are very small. Therefore this increase is judged to be very small at DAEC.

Increasing the Type A test interval on a permanent basis to a once in fifteen years frequency is not considered to be significant since it represents only a small change in DAEC risk profiles.

Details of the DAEC risk assessments are contained in Attachment 4 for DAEC.

3.6 CONCLUSION

NEI 94-01, Revision 3-A, describes an NRC accepted approach for implementing the performance-based requirements of Appendix J, Option B. It incorporates the regulatory positions stated in Regulatory Guide 1.163 and includes provisions for extending Type A test intervals to 15 years and Type C test intervals to 75 months. NEI 94-01, Revision 3-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies.

Based on the previous Type A tests conducted at DAEC, extension of the containment Type A test interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Appendix J, Option B, and the overlapping inspection activities performed as part of the following DAEC inspection programs:

- Primary Containment Inservice Inspection Program
- Containment Coatings Inspection and Assessment Program

This experience is supplemented by risk analysis studies, including the DAEC risk analysis provided in Attachment 4. The findings of the risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the Type A test interval from 10 to 15 years results in a very small change to the DAEC risk profile.

4.0 REGULATORY SAFETY ANALYSIS

4.1 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NextEra Energy Duane Arnold has evaluated the proposed changes to the Technical Specifications (TS) using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Description of Amendment Request: An amendment is proposed to the Duane Arnold Energy Center (DAEC) Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program." The proposed amendment to the TS would revise DAEC TS 5.5.12, by replacing the reference to Nuclear Regulatory Commission (NRC) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A, as the implementation document used by DAEC to implement the performance-based containment leakage rate testing program.

Basis for proposed no significant hazards determination: As required by 10 CFR 50.91(a), the NextEra analysis of the issue of no significant hazards consideration is presented below:

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

Response: No

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," for development of the DAEC performance-based containment testing program. NEI 94-01 allows, based on risk and performance, an extension of Type A and Type C containment leak test intervals. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses.

The findings of the DAEC risk assessment confirm the general findings of previous studies that the risk impact with extending the containment leak rate is small. Per the guidance provided in Regulatory Guide 1.174, an extension of the leak test interval in accordance with NEI 94-01, Revision 3-A results in an estimated change within the very small change region.

Since the change is implementing a performance-based containment testing program, the proposed amendment does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The requirement for containment leakage rate acceptance will not be changed by this amendment. Therefore, the containment will continue to perform its design function as a barrier to fission product releases.

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

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

Response: No

The proposed change to implement a performance-based containment testing program, associated with integrated leakage rate test frequency, does not change the design or operation of structures, systems, or components of the plant.

The proposed changes would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses. There are no accident initiators created or affected by these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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 change involve a significant reduction in a margin of safety?

Response: No

Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. The proposed change to implement a performance-

based containment testing program, associated with integrated leakage rate test frequency, does not affect plant operations, design functions, or any analysis that verifies the capability of a structure, system, or component of the plant to perform a design function. In addition, this change does not affect safety limits, limiting safety system setpoints, or limiting conditions for operation.

The specific requirements and conditions of the TS Primary Containment Leakage Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained. This ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by implementation of a performance-based containment testing program.

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

4.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Nuclear Power Reactors." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values and periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. In addition, Appendix J discusses leakage rate test methodology, frequency of testing, and reporting requirements for each type of test.

Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," (September 1995) provides a method acceptable to the NRC for implementing the performance-based option (Option B) of 10 CFR 50, Appendix J. The regulatory positions stated in Regulatory Guide 1.163 (September 1995) as modified by NRC Safety Evaluations of June 25, 2008 (ADAMS Accession No. ML081140105) and June 8, 2012 (ADAMS Accession No. ML121030286) are incorporated in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

The proposed license amendment would revise DAEC TS 5.5.12, "Primary Containment Leakage Rate Testing Program," Item b, by changing the wording to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A, instead of Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," and the listed Type A test exception.

The purpose of NEI 94-01 is to assist licensees in the implementation of Option B to 10 CFR Part 50, Appendix J. The NRC staff has reviewed NEI 94-01, Revision 3, and found that this

guidance, as modified to include two limitations and conditions, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing.

NextEra has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. Based on the foregoing, the proposed amendment will continue to ensure compliance with 10 CFR 50.54(o), and Option B of 10 CFR Part 50, Appendix J.

4.3 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 continue to 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

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment, if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. NextEra has reviewed this license amendment request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination is as follows.

Basis

This change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment does not involve a significant hazards consideration.

The proposed amendment does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed amendment does not change or modify the design or operation of any plant systems, structures, or components. The proposed amendment does not affect the amount or types of gaseous, liquid, or solid waste generated onsite. The proposed amendment does not directly or indirectly affect effluent discharges.

The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed amendment does not change or modify the design or operation of any plant systems, structures, or components. The proposed amendment does not directly or indirectly affect the radiological source terms.

6.0 PRECEDENT

This License Amendment Request is similar to a License Amendment Request approved by letter dated April 8, 2015 (ML15078AA058), "Beaver Valley Power Station, Unit Nos. 1 and 2 - Issuance of Amendment Re: License Amendment Request to Extend Containment Leakage Rate Frequency (TAC NOS. MF3985 and MF3986)."

7.0 REFERENCES

7.1 Nuclear Energy Institute (NEI) Topical Report, 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 (ML12221A202)

7.2 Letter from S. Bahadur (NRC) to B. Bradley (NEI), "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, 'Industry Guideline for Implementing Performance-based Option of 10 CFR Part 50, Appendix J,' (TAC No. ME2164)," dated June 8, 2012 (ML121030286)

7.3 Nuclear Energy Institute (NEI) Topical Report, 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008 (ML100620847)

7.4 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011 (ML100910006)

7.5 Electric Power Research Institute, TR-1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007 (ML072970208)

ATTACHMENT 2 to NG-15-0234

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-143)
EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY

PROPOSED TECHNICAL SPECIFICATIONS CHANGES
(MARKUP COPY)

1 page follows

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (1) and (2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions.

b. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":~~

NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A, as modified by the following exceptions:

1. ~~The first Type A test after the September 1993 Type A test shall be performed no later than September 2008.~~

DELETED

(continued)

ATTACHMENT 3 to NG-15-0234

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-143)
EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY

REVISED TECHNICAL SPECIFICATIONS PAGES

1 page follows

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (1) and (2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions.

b. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A, as modified by the following exceptions:

1. DELETED

(continued)

ATTACHMENT 4 to NG-15-0234

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-143)
EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY

PLANT SPECIFIC RISK ANALYSIS

57 pages follow

Duane Arnold Energy Center

PRA Evaluation

Permanent ILRT Extension Risk impact Assessment

Risk Applications and Methods II

Contents

1 Purpose of Analysis

1.1 Purpose

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) interval to a permanent fifteen (15) years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for Duane Arnold Energy Center (DAEC). The risk assessment follows the guidelines from NEI 94-01 (Reference 1), the methodology used in EPRI TR-104285 (Reference 2), the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 3), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 (Reference 30) as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4), the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion induced leakage of steel liners going undetected during the extended test interval (Reference 5), and the methodology used in EPRI 1009325, Revision 2-A (Reference 20).

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage rate was less than the limiting containment leakage rate of 1La[1].

The basis for the current fifteen (15) year test interval is provided in Section 11.0 of Reference 1, and was established in 2008. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 (Reference 6), provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

[1] La (percent/24 hours) is the maximum allowable leakage rate at pressure Pa (calculated peak containment internal pressure related to the design basis accident) as specified in the technical specifications.

The NRC report on performance-based leak testing, NUREG-1493 (Reference 6), analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant, containment isolation failures contribute less than 0.1% to the latent risks from reactor accidents. Consequently, it is required to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for DAEC.

The Guidance provided in Appendix H of EPRI Report No. 1009325, *"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,"* (Reference 20) for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. 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..

The associated change to NEI 94-01 (Reference 1) will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.3 Criteria

The acceptance guidelines in RG 1.174 (Reference 4) are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in Core Damage Frequency (CDF) less than 10^{-6} per reactor year and increases in Large Early Release Frequency (LERF) less than 10^{-7} per reactor year. As the DAEC Level I PRAs do not credit containment features, the Type A test does not impact CDF. Therefore, the relevant risk metric is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year. RG 1.174 discusses defense in depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense in depth philosophy, are met. Therefore, the increase in the Conditional Containment Failure Probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also

calculated. The criteria described below are taken from the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325, Revision 2 (Reference 25).

Regarding Conditional Containment Failure Probability (CCFP), the NRC concluded that a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one time fifteen (15) year ILRT extension requests. To this end the NRC has endorsed a small increase in CCFP as an increase in CCFP less than or equal to 1.5% (Reference 25).

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. The NRC concluded that for purposes of assessing the risk impacts of the Type A ILRT extension in accordance with the EPRI methodology, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1% of the total population dose, whichever is less restrictive (Reference 25).

2 References

1. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01, Revision 2-A, November 2008.
2. Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.
4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Revision 2, May 2011.
5. Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6. Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.
7. Evaluation of Severe Accident Risks: Peach Bottom Unit 2, Main Report NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 1990.
8. Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
10. Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

11. Reliability Analysis of Containment Isolation Systems, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

12. Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.

13. Review of Light Water Reactor Regulatory Requirements, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

14. Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.

15. Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, December 1990.

16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

17. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.

18. Letter from J.A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.

19. Letter from D.E. Young (Florida Power, Crystal River) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.

20. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA: 2008 (issued under EPRI TR-1018243).

21. Letter from P.P. Sena III (FENOC) to Document Control Desk (NRC), dated June 18, 2009, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, LER 2009-003-00, "Containment Liner Through Wall Defect Due to Corrosion."

22. Letter from E.A. Larson (FENOC) to Document Control Desk (NRC), dated February 14, 2014, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, LER 2013-002-01, "Containment Liner Through Wall Defect Discovered During Planned Visual Inspection."

23. Letter from J.E. Pollock (AEP Indiana Michigan Power) to Document Control Desk (NRC), dated March 16, 2001, submitting LER 316/2000-001-01, "Through-Liner Hole Discovered in Containment Liner."

24. Duane Arnold Energy Center Individual Plant Examinations of External Events (IPEEE), November 1995.

25. Final Safety Evaluation For NEI Topical Report 94-01 Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" and EPRI Report No. 1009325 Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Test Intervals".

26. 0493080001.004, Rev. 3, Duane Arnold Energy Center (DAEC) Fire Probabilistic Risk Assessment Quantification.

27. LTR-RAM-II-15-013, Revision 1 Duane Arnold Energy Center Inputs used for Integrated Leak Rate Test Interval Extension.

28. Virgil C. Summer Nuclear Station, Unit 1 – Issuance of Amendment Extending Integrated Leak Rate Test Interval (TAC No. MF1385), February 5th, 2014. Adams ML13326A204.

29. DAEC-PSA—L2-15 DAEC PSA Level 2 PSA Analysis, Revision 3, February 2011.

30. An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Regulatory Guide 1.200, Revision 2, March 2009.

31. NFPA 805 "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2015.

32. EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, August 1991.

33. Professional Loss Control, Inc., Fire-Induced Vulnerability Evaluation (FIVE), TR-100370, EPRI Research Project 3000-41, Revision 1, September 1993.

34. USNRC Letter to Mr. Richard L. Anderson, "Duane Arnold Energy Center – Issuance of Amendment Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) (TAC No. ME6818)", Docket No. 50-331, Amendment No. 286 to DPR-49, September 10, 2013.

35. ERIN Report Number 0493080001.006, "Duane Arnold Energy Center (DAEC) Fire Probabilistic Risk Assessment NFPA 805 RAI Model Updated Quantification Report," Revision 3.

2.1 Acronyms

ANS	American Nuclear Society
APB	Accident Progression Bin
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analyzer
CCFP	Conditional Containment Failure Probability
CD	Core Damage
CDF	Core Damage Frequency
CET	Containment Event Tree
CLRT	Containment Leak Rate Test
DAEC	Duane Arnold Energy Center
DCH	Direct Containment Heating
EPRI	Electric Power Research Institute
FIVE	Fire-Induced Vulnerability Evaluations

ILRT	Integrated Leak Rate Test
IPE	Individual Plant Examination
IPEEE	Individual Plant Examinations for External Events
ISI	In-Service Inspection
ISLOCA	Interfacing System Loss of Coolant Accident
IST	In-Service Testing
LER	Large Early Release
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SAMA	Severe Accident Mitigation Alternatives
SER	Safety Evaluation Report
SMA	Seismic Margins Assessment
SSE	Safe Shutdown Earthquake

3 Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2-A, *"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals"* (Reference 20), EPRI TR-104285 (Reference 2), NUREG-1493 (Reference 6) and the Calvert Cliffs liner corrosion analysis (Reference 5). The analysis uses results from the current DAEC Level 2 PRA models to establish frequency of fission product releases. Fission product release magnitudes are extrapolated from results of NUREG/CR-4551 (Reference 7) to account for plant specific characteristics. This risk assessment is applicable to DAEC.

The six (6) general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report No. 1009325, Revision 2-A (Reference 20).

2. Develop plant specific person-rem (population dose) per reactor year for each of the

eight containment release scenario types from plant specific consequence analyses.

3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.

4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 (Reference 4) and compare with the acceptance guidelines of RG 1.174.

5. Determine the impact of the ILRT interval extension on the Conditional Containment Failure Probability (CCFP) and the population dose and compare with the acceptance guidance of Reference 25.

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore:

- Consistent with the other industry containment leak risk assessments, the DAEC assessment uses LERF and delta LERF in accordance with the risk acceptance guidance of RG 1.174 (Reference 4). Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

- The evaluation for DAEC uses groundrules and methods to calculate changes in risk metrics that are consistent with those used in Appendix H of EPRI Report No. 1009325 (Reference 20), Revision 2-A, *"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals."*

4 Groundrules

The following groundrules are used in the analysis:

- The technical adequacy of the DAEC PRA models is consistent with the requirements of Regulatory Guide 1.200 (Reference 30) as is relevant to this ILRT interval extension.

- The current DAEC Level 1 and Level 2 internal events PRA models are explicitly used in this analysis to assess fission product release frequencies.

- It is appropriate to use the DAEC internal events PRA models as gauges to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume

that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations; this is evaluated in the sensitivity analysis which uses available information from the DAEC IPEEE (Reference 24) and DAEC Fire PRA (Reference 26).

- Dose results for the containment failures modeled in the PRA can be characterized by scaling information provided in NUREG/CR-4551 (Reference 7) for Peach Bottom. Specifically, Duane Arnold population dose estimates are obtained by scaling the NUREG/CR-4551 reference plant results by differences in population, reactor power level (assumed proportional to fission product inventory), and nominal containment maximum leakage rate (La). Using this reference plant is judged reasonable as it was used for the BWR example plant in Reference 20. Results of sensitivity studies are included which utilize release class doses modified to account for differences in ILRT methodology and appropriately adjusted for power level and population growth.

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology (Reference 2) and are summarized in Section 4.2 of the EPRI methodology.

- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.

- The representative containment leakage for Class 3a sequences is 10La based on the previously approved methodology performed for Indian Point Unit 3 (Reference 8 and Reference 9).

- The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2-A (Reference 20) and the NRC SE for NEI-94-01 Revision 2-A (Reference 25).

- The Class 3b is very conservatively categorized as LERF based on the previously approved methodology (References 8 and 9).

- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

- Where possible, the analysis should include a quantitative assessment of the

contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals. As DAEC has developed fire and seismic PRAs, these models were used in the assessment of external event risks. Where existing PRAs were not available or if the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document, the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

5 Inputs

This section summarizes the general resources available as input (Section 5.1) and the plant specific resources required (Section 5.2).

5.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 (Reference 10)
2. NUREG/CR-4220 (Reference 11)
3. NUREG-1273 (Reference 12)
4. NUREG/CR-4330 (Reference 13)
5. EPRI TR-105189 (Reference 14)
6. NUREG-1493 (Reference 6)
7. EPRI TR-104285 (Reference 2)
8. NUREG-1150 (Reference 15) and NUREG/CR-4551 (Reference 7)
9. NEI Interim Guidance for One-Time Extension of ILRT (Reference 3 and Reference 17)
10. Calvert Cliffs Liner Corrosion Analysis (Reference 5)
11. EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 20)

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50 mile radius surrounding a plant that is used as the bases for

the consequence analysis of the ILRT interval extension for DAEC. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent fifteen year extension of the ILRT interval.

5.1.1 NUREG/CR-3539 (Reference 10)

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 (Reference 16) as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

5.1.2 NUREG/CR-4220 (Reference 11)

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

5.1.3 NUREG-1273 (Reference 12)

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

5.1.4 NUREG/CR-4330 (Reference 13)

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

5.1.5 EPRI TR-105189 (Reference 14)

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a

BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals.

5.1.6 NUREG-1493 (Reference 6)

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

> Reduction in ILRT frequency from three per ten years to one per twenty years results in an "imperceptible" increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

5.1.7 EPRI TR-104285 (Reference 2)

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "... the proposed CLRT (containment leak rate tests) frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year..."

5.1.8 NUREG-1150 (Reference 15) and NUREG/CR 4551 (Reference 7)

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50 mile radial area surrounding Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the DAEC Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent Duane Arnold. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

5.1.9 NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals (Reference 17)

The guidance provided in this document builds on the EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

5.1.10 Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension (Reference 5)

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner. Licensees may consider approved LARs for one-time extensions involving containment types similar to their facility.

5.1.11 EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (Reference 20)

This report provides a generally applicable assessment of the risk involved in the extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the DAEC assessments to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI

Classes 3a and 3b scenarios in this analysis as described in Section 6. The Duane Arnold fifteen (15) year ILRT extension used an early version of this methodology.

5.2 Plant Specific Inputs

The plant specific information (Reference 27) used to perform the DAEC ILRT Extension Risk Assessments includes the following:

- Level 1 Model results
- Level 2 Model results
- Release category definitions used in the Level 2 Model
- Population within a 50 mile radius for the year 2040 based on the extrapolation of the actual population in 2000 (Reference 27). This represents the most recent projected growth for the area and by assuming positive population growth past the 2040 end of license this number conservatively bounds population estimates.
- Containment Fragility Curves

5.2.1 Level 1 Model

The Level 1 PRA models that are used for DAEC are characteristic of the as-built plants. The current Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = 4.24E-06/yr using a truncation value of 1.00E-12 (Table 1 of Reference 27). The model accounts for an increased CDF due to internal flood.

5.2.2 Level 2 Model

The Level 2 Models that are used for DAEC were developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 5-1 summarizes the pertinent DAEC results in terms of release category (from Reference 27). Note that the enumerated total internal events Level 2 release frequency is slightly larger than that of the internal events CDF. This difference arises as a result of the numerical truncation issues resulting from the full integration of core damage end-states into the Level 2 model and the impact of the CAFTA small number approximation as applied to the detailed containment failure model. The small number approximation is a standard modeling practice. While this difference is observable, it does not significantly impact the results of the simplified Level 2 PRA or the associated conclusions drawn with regard to the ILRT extension.

Table 5-1: DAEC Level 2 PSA Model Release Categories and Frequencies		
Release Category	Definition	Release Category Frequency/yr(1)
Intact	Containment Intact (Tech Spec Leakage only)	2.40E-07
DAEC-L2-H-E-RELEASE	High Early Releases	1.46E-06
DAEC-L2-H-I-RELEASE	High Intermediate Releases	5.02E-07
DAEC-L2-H-L-RELEASE	High Late Releases	1.76E-07
DAEC-L2-M-E-RELEASE	Medium Early Releases	1.27E-06
DAEC-L2-M-I-RELEASE	Medium Intermediate Releases	3.52E-08
DAEC-L2-M-L-RELEASE	Medium Late Releases	2.25E-07
DAEC-L2-L-E-RELEASE	Low Early Releases	5.69E-08
DAEC-L2-L-I-RELEASE	Low Intermediate Releases	5.74E-08
DAEC-L2-L-L-RELEASE	Low Late Releases	2.08E-08
DAEC-L2-LL-E-RELEASE	Low-Low Early Releases	2.37E-09
DAEC-L2-LL-I-RELEASE	Low-Low Intermediate Releases	2.90E-09
DAEC-L2-LL-L-RELEASE	Low-Low Late Releases	1.95E-07
CNTMT BYPASS	Associated with ISLOCA CD Scenarios	1.89E-08
CNTMT ISO FAILURES	Associated with CD scenarios with containment isolation failures leading to LERF	1.26E-08
	Total Release Category Frequency	4.00E-06

Note:

1. These values were quantified using a truncation value of 1.00E-12.

5.2.3 Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results to reflect the demographics around DAEC. Each of the release categories from Table 5-1 was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 5-2**Error! Reference source not found.** for reference purposes. Table 5-3 summarizes the calculated population dose for Peach Bottom associated with each APB from NUREG/CR-4551.

Table 5-2: Summary Accident Progression Bin (APB) Descriptions (Reference 7)	
Summary APB Number	Description
1	CD, VB, Early CF WW Failure, RPV Pressure > 200 psi at VB Core damage occurs, followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach), and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating [DCH] is possible).
2	CB, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core damage occurs, followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach), and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs, followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach), and the RPV pressure is greater than 200 psi at the time of the vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core damage occurs, followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach), and the RPV pressure is less than 200 psi at the time of the vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core damage occurs, followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction [MCCI]), and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core damage occurs, followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI), and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
7	CD, VB, No CF, Vent, N/A Core damage occurs, followed by vessel breach. The containment never structurally fails but is vented some time during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.

Table 5-2: Summary Accident Progression Bin (APB) Descriptions (Reference 7)	
Summary APB Number	Description
8	CD, VB, No CF, N/A, N/A Core damage occurs, followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

For the baseline analysis dose estimates are based on extrapolation of the results of the Peach Bottom assessment presented below, based on the values in Table 5-18 of Reference 20.

Table 5-3: Calculation of Peach Bottom Population Dose Risk at 50 Miles				
Collapsed Bin #	Collapsed Bin Frequencies (per year) (1)	Fractional APB Contributions to Risk (MFCR) (2)	Population Dose Risk (50 miles) (person-rem/yr) (3)	Population Dose (50 miles) (person-rem) (4)
1	9.55E-08	0.021	0.166	1.74E+06
2	4.77E-08	0.0066	0.0521	1.09E+06
3	1.48E-06	0.556	4.39	2.97E+06
4	7.94E-07	0.226	1.79	2.25E+06
5	1.30E-08	0.0022	0.0174	1.34E+06
6	2.04E-07	0.059	0.466	2.28E+06
7	4.77E-07	0.118	0.932	1.95E+06
8	7.99E-07	0.0005	3.95E-03	4.94E+03
9	3.85E-07	0.01	0.079	2.05E+05
10	4.34E-08	0	0	0
Totals	4.34E-06	1	7.9	N/A

Notes for Table 5-3:

(1) The total CDF of 4.34E-06 per year and the CDF subtotals by APB are taken from Figure 2.5-6 of NUREG/CR-4551, Volume 4, Revision 1, Part I.
(2) The individual APB contributions to the total 50-mile radius dose rate are taken from Table 5.2-3 of NUREG/CR-4551, Volume 4, Revision 1, Part I.
(3) The APB 50-mile dose rate is calculated by multiplying the individual APB dose rate fractional contributions (column 3) by the total 50-mile radius dose rate of 7.9 person-rem per year (taken from Table 5.1-1 of NUREG/CR-4551, Volume 4, Revision 1, Part I).
(4) The individual doses are calculated by dividing the individual APB dose risk (column 4) by the APB frequencies (column 2).

5.2.4 Population Dose Estimate Methodology

In accordance with Reference 1, the person-rem results in Table 5-3 can be used as an approximation of the dose for DAEC if it is corrected for allowable containment leak rate (La), reactor power level and the population density surrounding Duane Arnold.

La adjustment:

$$\text{FLeakage} = \frac{\text{La of Duane Arnold Energy Center (wt\%/day)}}{\text{La of reference plant (applicable only to those APBs affected by normal leakage)}}$$

La for DAEC is 2.0 wt%/day (Table 2 of Reference 27). La for Peach Bottom is 0.5 wt%/day (Reference 20).

FLeakage = 2.0 / 0.5

FLeakage = 4

Power level adjustment:

$$\text{FPower} = \frac{\text{Rated power level of Duane Arnold Energy Center (MWt)}}{\text{Rated power level of reference plant (MWt)}}$$

The rated power level for DAEC is 1912 MWt (Table 2 of Reference 27). The rated power level for Peach Bottom is 3293 MWt (Reference 2).

FPower = 1912 MWt / 3293 MWt

FPower = 0.581

Population density adjustment:

The total population within a 50 mile radius of DAEC is 9.626E+5 persons (Reference 27).

This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for DAEC. Note that the numbers reported below may represent a rounded result as displayed in the attached spreadsheets.

Total 2040 estimated DAEC Population within 50 miles = 9.626E+5 persons.

Peach Bottom Population within a 50 mile radius from the NUREG/CR-4551 reference plant = 3.2E+06 persons (Reference 2).

FPopulation = 9.626E+5 persons / 3.2E+06 persons = 0.301

The factors developed above are used to adjust the population dose for the surrogate plant (Peach Bottom) for DAEC. For intact containment endstates, the total population dose factor is as follows:

FIntact = FPopulation * FPower * FLeakage

FIntact = 0.301 * 0.581* 4

FIntact = 0.699

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

FOthers = FPopulation * FPower

FOthers = 0.301 * 0.581

FOthers = 0.175

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. The above adjustments provide an approximation for DAEC of the population doses associated with each of the release categories from NUREG/CR-4551.

Table 5-4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for DAEC.

Table 5-4: Calculation of DAEC Population Dose Risk at 50 Miles				
Accident Progression Bin (APB)	NUREG/CR-4551 Peach Bottom Population Dose at 50 miles (person-rem)*	Bin Multiplier used to obtain Duane Arnold Population Dose**		Duane Arnold Adjusted Population Dose at 50 miles (person-rem)
1	1.74E+06	FOthers	0.175	3.04E+05
2	1.09E+06	FOthers	0.175	1.90E+05
3	2.97E+06	FOthers	0.175	5.19E+05
4	2.25E+06	FOthers	0.175	3.93E+05
5	1.34E+06	FOthers	0.175	2.34E+05
6	2.28E+06	FOthers	0.175	3.98E+05
7	1.95E+06	FOthers	0.175	3.41E+05
8	4.94E+03	FIntact	0.699	3.45E+03
9	2.05E+05	FOthers	0.175	3.58E+04
10	0.00E+00	FOthers	0.175	0.00E+00

Notes:

* These values follow the template in EPRI Report 1009325, Section 5.2.2, Table 5-18.

** These values follow the guidance in EPRI Report 1009325 (See page 5-25).

5.2.5 Application of DAEC PRA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the DAEC PRA Level 2 models are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to match the Duane Arnold PRA Level 2 release categories to the collapsed APBs. The APB definitions are shown in Table 5-2. The Duane Arnold Level 2 release categories and frequencies are from Reference 27. The assignments are shown in Table 5-5, along with the corresponding EPRI classes (see below). The EPRI classes and descriptions are listed in Table 5-6.

Table 5-5: DAEC Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI Accident Classes

EPRI Class	EPRI Description	Leakage Basis	Peach Bottom APB for Dose (NUREG/CR-4551)	DAEC Level 2 Release Category Frequency (or other)	DAEC Level 2 Release Category Frequency (or other) basis
1	No containment failure	La	8	2.08E-07	Containment Intact Frequency (Intact-3a-3b)
2	Large Containment Isolation Failure	Plant value	3 (highest dose)	1.26E-08	Containment Isolation Failure (H-E with IS=F)
3a	Small pre-existing failure	10*La	10*dose of APB 8	2.56E-08	By Methodology = (CDF-(H-E))*0.0092
3b	Large pre-existing failure	100*La	100*dose of APB 8	6.39E-09	By Methodology = (CDF-(H-E))*0.0023
4	Small isolation failure - Type B	N/A	N/A	N/A	N/A
5	Small isolation failure - Type C	N/A	N/A	N/A	N/A
6	Small isolation failure - Dependent failure	N/A	N/A	N/A	N/A
7	Severe Accident Sequences	Plant value (weighted average)	See below	3.97E-06	Sum of the Class 7 subcategories
7a	Subcategory (not EPRI)		3 (highest dose)	1.43E-06	H-E without IS=F and V
7b	Subcategory (not EPRI)		6 (high does, late DW failure)	6.78E-07	H-I and H-L

Table 5-5: DAEC Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI Accident Classes					
EPRI Class	**EPRI Description**	**Leakage Basis**	**Peach Bottom APB for Dose (NUREG/CR-4551)**	**DAEC Level 2 Release Category Frequency (or other)**	**DAEC Level 2 Release Category Frequency (or other) basis**
7c	Subcategory (not EPRI)		1 (WW failure early)	1.27E-06	M-E
7d	Subcategory (not EPRI)		2 (WW failure late)	2.60E-07	M-I and M-L
7e	Subcategory (not EPRI)		9 (CD, no vessel breach)	3.35E-07	L & LL
8	Containment bypass	Plant value	3 (highest dose)	1.89E-08	H-E Class V Scenarios

5.2.6 Release Category Definitions

Table 5-6 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology (Reference 2). These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 6 of this report.

Table 5-6: EPRI Containment Failure Classification (Reference Table 4.2-6 of Reference 20)	
Class	**Description**
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there are a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

5.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellow arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 5-6 is divided into two sub-classes, Class 3a and Class 3b, which represent small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance (Reference 20). For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to 2/217=0.0092). For Class 3b, Jefferys non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5 / (217+1) = 0.0023).

In a follow-on letter (Reference 17) to their ILRT guidance document (Reference 3), NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> *The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.*

The application of this additional guidance to the analysis for DAEC, as detailed in Section 6 involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the DAEC Level 2 PRA analyses.

- Class 1 accident sequences may involve availability and or successful operation of containment sprays. It could be assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. However, in this assessment DAEC does not credit containment spray as a means of reducing releases from Class 3 events.

Consistent with the NEI Guidance (Reference 3), the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three year test interval is 1.5 years (3 yr/2), and the average time that a leak could exist without detection for a ten year interval is five years (10 yr/2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. An extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak compared to a three year interval.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension (Reference 9)) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

5.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis (Reference 5). The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner. DAEC employs a Mark I pressure suppression containment system which houses the reactor vessel, the reactor recirculation loops, and other branch connections of the Reactor Coolant System, a pressure suppression chamber that stores a large volume of water, a vent system connecting the drywell and the pressure suppression chamber, isolation valves, containment cooling systems, and other service equipment. The drywell is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion. The drywell is surrounded by a reinforced-concrete structure for shielding purposes. The concrete provides no drywell structural support. The pressure suppression chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell, (Section 2.1 of Reference 29).

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of a containment steel liner. It should be noted that this computation is being applied to provide an upper bound approach to quantify corrosion induced risk. Furthermore, the likelihood of detection of significant corrosion for the 80% of the containment accessible for visual inspection is very high. Regardless, the Calvert Cliffs corrosion likelihood methodology is then used to determine the resulting change in risk.

Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the upper containment (cylinder and dome regions in Calvert Cliffs evaluation)
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

5.4.1 Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is conservatively assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 5-7, Step 1).

- There are two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis. These events have been determined to be applicable at Duane Arnold. The events included in the Calvert Cliffs corrosion assessment process, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the nonvisible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is based on 70 steel-lined containments.

- The Calvert Cliffs analysis used the estimated historical liner flaw probability based on 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date. Since the time of the Calvert Cliffs submittal, three additional relevant liner corrosion events involving concealed corrosion (corrosion initiated on the inaccessible liner surface) were observed and are considered in the corrosion risk assessment. Two of these events occurred at Beaver Valley Unit 1 (References 21 and 22). The third occurred at D.C. Cook Unit 2 (Reference 23). Consistent with the addition of the three observed events, the historical liner flaw probability was established by incrementing the flaw observation time by 12.25 years (September 1996 to July 2014). This re-evaluation resulted in a reduction of the historical liner flaw likelihood to 4.02E-03/year ((2+3) / [70 * (5.5 + 12.25)] = 4.02E-03/year). This value is smaller than the value of 5.2E-03 which is used in the Calvert Cliffs analysis. The conservative value of 5.2E-03 will be used in this Duane Arnold report to remain consistent with the Calvert Cliffs analysis. This approach, while conservative, provides a simplified, direct comparison to the previously evaluated Calvert Cliffs analysis.

- Consistent with the Calvert Cliffs analysis, the steel plate flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to

address the increased likelihood of corrosion as the steel ages. (See Table 5-7, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.

- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the Calvert Cliffs ILRT target pressure of 37 psig. For DAEC, the containment failure probabilities are less than this at 37 psig because the DAEC design pressure is 56 psig (Reference 27). Given the above information and consistent with recently approved 15 year ILRT extensions (Reference 28) probabilities of 1% for the shell above the basemat and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 5-7, Step 4).

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the upper containment region (See Table 5-7, Step 4).

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection (See Table 5-7, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

5.4.2 Analysis

Table 5-7: Steel Liner Corrosion Base Case					
Step	**Description**	**Upper Containment**		**Containment Basemat**	
1	Historical Steel Liner Flaw Likelihood Failure Data: Containment location specific	Events: 2 (Brunswick 2 & North Anna 2) (2)/(70 * 5.5) = **5.2E-03**		Events: 0 (assume half a failure) 0.5/(70 * 5.5) = **1.3E-03**	
2	Age Adjusted Steel Liner Flaw Likelihood During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	Year 1 avg 5-10 15	Failure Rate 2.1E-03 5.2E-03 1.4E-02	Year 1 avg 5-10 15	Failure Rate 5.0E-04 1.3E-03 3.5E-03
		15 year average = 6.27E-03		**15 year average = 1.57E-03**	
3	Flaw Likelihood at 3, 10, and 15 years Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference 5).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)		**0.18% (1 to 3 years)** **1.02% (1 to 10 years)** **2.35% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)	

Table 5-7: Steel Liner Corrosion Base Case			
Step	**Description**	**Upper Containment**	**Containment Basemat**
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	Visual Inspection Detection Failure Likelihood Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	Likelihood of Non-Detected Containment Leakage (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the leakages for the upper containment and the containment basemat as summarized below for DAEC.

Total Likelihood of Non-Detected Containment Leakage Due To Corrosion for DAEC:

At 3 years: 0.00071% + 0.00018% = 0.00089%

At 10 years: 0.0041% + 0.0010% = 0.0051%

At 15 years: 0.0094% + 0.0024% = 0.012%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the three in ten year case is calculated as follows:

- Per Table 5-5, the EPRI Class 3b frequency is 6.39E-09/yr. As discussed in Section 6.1, this is the DAEC CDF associated with accidents that are not independently LERF [CDF-(DAEC-L2-H-E-RELEASE)] = 2.78E-06/yr times the conditional probability of class 3b (0.0023).

- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 2.78E-06/yr * 0.00089% = 2.47E-11/yr where 0.00089% was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at three years.

- The three in ten year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 6.39E-09/yr + 2.47E-11/yr = 6.42E-09/yr.

6 Results

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 20), EPRI-TR-104285 (Reference 2) and previous risk assessment submittals on this subject (References 5, 8, 18 and 19) have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 6-1 lists these accident classes.

The analysis performed examined DAEC specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).

- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 6-1: Accident Classes

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
4	Small Isolation Failures (Failure to Seal-Type B)

Table 6-1: Accident Classes	
Accident Classes (Containment Release Type)	Description
5	Small Isolation Failures (Failure to Seal-Type C)
6	Other Isolation Failures (e.g., Dependent Failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including Very Low and No Release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 6-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from three to fifteen and ten to fifteen years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP).

6.1 Step 1 - Quantify the Base-Line Risk in Terms of Frequency Per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 6-1 were developed for DAEC by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 5-5, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments

were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion per the methodology described in Section 5.4.

For DAEC, the total frequency of the categorized sequences is 4.24E-06/yr, the same as total CDF. Table 6-2 and Table 6-3 summarize the results.

Table 6-2: DAEC Categorized Accident Classes and Frequencies		
EPRI Class	**Duane Arnold Release Category**	**Frequency (per yr)**
1	Intact containment (INTACT)	2.08E-07
2	Containment Isolation failures	1.26E-08
7	Containment Failure due to Severe Accident Phenomena	3.97E-06
8	Containment Bypass (Associated with ISLOCA CD Scenarios)	1.89E-08

Class 1 Sequences: This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Containment Intact Level 2 Release Category listed in Table 5-6 minus the EPRI Class 3a and 3b frequency, which are calculated below.

Class 2 Sequences: This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Large Containment Isolation Failures Level 2 Release Category listed in Table 5-5.

Class 3 Sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 [see Section 5.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 [see Section 5.3]

As described in Section 5.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

Class 3a Frequency= 0.0092 * (CDF - DAEC-L2-H-E-RELEASE)
= 0.0092 * (4.24E-06 – 1.46E-06) = 2.56E-08/yr

Class 3b Frequency = 0.0023 * (CDF - DAEC-L2-H-E-RELEASE)
=0.0023 * (4.24E-06 – 1.46E-06) = 6.39-09/yr

For this analysis, the associated containment leakage for Class 3a is 10La and for Class 3b is 100La. These assignments are consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A (Reference 20).

Note, in the above equations for the Class 3a and 3b release frequencies, the total adjusted release frequency from the appropriate columns of Table 5-5 has been substituted for CDF. As discussed previously this process marginally over-estimates the Class 3 releases.

Class 4 Sequences: This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences: This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type C test components occurs. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, and their frequency is very low compared with the other classes, this group is not evaluated any further in this analysis. The frequency for Class 5 sequences is subsumed into Class 7, where it contributes insignificantly. Therefore, changes in the frequency of Type C tests do not affect the conclusions of this analysis.

Class 6 Sequences: This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution, typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the Class 2 assumptions. Consistent with guidance provided in EPRI Report No. 1009325, Revision 2-A, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences: This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from the Severe Accident Phenomena-Induced Failures Release Category from the DAEC Level 2 results shown in Table 5-5.

Class 8 Sequences: This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from the Containment Bypass Release Category from the DAEC Level 2 results shown in Table 5-1

6.1.1 Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 the NEI Interim Guidance, and guidance provided in EPRI Report No. 1009325, Revision 2-A. Table 6-3 summarizes these accident frequencies by accident class for DAEC.

Table 6-3: Radionuclide Release Frequencies as a Function of Accident Class (Base Case)			
Accident Classes (Containment Release Type)	**Description**	**Base Case Frequency (/yr)**	**Base Case Frequency Plus Corrosion(1) (/yr)**
1	No Containment Failure	2.08E-07	2.08E-07
2	Large Isolation Failures (Failure to Close)	1.26E-08	1.26E-08
3a	Small Isolation Failures (liner breach)	2.56E-08	2.56E-08
3b	Large Isolation Failures (liner breach)	6.39E-09	6.42E-09
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.97E-06	3.97E-06
8	Bypass (Interfacing System LOCA)	1.89E-08	1.89E-08
CDF	All CET end states	4.24E-06	4.24E-06

Notes:
(1). Note that this is based on data developed in Section 5.4. Only Class 3b is impacted by the corrosion.

6.2 Step 2 - Develop Plant Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant specific release analyses were performed to estimate the person-rem doses to the population within a 50 mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 5.2, and summarized in Table 5-4. The results of applying these releases to the EPRI containment failure classification are as follows:

Class 1 = 3.45E+03 person-rem (1)

Class 2 = 5.19E+05 person-rem (2)

Class 3a = 3.45E+03 person-rem x 10La = 3.45E+04 person-rem (3)

Class 3b = 3.45E+03 person-rem x 100La = 3.45E+05 person-rem (3)

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = 3.67E+05 person-rem [4]

Class 8 = 5.19E+05 person-rem [5]

Notes:

(1) The derivation is described in Section 5.2 for DAEC. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #8).

(2) The Class 2, containment isolation failures, dose is assigned from APB #3 (Early CF).

(3) The Class 3a and 3b dose are related to the Class 1 leakage rate as shown. While no pre-existing leakage in excess of 21 La has been identified for any historical ILRT event, Class 3b releases are conservatively assessed at 100La. Class 3a releases are conservatively assessed at 10La. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

(4) The Class 7 population dose and frequency are developed as follows:

Class 7 Frequency Weighted Population Dose				
DAEC Release Frequency (/yr)		**NUREG/CR 4551 APB**	**DAEC Population Dose (person-rem)**	**DAEC Dose Rate (person-rem/yr)**
Class 7a	1.43E-06	3	5.19E+05	7.41E-01
Class 7b	6.78E-07	6	3.98E+05	2.70E-01
Class 7c	1.27E-06	1	3.04E+05	3.86E-01
Class 7d	2.60E-07	2	1.90E+05	4.95E-02
Class 7e	3.35E-07	9	3.58E+04	1.20E-02
Total	**3.97E-06**	**N/A**	**N/A**	**1.46E+00**
DAEC Frequency weighted population dose (person-rem)			**3.67E+05**	

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass).

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology (Reference 2) containment failure classifications, and consistent with the NEI guidance (Reference 1) as modified by EPRI Report No. 1009325, Revision 2-A are provided in Table 6-4.

Table 6-4: DAEC Population Dose Estimates for Population Within 50 Miles		
Accident Classes (Containment Release Type)	**Description**	**Person-Rem (50 miles)**
1	No Containment Failure	3.45E+03
2	Large Isolation Failures (Failure to Close)	5.19E+05
3a	Small Isolation Failures (liner breach)	3.45E+04
3b	Large Isolation Failures (liner breach)	3.45E+05
4	Small Isolation Failures (Failure to seal-Type B)	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	3.67E+05
8	Bypass (Interfacing System LOCA)	5.19E+05

The above dose estimates, when combined with the results presented in Table 6-3, yield the DAEC baseline mean consequence measures for each accident class. These results are presented in Table 6-5.

Table 6-5: Duane Arnold Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Ctmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	3.45E+03	2.08E-07	7.18E-04	2.08E-07	7.18E-04	-8.54E-08
2	Large Isolation Failures (Failure to Close)	5.19E+05	1.26E-08	6.54E-03	1.26E-08	6.54E-03	0.00E+00
3a	Small Isolation Failures (liner breach)	3.45E+04	2.56E-08	8.83E-04	2.56E-08	8.83E-04	0.00E+00
3b	Large Isolation Failures (liner breach)	3.45E+05	6.39E-09	2.21E-03	6.42E-09	2.22E-03	8.54E-06
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.67E+05	3.97E-06	1.46E+00	3.97E-06	1.46E+00	0.00E+00

Table 6-5: Duane Arnold Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Ctmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
8	Bypass (Interfacing System LOCA)	5.19E+05	1.89E-08	9.80E-03	1.89E-08	9.80E-03	0.00E+00
CDF	All CET end states	N/A	4.24E-06	1.48E+00	4.24E-06	1.48E+00	8.45E-06

Notes:
(1) Only release Classes 1 and 3b are affected by the corrosion analysis.
(2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

6.3 Step 3 - Evaluate Risk Impact of Extending Type A Test Interval From 10 to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current ten year value to fifteen years. To do this, an evaluation must first be made of the risk associated with the ten year interval since the base case applies to a three year interval (i.e., a simplified representation of a three in ten interval).

6.3.1 Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is directly impacted. As it is assumed that the new Class 3 endstates arise from previously intact containment states, the intact state frequency is reduced accordingly. The risk contribution is changed based on the NEI guidance as described in Section 5.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a ten year interval are presented in Table 6-6.

6.3.2 Risk Impact Due to 15-Year Test Interval

The risk contribution for a fifteen year interval is calculated in a manner similar to the ten year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the three year interval value, as described in Section 5.3. The results for this calculation are presented in Table 6-7.

Table 6-6: Duane Arnold Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	3.45E+03	1.34E-07	4.61E-04	1.33E-07	4.61E-04	-2.84E-07
2	Large Isolation Failures (Failure to Close)	5.19E+05	1.26E-08	6.54E-03	1.26E-08	6.54E-03	0.00E+00
3a	Small Isolation Failures (liner breach)	3.45E+04	8.52E-08	2.94E-03	8.52E-08	2.94E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	3.45E+05	2.13E-08	7.35E-03	2.14E-08	7.38E-03	2.84E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-6: Duane Arnold Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	3.67E+05	3.97E-06	1.46E+00	3.97E-06	1.46E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	5.19E+05	1.89E-08	9.80E-03	1.89E-08	9.80E-03	0.00E+00
CDF	All CET end states	N/A	4.24E-06	1.49E+00	4.24E-06	1.49E+00	2.82E-05

Notes:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. In these cases, the corrosion analysis causes more cases that were previously Class 1 (containment intact) to become Class 3b (large isolation failures (liner breach). This lowers the frequency of Class 1 and raises the frequency of Class 3b.

(2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 6-7: Duane Arnold Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
1	No Containment Failure (2)	3.45E+03	8.02E-08	2.77E-04	8.00E-08	2.76E-04	-4.27E-07
2	Large Isolation Failures (Failure to Close)	5.19E+05	1.26E-08	6.54E-03	1.26E-08	6.54E-03	0.00E+00
3a	Small Isolation Failures (liner breach)	3.45E+04	1.28E-07	4.41E-03	1.28E-07	4.41E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	3.45E+05	3.20E-08	1.10E-02	3.21E-08	1.11E-02	4.27E-05
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-7: Duane Arnold Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	3.67E+05	3.97E-06	1.46E+00	3.97E-06	1.46E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	5.19E+05	1.89E-08	9.80E-03	1.89E-08	9.80E-03	0.00E+00
CDF	All CET end states	N/A	4.24E-06	1.49E+00	4.24E-06	1.49E+00	4.23E-05

Notes:
(1) Only release Classes 1 and 3b are affected by the corrosion analysis. . In these cases, the corrosion analysis causes more cases that were previously Class 1 (containment intact) to become Class 3b (large isolation failures (liner breach). This lowers the frequency of Class 1 and raises the frequency of Class 3b.
(2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

6.4 Step 4 - Determine the Change in Risk in Terms of LERF

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. The DAEC PRA does not credit containment overpressure, so loss of overpressure due to an undetected leak does not affect CDF. Because the ILRT does not impact CDF, the relevant metric is LERF.

For DAEC, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a ten year test interval from Table 6-6, the Class 3b frequency (conservatively including corrosion) is 2.14E-08/yr; and, based on a fifteen year test interval from Table 6-7, it is 3.21E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from three in ten years to one in fifteen years for DAEC is 2.57E-08/yr as shown in Table 6-8. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.07E-08/yr.

As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF for DAEC is below the threshold criteria for a very small change when comparing both the fifteen year results to the current ten year requirement, and the fifteen year results compared to the original three year requirement. See Table 6-8 for more information.

6.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability (CCFP)

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI Report No. 1009325, Revision 2-A. The NRC has previously accepted similar calculations (Reference 9) as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The list below shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects. Note that the numbers used are rounded to the second decimal place.

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

$CCFP_3 = [1 - (2.08E\text{-}07/yr + 2.56E\text{-}08/yr) / 4.24E\text{-}06/yr] * 100\% = 94.50\%$

$CCFP_3 = 94.50\%$

$CCFP_{10} = [1 - (1.33E\text{-}07/yr + 8.52E\text{-}08/yr) / 4.24E\text{-}06/yr] * 100\% = 94.85\%$

$CCFP_{10} = 94.85\%$

$CCFP_{15} = [1 - (8.00E\text{-}08/yr + 1.28E\text{-}07/yr) / 4.24E\text{-}06/yr] * 100\% = 95.10\%$

$CCFP_{15} = 95.10\%$

$\Delta CCFP_{15\text{-}3} = CCFP_{15} - CCFP_3 = 0.61\%$

$\Delta CCFP_{15\text{-}10} = CCFP_{15} - CCFP_{10} = 0.25\%$

$\Delta CCFP_{10\text{-}3} = CCFP_{10} - CCFP_3 = 0.35\%$

The change in CCFP of approximately 0.61% by extending the test interval to fifteen years from the original three in ten year requirement is judged to be very small (i.e., less than 1.5% per the EPRI submittal guidance).

6.6 Summary of Results

The results from this ILRT extension risk assessment for DAEC are summarized in Table 6-8.

<table>
<tr><th colspan="8">Table 6-8: Duane Arnold ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)</th></tr>
<tr><th rowspan="2">EPRI Class</th><th rowspan="2">DOSE Per-Rem</th><th colspan="2">Base Case 3 in 10 Years</th><th colspan="2">Extend to 1 in 10 Years</th><th colspan="2">Extend to 1 in 15 Years</th></tr>
<tr><th>CDF/Yr</th><th>Per-Rem/Yr</th><th>CDF/Yr</th><th>Per-Rem/Yr</th><th>CDF/Yr</th><th>Per-Rem/Yr</th></tr>
<tr><td>1</td><td>3.45E+03</td><td>2.08E-07</td><td>7.18E-04</td><td>1.33E-07</td><td>4.61E-04</td><td>8.00E-08</td><td>2.76E-04</td></tr>
<tr><td>2</td><td>5.19E+05</td><td>1.26E-08</td><td>6.54E-03</td><td>1.26E-08</td><td>6.54E-03</td><td>1.26E-08</td><td>6.54E-03</td></tr>
<tr><td>3a</td><td>3.45E+04</td><td>2.56E-08</td><td>8.83E-04</td><td>8.52E-08</td><td>2.94E-03</td><td>1.28E-07</td><td>4.41E-03</td></tr>
<tr><td>3b</td><td>3.45E+05</td><td>6.42E-09</td><td>2.22E-03</td><td>2.14E-08</td><td>7.38E-03</td><td>3.21E-08</td><td>1.11E-02</td></tr>
<tr><td>7</td><td>3.67E+05</td><td>3.97E-06</td><td>1.46E+00</td><td>3.97E-06</td><td>1.46E+00</td><td>3.97E-06</td><td>1.46E+00</td></tr>
<tr><td>8</td><td>5.19E+05</td><td>1.89E-08</td><td>9.80E-03</td><td>1.89E-08</td><td>9.80E-03</td><td>1.89E-08</td><td>9.80E-03</td></tr>
<tr><td>Total</td><td>N/A</td><td>4.24E-06</td><td>1.48E+00</td><td>4.24E-06</td><td>1.49E+00</td><td>4.24E-06</td><td>1.49E+00</td></tr>
<tr><td colspan="2">ILRT Dose Rate from 3a and 3b Per-Rem/Yr</td><td colspan="2">3.10E-03</td><td colspan="2">1.03E-02</td><td colspan="2">1.55E-02</td></tr>
<tr><td rowspan="2">Delta Total Dose Rate (1)</td><td>From 3 yr</td><td colspan="2">N/A</td><td colspan="2">6.96E-03</td><td colspan="2">1.20E-02</td></tr>
<tr><td>From 10 yr</td><td colspan="2">N/A</td><td colspan="2">N/A</td><td colspan="2">4.99E-03</td></tr>
<tr><td rowspan="2">% change in dose rate from base</td><td>From 3 yr</td><td colspan="2">N/A</td><td colspan="2">0.47%</td><td colspan="2">0.81%</td></tr>
<tr><td>From 10 yr</td><td colspan="2">N/A</td><td colspan="2">N/A</td><td colspan="2">0.34%</td></tr>
<tr><td colspan="2">3b Frequency (LERF) Per-Rem/Yr</td><td colspan="2">6.42E-09</td><td colspan="2">2.14E-08</td><td colspan="2">3.21E-08</td></tr>
<tr><td rowspan="2">Delta LERF</td><td>From 3 yr</td><td colspan="2">N/A</td><td colspan="2">1.50E-08</td><td colspan="2">2.57E-08</td></tr>
<tr><td>From 10 yr</td><td colspan="2">N/A</td><td colspan="2">N/A</td><td colspan="2">1.07E-08</td></tr>
<tr><td colspan="2">CCFP %</td><td colspan="2">94.50%</td><td colspan="2">94.85%</td><td colspan="2">95.10%</td></tr>
<tr><td rowspan="2">Delta CCFP %</td><td>From 3 yr</td><td colspan="2">N/A</td><td colspan="2">0.35%</td><td colspan="2">0.61%</td></tr>
<tr><td>From 10 yr</td><td colspan="2">N/A</td><td colspan="2">N/A</td><td colspan="2">0.25%</td></tr>
</table>

Notes:

(1) The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

7 Sensitivities

7.1 Sensitivity to Corrosion Impact Assumptions

The DAEC results in Table 6-5 through Table 6-8 show that including corrosion effects calculated using the assumptions described in Section 5.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the upper containment and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 7-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 2.47E-12/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 7-1: Steel Plate Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 Years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Upper Containment, 0.1% Basemat)	Base Case (10% Upper Containment, 100% Basemat)	2.57E-08	9.90E-11
Doubles every 2 yrs	Base	Base	2.58E-08	1.76E-10
Doubles every 10 yrs	Base	Base	2.56E-08	2.86E-11
Base	Base	15%	2.57E-08	1.38E-10
Base	Base	5%	2.56E-08	5.95E-11
Base	10% Upper Containment, 1% Basemat	Base	2.66E-08	9.90E-10
Base	0.1% Upper Containment, 0.01% Basemat	Base	2.56E-08	9.90E-12
Lower Bound				

Doubles every 10 yrs	0.1% Upper Containment, 0.01% Basemat	5% Upper Containment, 1% Basemat	2.56E-08	1.71E-15
Upper Bound				
Doubles every 2 yrs	10% Upper Containment, 1% Basemat	15% Upper Containment, 100% Basemat	2.56E-08	2.47E-12

7.2 Sensitivity to Class 3B Contribution to LERF

The Class 3b frequency for the base case of a three in ten year ILRT interval including corrosion is 6.42E-09/yr (Table 6-5). Extending the interval to one in ten years results in a frequency of 2.14E-08/yr (Table 6-6). Extending it to one in fifteen years results in a frequency of 3.21E-08/yr (Table 6-7), which is an increase of 2.57E-08/yr from three in ten years to once in fifteen years.

If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF for DAEC due to extending the interval from three in ten to one in fifteen is below the RG 1.174 threshold for very small changes in LERF of 1.00E-07/yr.

7.3 Potential Impact from External Events

The DAEC has systematically considered risk posed by the following external hazards: seismicity, fire (historically considered an external hazard although the analysis focuses on fires originating within the plant), external flood, high winds and tornadoes, transportation and nearby facility hazards, and other plant-unique hazards. These analyses are summarized in the DAEC Individual Plant Examination for External Events (IPEEE) (Reference 24) submittal and supporting documentation. Additionally, DAEC has performed a comprehensive fire PRA update in accordance with more current methodological guidance in support of its transition to the risk-informed fire protection program, National Fire Protection Association (NFPA) 805 (Reference 31).

Seismic Assessment

DAEC performed a Seismic Margins Assessment (SMA) in support of their IPEEE submittal (Reference 24) using the guidance of EPRI NP-6041 (Reference 32). While this analysis did not identify any credible seismic sequences exceeding the reportability criteria of the IPEEE program, a number of insights were identified that led to plant modifications to reduce seismic risk. For example, one masonry wall was identified as a potential outlier that could fall and damage seismic safe shutdown equipment, and this wall was subsequently qualified for Safe Shutdown Earthquake (SSE) loading. Elements of the control room ceiling were modified to improve integrity during an earthquake. In addition, unanchored gas storage bottles were identified near safe shutdown equipment, and modifications were made to properly secure the bottles.

While the SMA methodology used for the IPEEE does not estimate seismic CDF, in 2008 the DAEC assessment of Severe Accident Mitigation Alternatives (SAMA) developed a seismic CDF estimate of 6.99E 07/yr. This value had been relatively stable over five revisions to the PRA model and is used for the current ILRT assessment.

Fire Assessment

In the early 1990's DAEC performed a fire risk analysis in support of their IPEEE submittal (Reference 24) using the Electric Power Research Institute's Fire-Induced Vulnerability Evaluation (FIVE) methodology (Reference 33). This study identified the two essential 4kV switchgear rooms as the dominant contributors to station fire risk, and several controls were proposed to mitigate this risk, primarily involving configuration risk management of maintenance on the River Water System. Fire CDF for the remaining plant areas was assessed to be below the reportability criteria of the IPEEE program and not pose a significant safety concern.

More recently, DAEC completed a comprehensive fire PRA update in support of transition to the risk-informed fire protection program NFPA 805 (Reference 31). This update primarily applied the guidance of NUREG/CR-6850, was independently peer reviewed against ASME/ANS RA-Sa-2009, and was extensively reviewed and judged by the NRC to be of acceptable scope and quality for the NFPA 805 risk-informed application. Details of the NRC review can be found in the DAEC NFPA 805 SER dated September 10, 2013 (Reference 34) and its supporting references.

The most current DAEC fire PRA quantification notebook (Reference 35), inclusive of the NFPA 805 implementation items, reports a total fire CDF and LERF estimations of 1.20E-05/yr and 7.49E-06/yr, respectively. Note these values are lower than those identified in the NFPA 805 SER (Reference 34), and these reductions reflect refinements implemented during NFPA 805 implementation. Consistent with the IPEEE insights, fires in essential switchgear rooms, where fire can also fail offsite power, are the dominant contributors to plant fire risk.

External Flooding Assessment

The DAEC IPEEE (Reference 24) effort reviewed the plant design basis and confirmed that it meets the 1975 Standard Review Plant (SRP) criteria related to external flooding. Based on this conformance, DAEC judged the external flood contribution to core damage frequency to be negligible, and therefore external flooding risk is excluded numerically from this ILRT assessment.

High Winds Assessment

The DAEC IPEEE (Reference 24) included a conservative bounding evaluation of high winds and tornados, estimating a total wind-induced CDF of 1.41E-07/yr. The IPEEE submittal (Reference 24) further judged that a more realistic approach would show this CDF contribution to be approximately an order of magnitude lower. The analysis did not identify any significant vulnerabilities and concluded that high winds present an insignificant contribution to plant risk.

Transportation and Nearby Facility Hazards

The DAEC IPEEE (Reference 24) reviewed hazards involving marine, railroad, and truck accidents and concluded that they do not represent a significant contribution to plant risk, and this conclusion was based primarily on separation distance between the plant and these hazards.

The risk of aviation hazard associated with two nearby federal airways was assessed and determined to be less than the IPEEE reportability criteria.

Nearby facility hazards and hazardous material storage were reviewed and no accident could be postulated that would impact safe operation of the plant.

The storage of a propane tank near the emergency diesel generator rooms was identified and evaluated. Large barriers were placed around the tank to reduce its associated core damage frequency (Reference 24).

For this ILRT assessment, 1.00E-06 is conservatively used as the transportation and nearby facility hazard CDF. LERF impacts are judged to be negligible.

Other Plant-Unique Hazards

An exhaustive list of other potential plant-unique hazards was evaluated using a formal screening process. No other plant-unique events were identified as potentially significant contributors to plant risk.

External Events Total CDF, Total LERF, and ΔLERF for ILRT Application

Table 7-2 summarizes the conservative Duane Arnold external hazard CDF estimates used for the ILRT application.

Table 7-2: Conservative External Hazard CDF and LERF Estimates used for ILRT Application

Hazard	CDF (/yr)	LERF (/yr)
Seismicity	6.99E-07	<6.99E-07
Fire	1.20E-05	7.49E-06
External Flood	negligible	negligible
High Winds	1.41E-07	<1.41E-07
Transportation and Nearby Facilities	1.00E-06	negligible
Other Plant-Unique	insignificant	insignificant
Total	**1.38E-05**	**<8.33E-06**

An alternate estimate of the seismic and wind hazard LERF can be made assuming the ratio CDF to LERF for these hazards is similar to the ratio for internal events. The internal events CDF is 4.24E-06/yr (per Section 5.2.1), and the internal events LERF is 1.46E-06 (per high early release frequency specified in Table 5-1). The internal event ratio of CDF to LERF is:

$$CDF_{IE} / LERF_{IE} = 4.24E\text{-}06 / 1.46E\text{-}06 = 2.90$$

The total external events LERF is therefore approximated as:

$LERF_{EE\text{-}alternate} = (CDF_{seis} / 2.90) + (CDF_{wind} / 2.90) + (LERF_{fire})$
$= (6.99E\text{-}07 / 2.90) + (1.41E\text{-}07 / 2.90) + (7.49E\text{-}06)$
$= 7.78E\text{-}06/yr$

External events LERF attributed specifically to non-detected containment failures is conservatively estimated as follows, using the probabilities of a non-detected containment failure (p_{NDCF}) described in Section 5.3:

$LERF_{NDCF} = p_{NDCF,\lambda} * (CDF_{EE} - LERF_{EE})$

Where, $p_{NDCF,3/10} = 0.0023$
$p_{NDCF,1/10} = 0.0023 * 3.33$
$p_{NDCF,1/15} = 0.0023 * 5.00$
$CDF_{EE} = 1.38E\text{-}05/yr$
$LERF_{EE\text{-}alternate} = 7.78E\text{-}06/yr$
[Note CDF_{EE} and $LERF_{EE}$ are the core damage and large early release frequencies, respectively, associated with external hazard sequences only]

Note in the above equation, LERF is subtracted from CDF, such that the portion of CDF sequences that also progress to LERF (for example by containment isolation failures or severe accident processes that are independent of a degraded containment condition undiscovered due to ILRT extension) are excluded from the calculation. For conservatism, the lower estimated value of total LERF is used in this equation, which is used in Table 7-3 to calculate ΔLERF, to maximize the frequency of non-LERF core damage sequences in this ILRT evaluation. And also conservatively, the higher LERF estimate is used when assessing total plant LERF.

Table 7-3 summarizes the External Events LERF and ΔLERF values attributed specifically to non-detected containment failures. Reported ΔLERF values are relative to the 3 per 10 year surveillance interval.

Table 7-3: External Events LERF and ΔLERF specific to each ILRT Test Interval		
ILRT Interval	LERF (/yr)	ΔLERF (/yr)
3 per 10 years	1.39E-08	-
1 per 10 years	4.64E-08	3.25E-08
1 per 15 years	6.97E-08	5.58E-08

Assessment against RG 1.174 ΔLERF and Total LERF Acceptance Guidelines

The total Duane Arnold plant LERF across all hazards, based on the frequency of Internal Events LERF from Table 5-1 and the frequency of External Events LERF from Table 7-2, is calculated as follows:

= $LERF_{IE}$ (per high early release frequency specified in Table 5-1)+ $LERF_{EE}$ (Table 7-2)
= 1.46E-06 + 8.33E-06 = 9.79E-06/yr

The ΔLERF for the 1/10 and 1/15 ILRT intervals, relative to the base 3/10 interval, are as follows:

$$\Delta LERF_{1/10} = \Delta LERF_{IE,1/10} \text{ (Table 6-8)} + \Delta LERF_{EE,1/10} = 1.50E\text{-}8 + 3.25E\text{-}08 = 4.74E\text{-}08/yr$$
$$\Delta LERF_{1/15} = \Delta LERF_{IE,1/15} \text{ (Table 6-8)} + \Delta LERF_{EE,1/15} = 2.57E\text{-}08 + 5.58E\text{-}08 = 8.14E\text{-}08/yr$$

ΔLERF for both the 1/10 and 1/15 ILRT intervals falls within RG 1.174 Region III (Reference 4), where ΔLERF is less than 1.0E-07/yr. Proposed changes in this region are acceptable provided total LERF is less than 1.0E-04/yr. The Duane Arnold total LERF 9.79E-06/yr, and therefore the proposed 1/10 and 1/15 ILRT intervals satisfy RG 1.174 Region III (Reference 4).

8 Conclusions

Based on the results from Section 6 and the sensitivity calculations presented in Section 7, the following conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test frequency to once in fifteen years is as follows:

- Regulatory Guide 1.174 (Reference 4) provides guidance for determining the risk impact of plant specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes *in risk as resulting in increases of CDF below 10-6/yr and increases in LERF below 10-7/yr. Since* the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is conservatively estimated as 2.57E-08/yr for DAEC. As such, the estimated change in LERF for DAEC is determined to be "very small" using the acceptance guidelines of RG 1.174 (Reference 4).

- Regulatory Guide 1.174 also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. An additional assessment of the impact from External Events was also made. In this case, the total LERF including External Events was conservatively estimated as <9.79-06/yr for Duane Arnold. This is below the RG 1.174 acceptance criteria for total LERF of 1.00E-05/yr and therefore this change satisfies both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- The change in Type A test frequency to once per fifteen years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.55E-02 person-rem/yr for Duane Arnold. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 (Reference 1) and EPRI Report No. 1009325 (Reference 20). Moreover, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the three in ten year interval to a permanent one time in fifteen year interval is 0.61% for Duane Arnold. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of ≤1.5 percentage points are very small. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 (Reference 1) and EPRI Report No. 1009325 (Reference 20). DAEC proves to be below 1.5 percentage points and thus is considered to be very small.

Therefore, permanently increasing the ILRT interval to fifteen years is considered to be a very small change to the DAEC risk profile.

8.1 Previous Assessments

The NRC in NUREG-1493 (Reference 6) has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per ten years to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage rate tests is possible with minimal impact on public risk. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for DAEC confirm these general findings on a plant specific basis considering the severe accidents evaluated for DAEC, the DAEC containment failure modes, and the local population surrounding DAEC.

ATTACHMENT 5 to NG-15-0234

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-143)
EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY

DOCUMENTATION OF PROBABILISTIC RISK ASSESSMENT TECHNICAL ADEQUACY

2 pages follow

PRA Quality

The DAEC Internal Events PRA includes both Level 1 and Level 2 models for internal initiating events. The model is maintained and upgraded in accordance with DAEC PRA maintenance and update procedures. The model routinely incorporates plant design changes, procedure changes, plant operating data, current industry PRA methods, review comments, and general improvements identified by the NRC. The PRA has undergone a variety of assessments of its technical capability, as summarized in the following paragraphs.

DAEC was the first non-pilot plant to have a PRA Peer Evaluation (Reference 1) in 1997. The PRA certification process used a team of experienced PRA and system analysts to provide both an objective review of the PSA technical elements and a subjective assessment based on their PRA experience regarding the acceptability of the PRA elements.

During 2005 and 2006, the DAEC PRA model results were evaluated in the BWR Owners Group (BWROG) PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process.

Following issuance of the ASME PRA Standard and its endorsement by the NRC in RG 1.200 Rev. 1 (Reference 4), DAEC performed a detailed self-assessment (Reference 2) of the DAEC PRA model and documentation in preparation for the BWROG PRA Peer review in the fall of 2007. This review was performed using the NEI recommended self-assessment process as endorsed by the NRC in RG 1.200, Rev. 1 (Reference 4).

In December 2007 a peer review was held at the NextEra Energy offices in Juno Beach, FL, under the auspices of the BWROG, using the NEI 05-04 PRA Peer Review process and the ASME PRA Standard ASME RA-Sb–2005 (Reference 2) (along with the NRC clarifications provided in Regulatory Guide 1.200, Rev. 1 (Reference 4).). The 2007 DAEC PRA Peer Review was a full-scope review of all the Technical Elements of the internal events, at-power PRA. The BWROG peer review final report was issued in May 2008 (Reference 6).

A gap analysis for the DAEC PRA 5C model was completed in December 2007 with the final report being issued in May 2008. The 2007 DAEC PRA Peer Review was a full scope review of all the technical elements of the internal events, at-power PRA. This gap analysis was performed against PRA Standard RA-Sb-2005 (Reference 3) and associated NRC comments in Regulatory Guide 1.200, Rev. 1 (Reference 4). The gap analysis identified 83 supporting requirements for which potential gaps to Capability Category II existed. The PRA model was revised to address identified gaps.

In March 2011, a focused PRA Peer Review assessed all previous 2007 full scope peer review findings and observations, including the adequacy or their dispositions. This focused scope review was primarily performed to assess the internal events PRA model adequacy in support of the fire PRA and NFPA 805 transition. The review identified 4 supporting requirements as 'Not Met' and 3 as meeting Capability Category I (CC I) with a total of 12 findings. Letters NG-11-0299 (Reference 8) documents resolution of 5 of these findings, and NG-11-0135 (Reference 5)

documents evaluation and conclusion that the remaining 7 issues negligibly affect the RITS 5b application, and these dispositions also apply to the current ILRT application.

In conclusion, the DAEC PRA is judged sufficient for the ILRT interval risk-informed application in accordance with Regulatory Guide 1.200, Rev. 1.

References

1. DAEC PSA Peer Review Certification Report, BWROG/PSA-9701, March 1997.
2. Self-Assessment of the DAEC PRA against the ASME PRA Standard Requirements, November 2007.
3. American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-S-2002), Addenda RA-Sb-2005, December 2005.
4. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 1, January 2007.
5. Letter NG-11-0135 from DAEC to USNRC, "Clarification of Information Contained in License Amendment Request (TSCR-120): Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TSTF-425, Rev. 3)", dated April 20, 2011.