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RS-21-014 10 CFR 50.90

February 24, 2021

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: License Amendment Request for One-Time Extension of the Containment Type A Integrated Leakage Rate Test Frequency

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed change would allow for a one-time extension to the 15-year frequency of the CPS Unit 1 containment leakage rate test (i.e., integrated leakage rate test (ILRT) or Type A test). This test is required by Technical Specifications Section 5.5.13, "Primary Containment Leakage Rate Testing Program." The proposed one-time change would permit the current ILRT interval of 15 years to be extended by eight months.

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS page.
- Attachment 3 provides an assessment of risk associated with the one-time extension.

The proposed change has been reviewed by the Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed change by July 30, 2021. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 24th day of February 2021.

Respectfully,

Patrick R. Simpson
Sr. Manager Licensing

Attachments:

1. Evaluation of Proposed Change
2. Markup of Technical Specifications Page
3. Risk Assessment for CPS Regarding the ILRT (Type A) One-Time Extension Request

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Clinton Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

SUBJECT: License Amendment Request for One-Time Extension of the Containment Type A Integrated Leakage Rate Test Frequency

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed change would allow for a one-time extension to the 15-year frequency of the CPS Unit 1 containment leakage rate test (i.e., integrated leakage rate test (ILRT) or Type A test). This test is required by Technical Specifications (TS) Section 5.5.13, "Primary Containment Leakage Rate Testing Program." The proposed one-time change would permit the current ILRT interval of 15 years to be extended by eight months.

The last ILRT at CPS was completed in February 2008. Therefore, based on the 15-year ILRT frequency, the next ILRT must be performed by February 2023. In order to perform the test during a regularly scheduled refueling outage, the next ILRT at CPS would need to be completed no later than startup after the fall 2021 refueling outage.

On January 31, 2020, the U.S. Department of Health and Human Services declared a Coronavirus Disease (COVID-19) public health emergency (PHE) for the United States. Subsequently, the Centers for Disease Control and Prevention issued recommendations (e.g., social distancing, limiting assemblies) to limit the spread of COVID-19.

In response to concerns of a continuation of the COVID-19 public health emergency, in the interest of personnel safety, and to preclude the potential for transmittal and spread of COVID-19, EGC requests a one-time extension of the CPS Type A test interval. This request is part of an overall effort by EGC to reduce the number of outside personnel required on-site, and the overall outage scope, in response to the developing COVID-19 pandemic situation while maintaining the safety and reliability of the plant for the next operating cycle. This effort assures that the overriding priority of nuclear safety is maintained while providing for plant personnel and public safety and health. Performing the Type A test would require vendor personnel from across the United States working alongside plant personnel in close proximity for extended periods of time. Including the ILRT in the fall outage scope would also increase the overall outage duration by approximately three days, increasing the amount of time that supplemental workforce would remain onsite. Therefore, the proposed change to extend the ILRT interval by eight months would allow CPS personnel to more effectively follow recommendations for social distancing.

2.0 DETAILED DESCRIPTION

CPS TS 5.5.13, "Primary Containment Leakage Rate Testing Program," currently states, in part:

> 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. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance–Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as

> modified by the following exception: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests.

The proposed change to CPS TS 5.5.13 will add an exception to allow for the performance of the next Type A test no later than October 2023, which represents an extension of eight months. This will allow performance of the test during the fall 2023 refueling outage, which is currently scheduled to begin in September 2023. In addition, the proposed change adds a second exception to allow the Type A test to be extended indefinitely if the test interval ends while primary containment integrity is not required (i.e., TS 3.6.1.1, "Primary Containment," does not require the primary containment to be operable in Modes 4 and 5). In this case, the second exception requires that the Type A test be performed prior to entering Mode 2. The revised wording states:

> 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. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance–Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests, (2) the next Type A test performed after the February 2008 Type A test shall be performed no later than October 31, 2023, and (3) if the Type A test has not been performed by October 31, 2023, and the unit is in Mode 4 or 5, the Type A test shall be performed prior to entering Mode 2

A markup of the proposed change is provided in Attachment 2.

3.0 TECHNICAL EVALUATION

3.1 Description of Primary Containment System

The containment consists of a right circular cylinder with a hemispherical domed roof and a flat base slab. It is constructed of reinforced concrete and completely lined on the inside of the walls and dome with 1/4-inch stainless steel plate below elevation 735 feet 0 inch and with carbon steel plate of at least 1/4-inch thickness above elevation 735 feet 0 inch.

The principal dimensions of the containment are:

- height above basemat: 215 feet 0 inch;
- inside diameter: 124 feet 0 inch;
- wall thickness: 3 feet 0 inch;
- dome thickness: 2 feet 6 inches; and
- mat thickness: 9 feet 8 inches.

The containment structure supports the polar crane, galleries, and the access ramp to the refueling floor. The lower section of the containment acts as the outer boundary of the suppression pool. Two double-door personnel locks, one located at the refueling floor and the other located at the grade floor, permit access to the containment. An equipment hatch is located at the grade floor. The equipment hatch is sealed during normal operation, or at other times when primary containment is required.

3.1.1 Pipe Penetrations

Pipe penetrations for process pipes which pass through the containment and drywell walls may be classified into three types. Type 1 is used for high-energy lines requiring guard pipes when passing through both the containment and drywell walls. Types 2 and 3 are used for the remainder of process pipes which pass through the containment. CPS Updated Safety Analysis Report (USAR) Figure 3.8-11 (Reference 1) shows the basic design of the three penetration types along with the inclined fuel transfer tube detail.

Type 1 penetrations consist of a guard pipe anchored at the containment wall and welded to the flued head. The flued head is welded to the process pipe using a gradual buildup weld. The process pipe is allowed free axial thermal movement from the flued head through the drywell.

The guard pipe is allowed free axial thermal movement from the containment anchor point through its own sleeve at the drywell wall. Bellows, anchored to the drywell and welded to the guard pipe, will act as a seal for normal drywell environmental conditions. They are designed for thermal guard pipe expansion and relative seismic motion of guard pipe and drywell.

Type 2 penetrations consist of a penetration sleeve anchored in the containment and extending to just inside the liner. Full penetration welds are used to weld the flued head to the process pipe.

Type 3 penetrations consist of the sleeve anchored in the containment wall and extending just beyond the containment liner. Full penetration welds are used to attach the cover plate to the process pipe.

3.1.2 Electrical Penetrations

Dual header plate type electrical penetration assemblies are used to extend electrical conductors through the containment structure pressure boundary. These penetration assemblies are designed, fabricated, tested, and installed in accordance with the requirements of IEEE 317, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," dated December 1976 (Reference 2).

3.1.3 Personnel and Equipment Access Hatches

Two personnel access locks are provided for access to the interior of the containment.

Each personnel lock consists of an interlocked double door of welded steel assembly. Each door is equipped with a valve for equalizing pressure across the door such that the doors are not operable unless the pressure is equalized.

The two doors in each personnel lock are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. An emergency lighting and communication system operating from an external auxiliary energy source is provided within the personnel locks.

The equipment hatch is fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is welded to the containment liner.

Provisions are made to pressure test the space between the double gaskets of the door flanges. The weld seam test channels at the liner joint and the dished door are provided to monitor any leakage during leak rate testing.

3.1.4 Fuel Transfer Penetration

The inclined fuel transfer tube, along with the three types of process pipe penetrations, penetrates the containment wall through the fuel transfer penetration. This is essentially a 3/4-inch-thick carbon steel rolled plate pipe sleeve of 40-inch ID with a 36-inch standard flange on the containment side. The fuel transfer penetration forms a part of the containment boundary.

A containment isolation assembly containing a blind flange and a bellows that connects from the containment isolation assembly to the building containment penetration are provided to make containment isolation. A hand-operated 24-inch gate valve is provided to isolate the reactor building pool water from the transfer tube so that the blind flange can be installed.

Normally, containment isolation is made by the containment isolation assembly and blind flange, containment bellows and the steel containment penetration. Special gaskets and double ply bellows are provided for leak checking to assure containment isolation. Alternatively, the blind flange may be removed for short periods of time during power operation, as allowed by the Technical Specifications. Leak testing of this alternative configuration (including transfer tube, associated drain line isolation valves, bellows, and flange connections) is not required because:

- these periods of time are short with respect to the overall duration of power operations,
- the transfer tube terminates below the fuel building spent fuel pool water level, and
- the configuration of the transfer tube drain line is controlled by the TS.

3.1.5 Containment Liner

The containment wall liner is anchored to the wall with structural T sections. When a stiffener is cut to avoid interference with an insert assembly, welded studs are provided to restore anchorage of the liner plate.

Typical spacing of the liner anchors is 15 inches in the containment wall and the dome.

The top of the exposed base slab is lined with 1/2-inch and 1/4-inch stainless steel plate which serves as a leaktight boundary. The drywell wall and the sump floor are anchored through the base liner plate and into the base slab. The spans of liner panels in the basemat area are:

- pedestal cavity area: 3 feet 0 inch;
- sump floor area: 6 feet 0 inch; and 20 feet 0 inch;
- suppression pool area: 3 feet 0 inch to 4 feet 8 1/4 inches (max.).

Leak test channels are provided at the liner seams in the suppression pool area and in the containment wall up to elevation 757 feet 0 inch. The containment liner in the wet areas of the suppression pool is of stainless steel to minimize corrosion problems.

3.2 Description of Drywell

The drywell is a cylindrical reinforced concrete structure which surrounds the reactor pressure vessel and its support structure. The drywell is structurally designed as follows:

- to provide structural support to containment pools, main steam tunnel and reactor water cleanup (RWCU) compartments;
- to channel steam release from a LOCA through the horizontal vents for condensation in the suppression pool;
- to protect the containment vessel from internal missiles and/or pipe whip;
- to provide anchor points for pipes; and
- to provide a support structure for the work platforms, monorails, pipe supports, and restraints that are located in the annulus between the drywell and the containment vessel.

The inside diameter of the drywell cylinder is 69 feet 0 inch, and the wall thickness is 5 feet 0 inch. The top of the drywell consists of a flat annular slab 6 feet 0-inch-thick at elevation 803 feet 3 inches. The drywell wall is rigidly attached to the base slab at elevation 712 feet 0 inch. A steel head which can be removed to allow access to the reactor is located over the opening in the annular slab.

The lower portion of the drywell wall is submerged in the suppression pool. Three rows of circular suppression pool vents, 34 vents per row, penetrate the drywell wall below the normal level of the suppression pool. The surfaces of the drywell wall exposed to the suppression pool are lined with stainless steel clad plate 1-inch-thick, which is designed to act compositely with the drywell wall. Above the level of the suppression pool a carbon steel form plate 1/2-inch-thick is provided on the interior surfaces of the cylinder walls and top slab. Structural T's and headed studs are attached to the form plate to provide mechanical anchorage of the plate to the concrete and to stiffen the liner for construction loads. The form plate provides a surface for forming the drywell walls and ceiling and minimizes bypass leakage, if any, through the drywell wall under accident conditions.

3.2.1 Pipe Penetrations

Piping penetrations are of the types used in the containment wall and are discussed in Section 3.1.1 above.

3.2.2 Electrical Penetrations

Electrical penetrations feature an epoxy-based sealing compound qualified for harsh environmental conditions, which surrounds the cables that pass through the penetration. The penetration consists of a rigid steel conduit welded to the drywell liner.

3.2.3 Personnel and Equipment Access Hatches

Access to the drywell is provided by the drywell personnel lock, a personnel hatch located in the drywell ceiling, and the drywell equipment hatch. The personnel lock consists of an interlocked, double-door, welded steel assembly. Each door is equipped with a valve for equalizing pressure across the door such that the doors are not operable unless the pressure is equalized.

The two doors in the personnel lock are interlocked to prevent both being opened simultaneously, and to ensure that one door is completely closed before the opposite door can be opened. An emergency lighting and communication system operating from an external auxiliary energy source is provided within the personnel lock interior.

The personnel hatch located in the drywell ceiling consists of a double-gasketed bolted flange.

The equipment hatch is fabricated from welded steel and furnished with a double-gasketed flange and bolted, dished door. Provision is made to pressure test the space between the double gaskets of the door flanges. A shield wall is provided with the same shielding requirements as the drywell wall.

3.2.4 Access for Refueling Operations

The drywell head is removed during refueling operations. This head is held in place by bolts and sealed with a double seal. It is opened only when the primary coolant temperature is below 200°F and the core is sub-critical. The double seal provides a method for determining the leak tightness of the seal without pressurizing the drywell.

3.2.5 Suppression Pool Weir Wall

The suppression pool weir wall, located inside the drywell, acts as the inner boundary of the suppression pool. It is constructed of reinforced concrete and extends from the outer edge of the drywell sump floor. The weir wall is lined with 1/4-inch stainless steel plate on the suppression pool side to protect the concrete from demineralized water.

The principal dimensions of the weir wall are:

- Inside diameter: 61 feet;
- Wall thickness: 1 foot 10 inches;
- Height above basemat: 23 feet 9 inches; and
- Height above sump floor: 12 feet 7 1/4 inches.

3.2.6 Process Pipe Tunnel

The process pipe tunnel provides shielding for the process piping between the drywell and the containment. It is designed as an integral part of the drywell structure and is constructed of reinforced concrete. The arrangement at the containment wall permits differential movement between the tunnel and the containment. Doorways connect the tunnel to the containment volume.

3.2.7 Drywell Sump Floor

The drywell sump floor is a thick slab of reinforced concrete which rests on the basemat and supports the suppression pool weir wall and the reactor pedestal. A stainless steel liner is provided on the suppression pool side to protect the concrete from demineralized water.

The sump floor has the following principal dimensions:

- inside diameter: 18 feet 6 inches;
- outside diameter: 64 feet 8 inches; and
- thickness: 11 feet 1 3/4 inches.

3.3 Chronology of Testing Requirements of 10 CFR 50, Appendix J

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. Appendix J also ensures that 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 the systems and components penetrating primary containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations, and; (3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, RG 1.163 (Reference 3) was issued. The RG endorsed NEI 94-01, Revision 0, (Reference 4) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A (ILRT) test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 5) and Electric Power Research Institute (EPRI) TR-104285 (Reference 6) both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small. In addition to the 10-year ILRT interval, provisions for extending the test interval an additional 15 months was considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, but that this extension of interval "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Revision 2-A (Reference 7), was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the NRC Safety Evaluation (SE) on NEI 94-01. NEI 94-01, Revision 2-A, includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163 (Reference 3). It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

On December 8, 2008, the NRC issued Regulatory Issue Summary (RIS) 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50" (Reference 8). The RIS clarifies the NRC position concerning licensee requests to extend Type A test intervals beyond 15 years, stating that a licensee can commence the test no later than the last day of the month in which it becomes due, without seeking NRC approval through a license amendment. The RIS also endorses the statement made in NEI 94-01, Revision 2, that if the test interval ends while primary containment integrity is not required, or is required solely for shutdown activities, the test interval may be extended indefinitely, but a Type A test shall be completed prior to entering the operating mode requiring primary containment integrity. In addition, the RIS states that any extension beyond the end of the month in which the test is due would require a license amendment request. Per RIS 2008-27, the license amendment request should demonstrate:

- a sound technical justification and/or undue hardship or unusual difficulty;
- the requested amendment poses minimal safety risk;
- acceptable plant-specific containment performance, including a plant-specific risk-informed analysis; and
- that containment does not have a history of significant degradation issues.

In 2012, NEI 94-01, Revision 3-A (Reference 9), was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J and includes provisions for extending Type A ILRT intervals to up to 15 years. NEI 94-01 has been endorsed by RG 1.163 and NRC SEs of June 25, 2008 (Reference 10) and June 8, 2012 (Reference 11) as an acceptable methodology for complying with the provisions of Option B in 10 CFR 50, Appendix J. The regulatory positions stated in RG 1.163 as modified by References 10 and 11 are incorporated in this document. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

The NRC has provided guidance concerning the use of test interval extensions in the deferral of ILRTs beyond the 15-year interval in NEI 94-01, Revision 2-A, NRC SE Section 3.1.1.2 which states, in part:

> Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists.

The NRC has also provided the following concerning the extension of ILRT intervals to 15 years in NEI 94-01, Revision 3-A, NRC SE Section 4.0, Condition 2, which states, in part:

> 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.

3.4 Current CPS ILRT Requirements

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." On June 21, 1996, the NRC issued Amendment 105 for CPS (Reference 12) authorizing the implementation of 10 CFR 50, Appendix J, Option B for Type A, B and C tests with the following exemptions from the requirement of 10 CFR 50, Appendix J - Option B, paragraph III.B:

- Exempting the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for local leak rate tests; and
- Exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests.

In addition to the above exemptions, the following, previously approved exemptions from the requirements of 10 CFR 50, Appendix J - Option A were determined to be no longer applicable:

- exemption from paragraph III.D.2(b)(ii) to permit substituting the air lock door seal leakage for the entire primary containment air lock test; and
- exemption from paragraph III.D.1(a) pertaining to the requirement to conduct the third Type A test during the last outage within the 10-year inservice inspection interval.

On September 26, 2017, the NRC issued Amendment 214 for CPS (Reference 13). This amendment approved a permanent extension of the Type A ILRT frequency from once every 10 years to once every 15 years in accordance with NEI 94-01, Revision 3.A, and the conditions and limitations specified in NEI 94-01, Revision 2-A. The amendment also approved an extension of the containment isolation valve leakage rate testing frequency from 60 months to 75 months for Type C leakage rate testing of selected components in accordance with NEI 94-01, Revision 3-A.

With the issuance of Amendment 214, the due date for the CPS Type A test moved from February 2018 to February 2023. The proposed change would defer the Type A test for CPS until no later than October 2023. This will allow performance of the test during the fall 2023 refueling outage, which is currently scheduled to begin in September 2023. This represents an extension of approximately eight months. The intervals for the Type B and Type C tests at CPS would remain unchanged at 120 months and 75 months, respectively.

3.5 Integrated Leakage Rate Testing History

Previous CPS ILRT results have confirmed that the containment is acceptable, with considerable margin, with respect to the TS acceptance criterion of 0.65% of primary containment air weight per day at the design basis loss-of-coolant accident pressure. Since the last two Type A test results meet the performance leakage rate criteria from NEI 94-01, Revision 3-A (Reference 9), a test frequency of 15 years would be acceptable. Table 3.5-1 lists the past periodic Type A ILRT results for CPS.

Table 3.5-1, CPS Type A ILRT History		
Test Date	As-Found Leakage Rate (Containment air weight %/day)	As-Left Leakage Rate (Containment air weight %/day)
January 1986 (Preoperational) (1)	0.2930	0.3463
November 1986 (Preoperational)	0.2875	0.2933
February 1991	0.2209	0.2291
November 1993	0.2089	0.2204
February 2008	0.2708	0.226

(1) Subsequent to the ILRT, a hole was discovered through the containment liner plate. This hole was evidently in existence during the ILRT and was created by the removal of temporary attachment CL-J-12-4. After repair, the hole was retested utilizing a leak chase channel and bubbler. The retest showed zero leakage. ILRT was re-performed in November 1986.

No modifications that require a Type A test are planned at CPS prior to the fall 2023 refueling outage, when the next Type A test will be performed in accordance with this proposed change. Any unplanned modifications to containment prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J. There have been no pressure or temperature excursions in the CPS containment which could have adversely affected containment integrity. There is no anticipated addition or removal of plant hardware within containment which could affect leak-tightness.

3.6 Plant Specific Confirmatory Analysis

3.6.1 Methodology

An evaluation has been performed to provide an assessment of the risk associated with implementing a one-time extension of the CPS containment Type A ILRT interval by approximately eight months from the currently approved value of 15 years to 15.7 years. The risk assessment follows the guidelines from NEI 94-01, Revision 3-A (Reference 9), the methodology outlined in Electric Power Research Institute (EPRI) Topical Report (TR) 104285 (Reference 6), as updated by the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (EPRI TR-1018243) (Reference 14), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change to a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 15), and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage going undetected during the extended test interval (Reference 16).

3.6.2 Summary of Plant-Specific Risk Assessment Results

The risk assessment determined that increasing the ILRT interval on a one-time basis to a one-in-15.7-year frequency is not considered to be significant since it represents only a small change in the CPS risk profile. Details of the CPS risk assessment are contained in Attachment 3 to this letter. The plant-specific results for a one-time extension of the CPS ILRT surveillance interval from the current 15 years to 15.7 years are summarized below.

- RG 1.174 (Reference 15) 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 1E-6/yr and increases in large early release frequency (LERF) below 1E-7/yr. "Small" changes in risk are defined as increases in CDF below 1E-5/yr and increases in LERF below 1E-6/yr. Since the ILRT extension has no impact on CDF for CPS, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT interval with corrosion included is 9.80E-10/yr, which falls within the "very small" change region of the acceptance guidelines in RG 1.174.

- The change in dose risk for changing the Type A ILRT interval from once-per-15 years to once-per-15.7-years, measured as an increase to the total integrated dose risk for all accident sequences, is 3.78E-04 person-rem/yr using the EPRI guidance with the corrosion included. NEI 94-01 (Reference 4) states that a "small" population dose is defined as an increase of $\leq$ 1.0 person-rem per year, or $\leq$ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The results of this calculation meet these criteria.

- The increase in the conditional containment failure probability from the 15-year interval to the 15.7-year interval including corrosion effects using the EPRI guidance is 0.03%, which is less than the acceptance criteria of 1.5%.

- To determine the potential impact from external events, an additional assessment from the risk associated with external events was performed. The total increase in LERF due to internal events and external events is 4.22E-08/yr, which falls within the "very small" change region of the acceptance guidelines in RG 1.174.

3.7 Non-Risk Based Assessment

Consistent with the defense-in-depth philosophy discussed in RG 1.174, CPS has assessed other non-risk based considerations relevant to the proposed amendment. CPS has multiple inspections and testing programs that ensure the containment structure remains capable of meeting its design functions and that are designed to identify any degrading conditions that might affect that capability. These programs are discussed below.

3.7.1 CPS Protective Coating Program

CPS has committed to follow NRC RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0. The RG describes a method to comply with requirements of Appendix B to 10 CFR 50, and invokes several ANSI Standards. Standards pertinent to coatings are ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," ANSI N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," and ANSI N5.12, "Protective Coatings for the Nuclear Industry." (ANSI N5.9, referenced in ANSI N101.2, was replaced by ANSI N5.12 in 1974, prior to CPS obtaining a construction permit).

A program to maintain containment coatings was developed to meet the requirements of Regulatory Guide 1.54, Revision 0. This program is implemented using CPS Procedure 1080.01, "CPS Protective Coating Program." Every refueling outage (i.e., every 24 months), a preventive maintenance activity to inspect the protective coatings in the containment building, including the drywell, is performed.

The most recent inspection was performed during a refueling outage (i.e., C1R19) in September 2019. During outage C1R19, all elevations of the Drywell Liner Plate, Inner Wall and floors, Containment Liner Plate, Inner Wall and Floors and Containment Steam Tunnel were inspected to identify degraded coatings.

There will be no change to the schedule for these inspections as a result of the extended ILRT interval.

Unqualified/Degraded Coatings in Containment

CPS 1080.01 requires that a visual inspection be performed to establish baseline condition of Containment and Drywell coatings, and to identify Unqualified and Degraded coatings. The baseline condition is the first issue of the Combined Degraded and Unqualified Coatings list. It was completed on August 19, 1998.

As of September 2019, there are 2,303.07 pounds of combined Degraded and Unqualified coatings. Allowed is 20,000 pounds per Engineering Evaluation EE-00-143, Rev. 0. Therefore, 2,303.07 pounds of combined Degraded and Unqualified coatings would not be of concern during a LOCA.

3.7.2 Containment Inservice Inspection Program (CISI)

CPS performs a comprehensive primary containment inspection to the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Inservice Inspection," Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants." The CPS Containment Inservice Inspection Program began development in 1996 and the initial inspections were completed in September 2001. The components subject to Subsection IWE and IWL requirements are those which make up the containment structure, its leak-tight barrier (including integral attachments), and those that contribute to its structural integrity. Specifically included are Class MC pressure retaining components, including metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. The ASME Code Inspection Plan was developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by NRC final rulemaking to 10 CFR 50.55a published in the Federal Register on August 8, 1996.

The initial inspections of the CPS metal/concrete containment have been completed. Various indications were observed, documented, and evaluated and determined to be acceptable. No areas of the containment liner surfaces require augmented examination. No loss of structural integrity of primary containment was observed.

There will be no change to the schedule for these inspections as a result of the extended ILRT interval. Inspection period dates for the 2nd and 3rd CISI inspection intervals are displayed in Tables 3.7.2-1 and 3.7.2-2.

Table 3.7.2-1, CPS IWE Examination Schedule 2nd and 3rd Ten-Year Inspection Interval	
2nd Interval 1st Period	9/10/08 to 9/9/11 C1R12
2nd Interval 2nd Period	9/10/11 to 9/9/15 C1R13 C1R14 C1R15
2nd Interval 3rd Period	9/10/15 to 9/9/18 C1R16 C1R17 C1R18
3rd Interval 1st Period	9/10/18 to 9/9/21 C1R19
3rd Interval 2nd Period	9/10/21 to 9/9/25 C1R20 C1R21
3rd Interval 3rd Period	9/10/25 to 9/9/28 C1R22 C1R23

Table 3.7.2-2, CPS IWL Examination Schedule 2nd and 3rd Ten-Year Inspection Interval
C1R12 – 2010
C1R15 – 2015
C1R19 – 2019
C1R22 – 2025

Edition and Addenda of the ASME Section XI Code

CPS is committed to the following Editions and Addenda of the ASME Section XI Code.

- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Component, 2013 Edition with no Addenda (hereafter referred to as the ASME Section XI Code).

- The applicable requirements of Subsection IWA (General Requirements), Subsection IWE (Requirements for Class MC and Metallic Liners of Class CC Components), and Subsection IWL (Requirements for Class CC Concrete Components) of the 2013 Edition with no Addenda and the ASME Section XI Code shall apply to components and items classified as ASME Code Class MC or ASME Code Class CC.

- In addition to the requirements of the ASME Section XI Code, the applicable modifications and limitations outlined in 10 CFR 50.55a(b)(2)(viii) and 50.55a(b)(2)(ix) shall also be implemented.

Code Cases

There are no Code Cases implemented at this time.

Relief Requests

There are no Relief Requests implemented at this time.

Identification of Class MC and/or CC Exempt Components

Table 3.7.2-3, Class MC and/or CC Exempt Components			
Exam Category	**Item Number**	**Description**	**Applicability to CPS**
E-A	E1.30	Moisture Barriers	Not Applicable
L-B	L2.10	Tendon	Not Applicable
	L2.20	Wire or Strand	Not Applicable
	L2.30	Anchorage Hardware and Surrounding Concrete	Not Applicable
	L2.40	Corrosion Protection Medium	Not Applicable
	L2.50	Free Water	Not Applicable

Augmented Inspections

Table 3.7.2-4, Augmented Inspections			
Exam Category	**Item Number**	**Description**	**Total Number of Components**
E-C Containment Surfaces Requiring Augmented Examination	E4.11	Visible Surfaces	0
	E4.12	Surface Area Grid Minimum Wall Thickness Location	0
L-A Concrete	L1.12	Suspect Areas	0

Inaccessible Areas

For Class MC applications, CPS shall 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, CPS shall 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.

In addition, for Class CC applications, for each inaccessible area of concrete identified for evaluation under IWL-2512(a), or identified as susceptible to deterioration under IWL-2512(b), CPS shall provide the following in the ISI summary report, as required by 10 CFR 50.55a(b)(2)(viii)(E):

- 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.

The most recent ISI summary report for CPS was submitted to the NRC in Reference 17. As stated in Reference 17, all Class MC components required to be examined, and all Class CC components, were scheduled and examined in C1R19. No unacceptable indications were identified.

CPS has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, EGC actively participates in various nuclear utility owner's 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 applicably to CPS. 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.

3.7.3 Supplemental Inspection Requirements

TS 5.5.13 requires, in part, that a primary containment leakage rate testing program be established in accordance with the guidelines contained in NEI 94-01, Revision 3-A. This requires that a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity be conducted. This inspection must be conducted prior to each Type A test and during at least three (3) other outages before the next Type A test if the interval for the Type A test has been extended to 15 years in accordance with the following sections of NEI 94-01, Revision 3-A:

- Section 9.2.1, "Pretest Inspection and Test Methodology"
- Section 9.2.3.2, "Supplemental Inspection Requirements"

In addition to the inspections performed by the IWE/IWL Containment Inspection Program, procedure CPS 9861.01, "Integrated Leak Rate Test," requires that the structural integrity of the exposed accessible interior and exterior surfaces of the drywell and the containment, including the liner plate, shall be determined by a visual inspection of those surfaces prior to the Type A Containment Leak Rate Test. This inspection also fulfills the surveillance requirement of TS SR 3.6.1.1.1 and NEI 94-01.

3.7.4 Type B and Type C Testing Programs

The CPS Type B and Type C testing programs require testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with 10 CFR 50, Appendix J, Option B, and RG 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Types B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. In accordance with TS 5.5.13, the allowable maximum pathway total Types B and C leakage is 0.6 L_a where L_a equals approximately 361,277 sccm.

As discussed in NUREG-1493 (Reference 5), Type B and Type C tests can identify the vast majority of all potential containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

A review of the Type B and Type C test results from 2008 through 2019 for CPS has shown substantial margin between the actual As-Found (AF) and As-Left (AL) outage summations and the regulatory requirements. Table 3.7.4-1 provides local leak rate test (LLRT) data trend summaries for CPS Unit 1 inclusive of the CPS 2008 ILRT.

Table 3.7.4-1, CPS Type B and C LLRT Combined As-Found/As-Left Trend Summary				
RFO	**As-Found Min Path (sccm)**	**As-Found Min Path as % of L_a**	**As-Left Max Path (sccm)**	**As-Left Max Path as % of L_a**
C1R11 2008	11597.7	3.21	24924.6	6.89
C1R12 2010	18425.3	5.1	36848.3	10.19
C1R13 2011	31551.3	8.73	52689.2	14.58
C1R14 2013	33798.1	9.35	66201.4	18.32
C1R15 2015	35305.4	9.77	66465.1	18.39
C1R16 2016	35389.95	9.79	66867.6	18.51
C1R17 2017	31361.3	8.68	66663.7	18.45
C1R18 2018	32931.69	9.11	66270.9	18.34
C1R19 2019	36982.85	10.24	65937.7	18.25

3.8 Results of Recent Inspections

3.8.1 Containment and Drywell Coatings – C1R19 - 2019

3.8.1.1 Drywell

<u>723' Elevation</u>

On the 723' elevation, the condition of the Liner Plate in the Drywell basement is in good condition. The liner plate was originally coated with an inorganic zinc primer and top coated with a phenolic epoxy. Equipment and scaffolding transport during outages have resulted in the most common cause of impact damage to the containment floor and liner plate. This is a normal coating maintenance issue that does not generally impact operability. No identifiable additional areas of mechanical damage or coating defects were observed from what was reported in the previous outage report. Generally, the coatings at the concrete wall, liner plate, and existing steel are in good condition. However, the coating on the concrete drywell floor is in poor condition in limited areas. The degraded coating has been assessed and determined to support operability. Future repairs are being considered.

<u>737' Elevation</u>

The liner plate on this elevation exhibits a few areas of damaged topcoat coating flaking and areas of mechanical damage. During the inspection, areas of light rusting on components and supports were also noted.

<u>768' Elevation</u>

On the 768' elevation, the liner plate is in good condition. Less than a dozen areas of coating defects were found on this elevation. These areas are typical mechanical damaged areas caused by scaffolding and equipment movements during refueling outages. Other defects were from testing areas performed and not repaired or from supports and welding during previous outages and not repaired or repaired improperly. The liner coating is tightly adhering and intact throughout. Handrails on this elevation and throughout the drywell, exhibit mechanical damage but the remaining coating that is still in place appears to be tightly adhered.

3.8.1.2 Containment

<u>737' Elevation and 755' Elevation</u>

During the inspection on Elevation 737' and Elevation 755' the liner plate, floor, piping and inner wall exhibited numerous areas of minor mechanical damage. The design of the BWR/6 Mark III containment makes it harder to inspect the liner plate closely on the suppression pool wall. The suppression pool wall is made of stainless-steel material. The liner plate on Elevation 737' is approximately 15 feet away. On this elevation, there are areas of cracked and flaked coating on the liner plate. On Elevation 755', flaking and cracking are present on the liner plate and concrete inner wall. The floor at this elevation has mechanical damage due to wear and traffic. Plating on the floor at this elevation exhibits surface rust.

<u>778' Elevation and 803' Elevation</u>

On Elevation 778' and 803' the liner plate, supports and valve bodies and inner wall exhibit numerous areas of minor mechanical damage. On elevation 778', areas of mechanical damage, flaking and cracking are present on the liner plate and concrete inner wall. On both elevations, the degraded areas that are cracking and flaking show signs of corrosion with rust coming through the cracks. On elevation 803', several areas of mechanical damage on the inner concrete wall were repaired with new coatings. Overall, for these elevations, the liner plate and inner wall are in good condition.

<u>828' Elevation</u>

Most of the containment floor is usually covered with plastic for FME and to facilitate refuel services activities during outages and as a result, it was not feasible to inspect. However, the area is usually inspected when online. No concerns or issues about coatings degradation on this elevation were identified. The dome was remotely observed from the 828' floor elevation by

UESI level III coatings inspectors with the plant online in 2018, but no issues or concerns were identified.

Containment Steam Tunnel

In the Containment Steam Tunnel, the liner plate, floors and piping exhibit a few areas of minor mechanical damage. Areas of mechanical damage and cracking are present on the liner plate. The degraded areas on the liner plate that are cracking show signs of corrosion with rust coming through. Mechanical damage also has corrosion to substrate on the liner plate. Overall, the liner plate and inner wall are in good condition.

3.8.1.3 Conclusion

The coating assessment identified areas of coating degradation requiring repair. Importantly, no current coating conditions were identified that impact structural integrity, plant operations, or the safe shutdown of the plant. Many of the degraded areas have been identified in previous reports and are repaired by the CPS Protective Coating Program to protect surfaces and equipment from contamination and corrosion. The objective of the CPS Protective Coating Program is to protect plant systems, structures and components from degradation by applying and maintaining protective coatings. The program assures that the station shall be clean, neat and easily maintained from the aspect of personnel safety, housekeeping, and radiological control. Timely repairing of degraded coatings will continue to maintain higher coatings margin for the safe operability of the suction strainers in the suppression pool. The routine coating assessment during outages will continue to be performed.

3.8.2 CISI Program Inspection Results

3.8.2.1 CISI on Containment Liner – C1R19 - 2019

737' Elevation
CISI of the containment liner was performed on the 737' elevation during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

- At 114 degrees, 748' elevation, two 6" diameter peeled, cracked, and stained coatings were identified.
- At 153 degrees, 751' elevation, 4" x 4" peeled, cracked, and stained coatings were identified.
- At 157 degrees, 748' elevation, 1" x 2" cracked and stained coatings were identified.
- At 245 degrees, 738' elevation, two areas 2" x 3" each, missing and cracked paint was identified.
- At 255 degrees, 740' elevation, 5" x 12" area was missing paint.
- At 268 degrees, 735' elevation, 3" x 18'"cracked and stained coatings were identified.
- At 269 degrees, 737' elevation, 1" x 6" cracked and stained coatings were identified.
- At 269 degrees, 735' elevation, 3" x 3" peeled and stained coatings were identified.
- At 273 degrees, 743' elevation, 8" x 12" peeled, cracked, and stained coatings were identified.
- At 308 degrees, 746' elevation, 6" x 3" cracked and stained coatings were identified.

- At 316 degrees, 746' elevation, 12" x 3" cracked and stained coatings were identified.
- At 338 degrees, 746' elevation, 8" x 8" cracked and stained coatings were identified.
- At 340 degrees, 748' elevation, four areas 4" x 4" each, missing and cracked paint was identified.
- At 360 degrees, 745' elevation, 6" x 6" cracked and stained coatings were identified.

<u>755' Elevation</u>
CISI of the containment liner was performed on the 755' elevation during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

- At 8, 12, and 20 degrees, 775' elevation, 6" x 6" no coating and rust were identified.
- At 24 degrees, 776' elevation, 1" x 2" peeled, cracked, and stained coatings were identified.
- At 25 degrees, 756' elevation, 3' x 1' no coating and rust were identified.
- At 70 degrees, 774' elevation, 4" x 4' peeled, cracked, and stained coatings were identified.
- At 96 degrees, 757' elevation, 8" x 8" peeled, cracked, and stained coatings were identified.
- At 192 degrees, 767' elevation, two 1" diameter peeled, cracked, and stained coatings were identified.
- At 200 degrees, 769' elevation, 1" x 2" peeled and stained coatings were identified.
- At 266 degrees, 758' elevation, three small areas cracked and stained coatings were identified.

<u>781' Elevation</u>
CISI of the containment liner was performed on the 781' elevation during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

- At 27 degrees, 780' elevation, 5" x 14" area missing paint was identified.
- At 29 degrees, 777' elevation, four areas 3" x 3" each, missing paint and discoloration were identified.
- At 31 degrees, 798' elevation, 10" x 10" area cracked paint and discoloration were identified.
- At 32 degrees, 786' elevation, 4" x 3' peeled and stained coatings were identified.
- At 46 degrees, 781' elevation, 4" x 4' peeled and stained coatings were identified.
- At 72 degrees, 781 '-784' elevation, three areas 8" x 8" of cracked and stained coatings were identified.
- At 90 degrees, 781' elevation, 2" x 2", 2" x 4", and 2" x 8" areas of cracked and stained coatings were identified.
- At 90 degrees, 790' elevation, 8" x 8" area cracked paint and discoloration were identified.
- At 93 degrees, 798' elevation, 4" x 16' cracked and stained coatings were identified.
- At 93 degrees, 786' - 794' elevation, small areas and a 6" x 6' peeled and stained coatings were identified.

- At 95 degrees, 791' elevation, small areas and 6" x 12" area of cracked and stained coatings, along with a small area of peeled and stained coatings were identified.
- At 125 degrees, 785' elevation, 6" x 24" peeled and stained coatings were identified.
- At 130 degrees, 794' elevation, 8" x 8" cracked and stained coatings were identified.
- At 228 degrees, 788' elevation, 3" x 3' small area of peeled coatings were identified.
- At 265 degrees, 786' -790' elevation, three small areas of cracked and stained coatings were identified.
- Penetration 1MC-50 at 790' elevation, identified cracks in the coatings on the outer ring, at 2:00 and between 10:00 and 11:00. Minor discoloration coming from cracks in coatings.

800' Elevation

CISI of the containment liner was performed on the 800' elevation during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

- At 28 degrees, 824' elevation, 8" x 8" area of cracked coatings were identified.
- At 32 degrees, 803' elevation, six areas 6" x 6" of cracked paint, missing paint, and discoloration were identified.
- At 32 degrees, 822' elevation, 4" x 6" area of cracked paint and discoloration were identified.
- At 35 degrees, 822' elevation, 4" x 12" area of cracked paint and discoloration were identified.
- At 42 degrees, 803' elevation, three areas 2" x 3" of missing and cracked paint were identified.
- At 85 degrees, 815' elevation, 8" x 8" area of peeled coatings were identified.
- At 92 degrees, 806' elevation, small areas had blistered and stained coatings identified.
- At 120 degrees, 809' elevation, small areas of cracked and stained coatings were identified.
- At 131 degrees, 818' elevation, 4" x 6" area missing paint was identified.
- At 135 degrees, 813' elevation, 3" x 3" area missing paint and discoloration were identified.
- At 190 degrees, 823' elevation, stained coatings were identified.
- At 193 degrees, 806' elevation, nine areas 2" x 2" each of missing and cracked paint were identified.
- At 235 degrees, 817' elevation, 6" x 6" areas of cracked and stained coatings were identified.
- At 235 degrees, 817' elevation, 6" x 8" areas of cracked and stained coatings were identified.
- At 257 degrees, 803'-805' elevation, small areas of cracked, stained, and damaged coatings were identified.
- At 268 degrees, 807' elevation, small areas of cracked and stained coatings were identified.
- At 269-276 degrees, 804'-811' elevation, approximately thirty various areas of cracked and stained coatings were identified.
- At 276 degrees, 812'-816' elevation, two areas of cracked and stained coatings were identified.

- At 278 degrees, 817' elevation, 6" x 12" area of cracked and stained coatings were identified.
- At 290 degrees, 813' elevation, 12" x 15" area no coating and mild corrosion were identified.
- At 290-300 degrees, 811'-815' elevation, 2" x 2" and 6" x 6" areas of cracked and stained coatings were identified.
- At 300-310 degrees, 811'-815' elevation, three 2" x 2" areas and a 6" x 6" area of cracked and stained coatings were identified.
- At 310-320 degrees, 806'-815' elevation, 11 small areas of cracked and stained coatings were identified.
- At 320-330 degrees, 811'-827' elevation, approximately 20 small areas of cracked and stained coatings were identified.

<u>828' Elevation</u>
CISI of the containment liner was performed on the 828' elevation during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

- At 5 degrees, 840' elevation, four 6" x 6" areas of discolored and cracked paint were identified.
- At 230 degrees, 848' elevation, three areas with peeling paint were identified.
- At 290 degrees, 832' elevation, one square foot area of discoloration and flaking were identified.

<u>Containment Dome</u>
CISI of the containment dome from inside containment was performed during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

- At 30 degrees, 858' elevation, three areas of peeling paint were identified.
- At 40 degrees, 890' elevation, 8" x 12" area of cracked and stained paint were identified.
- At 290 degrees, 860' elevation, four 8" x 8" areas of missing and cracked paint with coating discoloration were identified.
- Various areas of discolored paint with cracking and missing paint were identified. The indications are located on upper portion of the dome above Containment Spray.

3.8.2.2 CISI on Concrete Containment – C1R19 - 2019

CISI of the concrete containment was performed during C1R19 (September 2019) for ASME Section XI IWL inspection requirements. The following observations were documented.

- OD Concrete – 707' AB, 270 - 90 deg. Concrete coatings were satisfactory except for two areas: (1) Az. 285° El. 707' has flaking and peeling paint. (2) Az. 293° El. 710' has peeling paint.

- OD Concrete – 712' FB. 90-270 deg. Satisfactory. No recordable indication.

- OD Concrete – Reactor Water Cleanup Mezz. 750' AB. 40 - 90 deg. Satisfactory. No recordable indication.

- OD Concrete – FB 737', 90 - 270 deg. Satisfactory. No recordable indication.

- OD Concrete – AB 737'. 270- 90 deg. Satisfactory. No recordable indication.

- OD Concrete – AB 762'. 270 - 90 deg. Concrete and coatings were satisfactory except for two areas: (1) Az. 290° El. 770' has flaking paint. (2) Az. 285° El. 785' has flaking paint.

- OD Concrete – FB 755', 90 -270 deg. Satisfactory. No recordable indication.

- OD Concrete – AB 781', and AB Steam Tunnel. Upper Aux Steam Tunnel concrete is satisfactory. Recordable indications are: (1) missing surface coating and coating damage between feedwater penetrations and around penetration 1MC-14, apparent moisture damage, and (2) flaking paint on Az. 80°, El. 780' and Az. 85°, El. 785'.

- OD Concrete – FB 781', 90 -270 deg. Satisfactory. No recordable indication.

- OD Concrete – Gas Control Boundary 800' and above AB and FB 0-360 deg. Satisfactory. No recordable indication.

3.8.2.3 CISI on Penetrations from Inside and Outside Containment – C1R19 - 2019

CISI of containment penetrations, from inside and outside containment, was performed during C1R19 (September 2019) for ASME Section XI IWE inspection requirements. The following observations were documented.

<u>From Inside Containment</u>

- Penetration IMC-202, 769' elevation, 82 degrees: flaking, peeling, coating damaged
- Penetration IMC-42, 756' elevation, 0 degrees: flaking, light rust, and coating damaged
- Penetration IMC-45, 756' elevation, 0 degrees: flaking, light rust, and coating damaged
- Penetration IMC-50, 790' elevation, 95 degrees: flaking, coating damaged, and discoloration
- Penetration IMC-114, 780' elevation, 51 degrees: peeling, blistering, and missing paint or coating

<u>From Outside Containment</u>

All containment penetrations from outside were inspected utilizing VT-3 inspectors. Several indications of minor significance were identified, such as paint or coating damage, flaking, light rust, missing paint, peeling, corrosion, cracked paint, tear, and discoloration. There was no apparent loss of material identified.

3.8.2.4 Containment Liner, IWE – Wetted Surfaces of Submerged Areas and BWR Vent System – C1R18 - 2018

In accordance with the 2nd 10-Year Inspection Interval the scheduled examinations for ASME Section XI, IWE, wetted surfaces of submerged areas and BWR vent system were completed during outage C1R18 (May 2018). This inspection was conducted for accessible areas in accordance with ER-AA-335-018, "Visual Examination of ASME IWE Class MC and Metallic Liners of IWL Class CC Components." All results were acceptable.

3.8.2.5 Conclusion

The above observations identified through CISI inspections on the containment liner, concrete containment, and containment penetrations (i.e., from inside and outside containment) were evaluated. There is no adverse effect on the metallic liner with respect to the design intent and overall structural integrity of the liner system. The conditions identified above are considered acceptable.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The proposed change 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 Power Reactors." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B and Type C testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that reviewed "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The one-time extension of the frequency of the containment Type A test will not affect the design, fabrication, or construction of the containment structure, and the design will continue to account for the effects of natural phenomena. The containment Type A test will continue to be done in accordance with 10 CFR 50 Appendix J using 10 CFR 50 Appendix B quality standards. The frequency of the containment Type A test is being changed in accordance with standards reviewed and approved as compliant with Appendix J. Therefore, there will be no instances where the applicable regulatory criteria are not met.

EGC has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements/criteria.

4.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed change would allow for a one-time extension to the 15-year frequency of the CPS Unit 1 containment leakage rate test (i.e., integrated leakage rate test (ILRT) or Type A test). This test is required by Technical Specifications (TS) Section 5.5.13, "Primary Containment Leakage Rate Testing Program." The proposed one-time change would permit the current ILRT interval of 15 years to be extended by eight months.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

   Response: No

   The proposed change involves changes to the CPS Unit 1 containment leakage rate testing program. The proposed change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself, and the testing requirements to periodically demonstrate the integrity of containment, exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators.

   The proposed change modifies TS 5.5.13 to allow for a one-time extension to the containment Type A test interval. The potential consequences of extending the containment Type A test interval one-time by eight months have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in the NRC Final Safety Evaluation for NEI Topical Report (TR) 94-01, Revision 3-A. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. EGC has determined that the increase in conditional containment failure probability due to the proposed change would be very small.

Therefore, the proposed change does 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 modifies TS 5.5.13 to allow for a one-time extension to the containment Type A test interval. Containment, and the testing requirements to periodically demonstrate the integrity of containment, exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

   Therefore, the proposed change does 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

   The proposed change modifies TS 5.5.13 to allow for a one-time extension to the containment Type A test interval. The proposed change does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the containment leakage rate testing program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A. Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that containment would not degrade in a manner that is not detectable by an ILRT. A risk assessment concluded that extending the ILRT test interval one-time by eight months results in a very small change to the risk profile.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, EGC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

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 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 the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. CPS USAR Figure 3.8-11, "Containment Building Penetrations"

2. IEEE 317, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," dated December 1976

3. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995

4. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," July 1995

5. NUREG-1493, "Performance-Based Containment Leak-Test Program," January 1995

6. EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994

7. NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," October 2008

8. NRC Regulatory Issue Summary 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50," dated December 8, 2008

9. NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," July 2012

10. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J' and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, 'Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals' (TAC No. MC9663)," dated June 25, 2008

11. 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 50, Appendix J (TAC No. ME2164)," dated June 8, 2012

12. Letter from D. Pickett (NRC) to M. Lyon (CPS), Clinton Power Station Unit 1 – Issuance of Amendment 105 Regarding Implementation of 10 CFR 50, Appendix J – Option B, (TAC NO. MB95321), dated June 21, 1996

13. Letter from J. Rankin (NRC) to B. C. Hanson (Exelon Generation Company, LLC), "Clinton Power Station, Unit No. 1 – Issuance of Amendment Regarding Permanent Extension of Type A and Type C Leak Rate Test Frequencies (CAC No. MF7290)," dated September 26, 2017

14. Electric Power Research Institute, EPRI TR-1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325," dated October 2008

15. NRC Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018

16. Letter from C. H. Cruse (Constellation Nuclear, Calvert Cliffs Nuclear Power Plant) to U.S. NRC, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002

17. Letter from J. J. Kowalski (Exelon Generation Company, LLC) to U.S. NRC, "Post Outage 90-Day Inservice Inspection (ISI) Summary Report," dated January 15, 2020

ATTACHMENT 2
Markup of Technical Specifications Page

Clinton Power Station, Unit 1

Facility Operating License No. NPF-62

REVISED TECHNICAL SPECIFICATIONS PAGE

5.0-16a

5.5 Programs and Manuals (continued)

5.5.13 Primary Containment Leakage Rate Testing Program

, (2) the next Type A test performed after the February 2008 Type A test shall be performed no later than October 31, 2023, and (3) if the Type A test has not been performed by October 31, 2023, and the unit is in Mode 4 or 5, the Type A test shall be performed prior to entering Mode 2

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. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following ~~exception~~ exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 9.0 psig.

The maximum allowable primary containment leakage rate L_a, at P_a, shall be 0.65% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is ≤ 5 scfh when tested at $\geq P_a$,

2) For each door, leakage rate is ≤ 5 scfh when the gap between door seals is pressurized $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

ATTACHMENT 3

Risk Assessment for CPS Regarding the ILRT (Type A) One-Time Extension Request

Risk Management Team

Clinton Power Station (CPS)

Risk Assessment for CPS Regarding the ILRT (Type A) One-Time Extension Request

CL-LAR-12
Revision 2

Exelon Generation

Risk Management Team

RM DOCUMENTATION NO.	**CL-LAR-12**	**REV: 2**	**PAGE NO. 1**

STATION: Clinton Power Station (CPS)

UNIT(s) AFFECTED: Unit 1

TITLE: Risk Assessment for CPS Regarding the ILRT (Type A) One-Time Extension Request

SUMMARY:
CPS is pursuing a License Amendment Request (LAR) for a one-time extension of the Integrated Leak Rate Test (ILRT) interval from 15 years to approximately 15.7 years.

The purpose of this document is to provide an assessment of the risk associated with implementing a one-time extension of the CPS Unit 1 containment ILRT interval to 15.7 years.

This is a Category I Risk Management Document in accordance with ER-AA-600-1012, which requires independent review and approval.

[ ] Review required after periodic update

[X] Internal RM Documentation **[ ] External RM Documentation**

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This RM documentation supersedes: N/A in its entirety.

	Print	Sign	Date
Prepared by:	Justin Sattler	see email approval	2/19/21
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Reviewed by:	Victoria Warren	see email approval	2/19/21
Approved by:	Eugene M. Kelly	see email approval	2/19/21

Email approvals are appended to the end of this document

Revisions:

REV.	DESCRIPTION	PREPARER/DATE	REVIEWER/DATE	APPROVER/DATE
0	Initial issue	see email approval	see email approval	see email approval
1	Incorporated editorial changes	see email approval	see email approval	see email approval
2	Incorporated editorial change	see email approval	see email approval	see email approval

TABLE OF CONTENTS

1.0 PURPOSE OF ANALYSIS

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Clinton Power Station (CPS) containment Type A integrated leak rate test (ILRT) interval by approximately 8 months, i.e., from 15 years to 15.7 years (a value of 15.7 years is used in this analysis). The risk assessment follows the guidelines from NEI 94-01 [1], the methodology outlined in EPRI TR-104285 [2], as updated by the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (EPRI TR-1018243) [3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change to a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage going undetected during the extended test interval [5]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the EPRI TR-1018243. Consistent with other previous ILRT extension requests for BWR Mark III containments [19, 36], the risk assessment also includes the impact of extending the Drywell Bypass Test (DWBT) interval from 15 years to 15.7 years on the ILRT risk metrics. The DWBT is in the scope of the surveillance frequency control program (SFCP); however, the DWBT has been historically associated with the ILRT frequency because the plant line-ups are similar, and the same equipment is used to perform both tests. Although not formally requested in this submittal, change to the DWBT could impact the ILRT dose metrics because bypass leakage above nominal through the drywell to the outer containment would bypass potential fission product scrubbing in the suppression pool that are released from the reactor into the drywell.

1.2 BACKGROUND

Option B to Appendix J Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 [6], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consists 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 EPRI Research Project TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" [2].

The NRC report on performance-based leak testing, NUREG-1493, 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 for a BWR plant that increasing the containment leak rate from the nominal 0.5% per day to 5% per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50% per day increases the total population exposure by less than 1%. Because ILRTs represent substantial resource expenditures, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures to support a reduction in the test frequency for CPS.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 [2] methodology to perform the risk assessment. In October 2008, EPRI 1018243 [3] was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily RG 1.174 [4]. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose. This ILRT interval extension risk assessment for CPS employs the EPRI 1018243 methodology, with the affected System, Structure, or Component (SSC) being the primary containment boundary. Similar methodology is used for this analysis as was used in the 2015/2016 CPS ILRT / DWBT permanent 15-year ILRT interval extension [36] that was approved by the NRC [37].

1.3 ACCEPTANCE CRITERIA

The acceptance guidelines in RG 1.174 [4] are used to assess the acceptability of this one-time 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 1.0E-06 per reactor year and increases in large early release frequency (LERF) less than 1.0E-07 per reactor year. Since the Type A test does not impact CDF for CPS (CPS does not credit containment accident pressure to maintain net positive suction head (NPSH) for pumps that take suction from the suppression pool; MAAP analysis performed for the permanent ILRT extension analysis demonstrates there is no impact to containment overpressure failure probability [36]), the relevant criterion is the change in LERF. RG 1.174 also defines “small” changes in LERF as below 1.0E-06 per reactor year, provided that the total LERF from all contributors (including external events) can be reasonably shown to be less than 1.0E-05 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) is also calculated to help ensure that the defense-in-depth philosophy is maintained.

With regard to population dose, examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G of [3]) indicate a range of incremental increases in population dose(1) that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (Figure 7-2 of NUREG-1493 [6]) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, the NRC SER on this issue [7] defines a “small” increase in population dose as an increase of ≤1.0 person-rem per year, or ≤1 % of the total population dose,

(1) The one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35La, whereas this analysis uses 100La.

whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This definition has been adopted for the CPS analysis.

The acceptance criteria are summarized below.

1) The estimated risk increase associated with temporarily extending the ILRT surveillance interval to 15.7 years must be demonstrated to be “small.” (Note that Regulatory Guide 1.174 [4] defines “very small” changes in risk as increases in CDF less than 1.0E-6 per reactor year and increases in LERF less than 1.0E-7 per reactor year. Since the type A ILRT does not impact CDF for CPS, the relevant risk metric is the change in LERF. Regulatory Guide 1.174 also defines “small” risk increase as a change in LERF of less than 1.0E-6 reactor year.) Therefore, a small change in risk for this application is defined as a LERF increase of less than 1.0E-6.
2) The increase in population dose must be “small.” Per the NRC SER, a small increase in population dose is also 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.
3) In addition, the SER notes that a small increase in Conditional Containment Failure Probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests (typically about 1% or less, with the largest increase being 1.2%). This requires the increase in CCFP be less than or equal to 1.5% (i.e., marginally greater than 1.2%).

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI methodology [3] is used for evaluating the one-time change in risk associated with increasing the test interval to 15.7 years. The analysis uses results from the core damage and large early release scenarios from the current CPS PRA analyses of record [24, 25] and the subsequent containment responses to establish the various fission product release categories including the release size.

The six 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 [3].
2. Develop plant-specific population dose rates (person-rem 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 from 15 years to 15.7 years.
4. Determine the change in risk in terms of LERF in accordance with RG 1.174 and compare this change with the acceptance guidelines of RG 1.174 [4].
5. Determine the impact on the CCFP.
6. Evaluate the impact of external events.

Furthermore,

- Consistent with the previous industry containment leak risk assessments, the CPS assessment uses population dose as one of the risk measures. The other risk measures used in the CPS assessment are the conditional CCFP for defense-in-depth considerations and change in LERF to demonstrate that the acceptance guidelines from RG 1.174 are met.

This evaluation for CPS uses ground rules and methods to calculate changes in the above risk metrics that are consistent with those outlined in the current EPRI methodology [3].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The technical adequacy of the CPS Level 1 and Level 2 full power internal events (FPIE) PRA models is consistent with the requirements of Regulatory Guide 1.200 as relevant to this ILRT interval extension. See Appendix A for additional information.
- The CPS Level 1 and Level 2 FPIE PRA models provide representative core damage frequency and release category frequency distributions to be used in this analysis.
- It is appropriate to use the CPS FPIE PRA model as a gauge 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) would not substantially differ if other hazard groups were included in the calculations. An analysis is performed in Section 5.7 to show the effect of including external event models for the ILRT extension. The Seismic CDF and LERF estimates [38] and Fire PRA [8] are used for this analysis.
- Dose results for the containment failures modeled in the FPIE PRA can be characterized by information provided in NUREG/CR-4551 [17]. They are estimated by scaling the NUREG/CR-4551 population dose results by power level, population, and Tech Spec leak rate differences for Clinton Power Station compared to the NUREG/CR-4551 Mark III reference plant, Grand Gulf.
- The use of the estimated 2030 population data from SECPOP version 4.2 [31], the Illinois Department of Public Health (IDPH) [30], and 2019 US Census Bureau statistics [10] are appropriate for this analysis.
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a is $10L_a$ and for Class 3b sequences is $100L_a$, based on the recommendations in the latest EPRI report [3] and as recommended in the NRC SER on this topic [7]. It should be noted that this is more conservative than the earlier previous industry ILRT extension requests, which used $35L_a$ for the Class 3b sequences.
- Based on the EPRI methodology and the NRC SER, the Class 3b sequences are categorized as LERF and the increase in Class 3b frequencies is used as a surrogate for the ΔLERF metric.

- 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.

4.0 INPUTS

This section summarizes the following:

- Section 4.1 General Resources Available as input
- Section 4.2 Plant Specific Resources Required
- Section 4.3 CPS Population Dose Derivation
- Section 4.4 Details on the EPRI Methodology that is followed
- Section 4.5 Details of the Calvert Cliffs corrosion analysis method that is also used as a sensitivity for this assessment
- Section 4.6 Details of the analysis performed on the available Mark III DWBT data to estimate the likelihood and magnitude of DWBT leakage rates that may occur due to extending the DWBT interval in addition to the ILRT interval (note: extending the DWBT interval is not part of the proposed LAR scope).

4.1 GENERAL RESOURCES AVAILABLE

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

- NUREG/CR-3539 [13]

  This study, which also used information from WASH-1400 [22], 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.

- NUREG/CR-4220 [14]

  This study provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident.

- NUREG-1273 [15]

  This is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database.

- NUREG/CR-4330 [16]

  This study provides an assessment of the impact of different containment leakage rates on plant risk.

- EPRI TR-105189 [12]

  This study provides an assessment of the impact on shutdown risk from ILRT test interval extension.

- NUREG-1493 [6]

  This 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.

- EPRI TR-104285 [2]

  This study is an EPRI study of the impact of extending ILRT and local leak rate test (LLRT) intervals on at-power public risk.

- NUREG-1150 [23] and NUREG/CR 4551 [29]

  These two reports provide an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the basis for the consequence analysis of the ILRT interval extension for CNP.

- NEI Interim Guidance [28, 35]

  This 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.

- Calvert Cliffs corrosion analysis [5]

  This study addresses the impact of age-related degradation of the containment liners on ILRT evaluations.

- EPRI 1018243 [3]

  This study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

- NRC Final Safety Evaluation Report [7]

  This report documents the NRC staff's evaluation and acceptance of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2.

- Prior Mark III ILRT Extension Risk Assessments [19, 20, 21]

  Reference is made to other extension requests for Mark III containments that considered extensions to the ILRT interval. While the DWBT interval is not being extended as part of this analysis, accounting for the status of the drywell in the ILRT extension PRA is a surrogate for the impact to the population dose associated with an intact containment but with a suppression pool bypass via the drywell leakage.

The DWBT has been historically associated with the ILRT frequency because the plant line-ups are similar and the same equipment is used to perform both tests. The DWBT is to verify that pre-existing drywell bypass leakage does not exceed the maximum requirements. The DWBT thus affects the likelihood of a suppression pool bypass in the Level 1 and 2 PRA analyses. The methodology for extending the DWBT has previously been accepted by the NRC for analysis of Clinton, Grand Gulf, and River Bend [19, 20, 21].

Even though the methodologies used for the ILRT extension do not directly address the DWBT, it is judged that the ILRT methodology can be used to address the impact of extending both the ILRT and DWBT with a few additional considerations and assumptions. The primary difference in the methodology used to evaluate the extension of the DWBT is in the determination of the conditional probability of an existing drywell leak. In the base case DWBT analysis, the same release categories, consequence calculations, and acceptance criteria are used as in the ILRT analysis. The risk analysis will be performed assuming that both the ILRT and the DWBT are on the same frequencies. The impact of drywell leakage is to allow drywell atmosphere, including fission products, to be passed at some rate directly to the containment, without benefit of quenching and fission product retention in the suppression pool.

It is assumed in this methodology that the special leakage categories established by EPRI for use in ILRT risk assessments can also be applied to the drywell for the DWBT risk assessment. The Mark III drywell, which includes the RPV, is completely enclosed by the outer containment. As such, the drywell leakage does not leak directly to the environment but is further mitigated by the outer containment leakage barrier. Because of this “dual” containment, there are several possible leakage path combinations that must be considered. The drywell can be intact (base leakage assumed), it can have a small pre-existing failure (10 times base leakage using the EPRI ILRT assumption), or it can have a large pre-existing failure (100 times base leakage using the EPRI ILRT assumption). As discussed in Section 4.6.1, this leads to nine combinations of drywell and containment leakage sizes (refer to Figure 4.1-1). Each combination has an impact on radionuclide releases that corresponds to one of the containment failure categories.

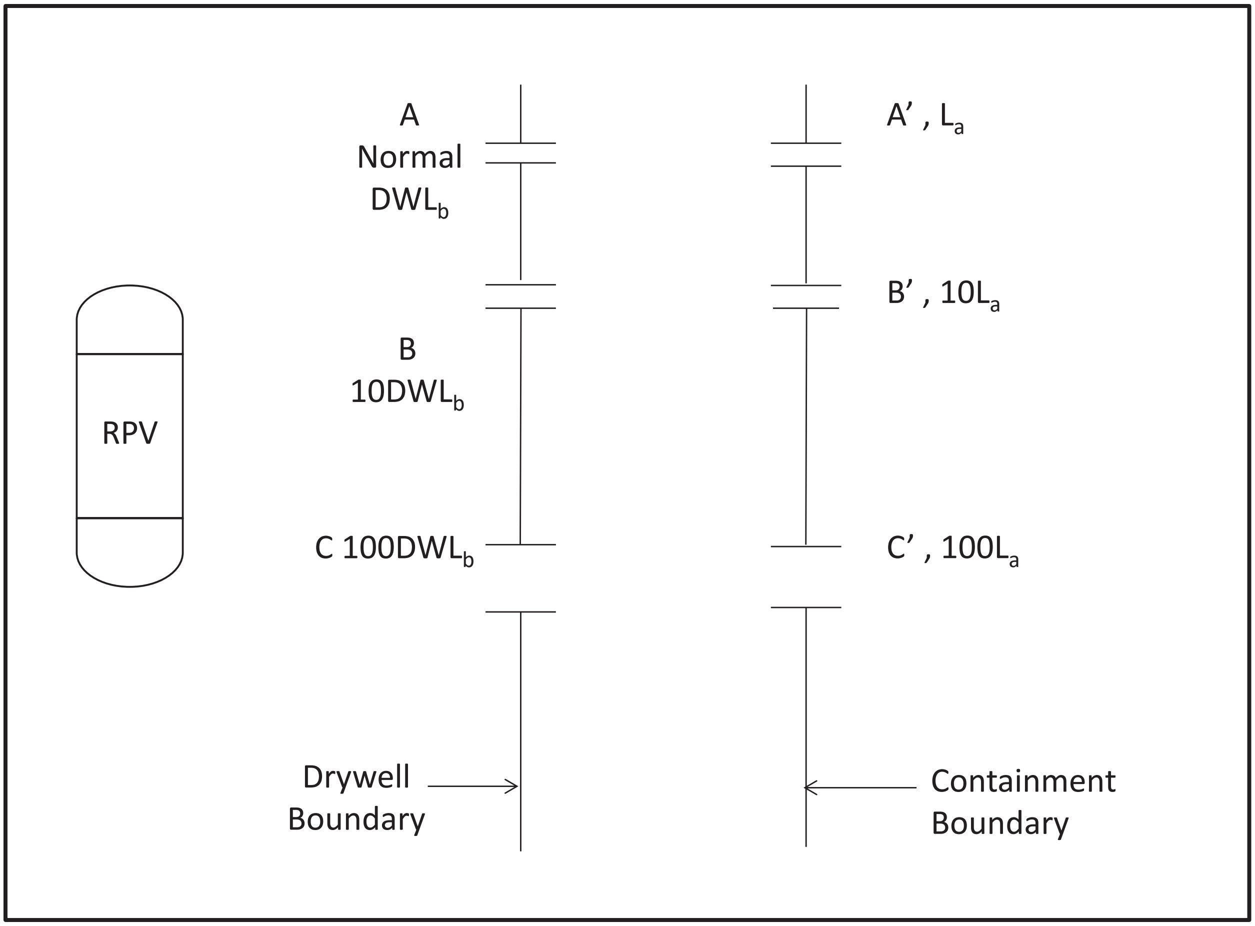


FIGURE 4.1-1
CPS DRYWELL AND CONTAINMENT LEAKAGE CATEGORIES

The Mark III and CPS plant-specific data used for the DWBT portion of this risk assessment is provided in Section 4.6.

4.2 PLANT SPECIFIC INPUTS

The CPS specific information used to perform this ILRT interval extension risk assessment includes the following:

- FPIE PRA model Level 1 and LERF quantification results [24, 25]
- Population within a 50-mile radius (see Section 4.3)
- Reactor Power Level [18]
- Allowable Containment Leakage [18]

<u>CPS FPIE Core Damage Frequency</u>

The current CPS FPIE PRA analysis of record is an event tree/linked fault tree model characteristic of the as-built, as-operated plant. Based on the subsumed merged sequence cutset file results reported in the CPS PRA Summary Notebook for the 2017B PRA Update (CL117B) completed in 2020 [24], the mean value of the internal events core damage frequency (CDF) is 3.33E-06/yr (truncation limit 5E-13/yr). Core Damage Frequency by Class is provided in Table 4.2-1.

<u>Release Category Definitions</u>

Table 4.2-2 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [3]. 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 5.0 of this report.

TABLE 4.2-1
2017 CLINTON LEVEL 1 CDF RESULTS[3]

CLASS	DESCRIPTION	CDF (/YR)	% OF CDF[1]
IA/IE	Loss of Makeup at High RPV Pressure (Transient Initiators)	6.34E-07	19.05%
IBE	Early Station Blackout (less than 4 hours)	5.17E-07	15.52%
IBL	Late Station Blackout (greater than 4 hours)	1.03E-06	30.95%
IC	ATWS-Induced LOCA (Containment Intact)	7.76E-09	0.23%
ID	Loss of Makeup at Low RPV Pressure (Transient Initiators)	1.37E-07	4.11%
IIA	Loss of Containment Heat Removal (RPV initially Intact)	6.81E-07	20.46%
IIL	Loss of Containment Heat Removal (RPV breached but no initial core damage)	1.50E-09	0.05%
IIT	Loss of Containment Heat Removal (RPV initially Intact; core damage induced post high containment pressure)	1.32E-08	0.40%
IIV	Class IIA and IIL except that the vent operates as designed	9.50E-08	2.85%
IIIA	Excessive LOCA	9.66E-10	0.03%
IIIB	Loss of Makeup at High RPV Pressure (LOCA Initiators)	2.33E-09	0.07%
IIIC	Loss of Makeup at Low RPV Pressure (LOCA Initiators)	1.27E-08	0.38%
IIID	Loss of Vapor Suppression (LOCA Initiators)	4.46E-09	0.13%
IV[2]	Loss of Adequate Reactivity Control (ATWS)	1.53E-07	4.61%
V	Containment Bypass	3.88E-08	1.17%
Total	-	3.33E-06	100.0%

Notes to Table 4.2-1:

(1) Level 1 model results used as input to Level 2 update (based on 5E-13/yr truncation frequency).

(2) Classes IVA and IVL included in Class IV.

(3) From Table 3.3-2 of The CPS PRA Summary Notebook [24]. The update is referred to as the “2017 update” but reflects information incorporated in 2020.

TABLE 4.2-2
EPRI [2] /NEI CONTAINMENT FAILURE CLASSIFICATIONS

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 L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is 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.

CPS Internal Events Release Category Frequencies

The CPS PRA Combined Level 1/Level 2 Model [25] is used to develop the initial set of internal events release categories for use in this analysis. Table 4.2-3 summarizes the pertinent CPS results in terms of release category. The OK release (i.e., intact containment limited to normal leakage) is 9.61E-7/yr. The total LERF, which corresponds to the High/Early (H/E) release category in Table 4.2-3, was calculated to be 1.65E-7/yr. While the total release frequency shown in Table 4.2-3 is 2.43E-6, the total release frequency used in this ILRT extension analysis is 2.37E-6/yr, which is calculated by subtracting the OK release from total CDF (3.33E-6/yr - 9.61E-7/yr = 2.37E-6/yr). While the total Level 2 frequency includes some double counting of release categories that leads to a slight overcounting, the OK release, by definition, is only one release category so its frequency is preserved.

Based on the Level 1 and LERF PRA model results described above, Table 4.2-4 lists the relevant EPRI release category frequencies pertinent for the ILRT extension risk assessment, including the delineation of LERF and non-LERF frequencies for Class 7.

The non-LERF frequencies for Class 7 are determined by reviewing 128 top L2 release sequences, which constitute more than 99% of all L2 release frequency. The sequences that did not end in a LERF endstate or did not include containment isolation failures (Class 2) were identified as Class 7 non-LERF contributors.

TABLE 4.2-3
SUMMARY OF CONTAINMENT EVALUATION[1]

INPUT		*OUTPUT*	
LEVEL 1 PSA		**CET EVALUATION**	
CORE DAMAGE FREQUENCY	**CHARACTERIZE RELEASE**	**RELEASE BIN[3]**	**RELEASE FREQUENCY (PER YEAR)**
3.33E-6/year	Little or No Release (Intact)	OK	9.61E-07
	Low Public Risk Impact	LL & Late	5.09E-08
		LL & I	ε[2]
		LL & E	2.83E-09
		L & Late	3.08E-07
		L & I	ε[2]
		L & E	4.10E-07
	Moderate Release	M & Late	5.51E-07
		M & I	ε[2]
		M & E	1.94E-07
	High Release	H & Late	7.46E-07
		H & I	ε[2]
		H & E	1.65E-07

Notes to Table 4.2-3:

(1) From Table 3.4-6B of The CPS PRA Summary Notebook [24].

(2) ε = negligibly small number

(3) The Release Bin nomenclature is the following:

First Designator (Radionuclide release type)

1) High (H) - A radionuclide release of sufficient magnitude to have the potential to cause prompt fatalities.
2) Medium or Moderate (M) - A radionuclide release of sufficient magnitude to cause near-term health effects.
3) Low (L) - A radionuclide release with the potential for latent health effects.
4) Low-Low (LL) - A radionuclide release with undetectable or minor health effects.
5) Negligible (OK) - A radionuclide release that is less than or equal to the containment design base leakage.

Second Designator (Timing)

1)	Early (E)	Less than time when evacuation is effective (i.e., 4 hours)
2)	Intermediate (I)	Greater than or equal to 4 hours but less than 24 hours
3)	Late (L)	Greater than or equal to 24 hours.

TABLE 4.2-4
RELEVANT LEVEL 2 RELEASE CATEGORY FREQUENCIES FOR CPS

EPRI RELEASE CATEGORY	FREQUENCY/YR	SOURCE
1: No Containment Failure	9.61E-07	Table 4.2-2
2: Containment Isolation Failure[1]	4.39E-07	Level 2 contribution of sequences (top 99.5%) that include failure of the containment isolation function (Event Tree Node IS=F)
7: Phenomena-induced containment failures (LERF)	1.27E-07	Table 4.2-2 LERF (H/E) – Containment Bypass (Class V)
7: Phenomena-induced containment failures (non-LERF)	3.33E-06 - 9.61E-07 - 1.65E-07 <u>- 4.39E-07</u> = 1.76E-06	Table 4.2-2 and sequence evaluation CDF – OK (EPRI Class 1) – LERF (EPRI Class 7 LERF & 8) – Cont. Isolation (EPRI Class 2)
8: Containment Bypass	3.88E-08	Class V (ISLOCA + BOC Sequences)
Total:	**3.33E-06**	

<u>Note to Table 4.2-4:</u>

(1) Not all Containment Isolation Failure sequences are found to be H/E (LERF) in the CPS PRA based on MAAP calculations. Nevertheless, EPRI Class 2 is conservatively assigned to Bin 1, which is the largest person-rem assignment (see Table 4.3-3). This category does not have a significant impact on results and only affects the total dose and % change metrics.

4.3 CPS POPULATION DOSE DERIVATION

Since CPS does not maintain a detailed Level 3 PRA model, the approach recommended in EPRI 1018243 [3] is used. For the cases where plant-specific PRA dose information is not available, a representative population dose can be calculated using other references, such as NUREG/CR-4551 [17]. This approach was taken for the 2003 CPS ILRT / DWBT one-time extension [19] that was approved by the NRC [11] and the 2015/2016 CPS ILRT / DWBT permanent extension [36] that was approved by the NRC [37]. To develop a representative population dose, the NUREG/CR-4551 plant that most closely resembles the analysis plant is chosen and the following steps are performed.

- Relate the NUREG/CR-4551 accident progression bins (APBs), EPRI Accident Classes, and plant-specific plant damage states (PDSs) based on the definitions contained in NUREG/CR-4551, and plant-specific PDSs.
- Adjust the resulting EPRI Accident Class 1, 2, 7, and 8 population doses to account for substantial differences in reactor power level, population density, allowable containment leak rate (L_a), and other plant-specific factors that may affect population dose as follows:
  - Population density adjustment = (population within 50 miles of the CPS ÷ population within 50 miles of the NUREG/CR-4551 reference plant)
  - Power level adjustment = (rated power level of CPS (MWt) ÷ rated power level of reference plant)
  - L_a adjustment = L_a of CPS (%wt/day) ÷ L_a of reference plant

Note that the population density and power level adjustments are applicable to all EPRI accident classes; however, the L_a adjustment should be made only to intact containment end states.

<u>Reference Plant Population Dose Information</u>

Consistent with the EPRI guidance [3], the ex-plant consequence analysis for Grand Gulf is used as the reference plant for CPS since Grand Gulf is also a BWR Mark III containment. Table 4.3-1 reproduces the APB descriptions for Grand Gulf provided in Section 2.4.2 of NUREG/CR-4551 [17], and Table 4.3-2 provides a calculation to determine the relevant population dose associated with each APB. Note that Table 4.3-2 is consistent with the calculations previously performed for the Clinton ILRT interval extension submittals [19, 36].

TABLE 4.3-1
COLLAPSED ACCIDENT PROGRESSION BIN DESCRIPTIONS FOR GRAND GULF [17]

COLLAPSED APB NUMBER	DESCRIPTION
1	CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.
2	CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.
3	CD, vessel breach, Early CF, Late SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail until the late time period and, thus, both the in-vessel releases and the releases associated with vessel breach are scrubbed by the suppression pool. Therefore, the availability of containment sprays during the time period that the suppression pool is not bypassed is not very important and, thus, the CS characteristic has been dropped.
4	CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.
5	CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.
6	CD, vessel breach, Vent This summary bin represents the case in which vessel breach occurs and the containment was vented during any of the time periods in the accident.
7	CD, VB, No CF Vessel breach occurs but there is no containment failure and any releases associated with normal containment leakage are minor. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped. The risk associated with this bin will be negligible.
8	CD, No vessel breach Vessel breach is averted. Thus, there are no releases associated with vessel breach and there are no CCI releases. It must be remembered, however, that the containment can fail even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment. It follows that there will be some risk associated with this bin.

Legend for Table 4.3-1:

CCI = Core Concrete Interaction
CD = Core Damage
CF = Containment Failure
CS = Containment Sprays
SP = Suppression Pool
VB = Vessel Breach

TABLE 4.3-2
GRAND GULF NUREG/CR-4551 [17] 50-MILE RADIUS POPULATION DOSE

APB #(1)	APB DEFINITION	APB FREQUENCY (PER YEAR)(2)	APB FRACTIONAL CONTRIBUTION TO 50-MILE RADIUS TOTAL DOSE RISK(3)	APB 50-MILE RADIUS DOSE RISK (PERSON-REM/YEAR)(4)	APB 50-MILE RADIUS DOSE (PERSON-REM)(5)
1	CD, VB, Early CF, Early SP Bypass, CS Not Available	6.46E-7	.268	0.139	2.15E+5
2	CD, VB, Early CF, Early SP Bypass, CS Available	2.00E-7	.056	0.029	1.45E+5
3	CD, VB, Early CF, Late SP Bypass	2.86E-8	.011	5.7E-3	1.99E+5
4	CD, VB, Early CF, No SP Bypass	8.92E-7	.267	0.139	1.56E+5
5	CD, VB, Late CF	1.16E-6	.281	0.146	1.26E+5
6	CD, VB, Vent	1.55E-7	.039	0.0203	1.31E+5
7	CD, VB, No CF	2.05E-7	3E-4	1.56E-4	7.61E+2
8	CD, No VB	7.36E-7	.077	0.040	5.43E+4
Total		**4.09E-6**	**1.0**	**0.52**	**--**

Notes to Table 4.3-2:

(1) This table is presented in the form of a calculation because NUREG/CR-4551 [17] does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose risk results in NUREG/CR-4551.

(2) The total (i.e., internal accident sequences) CDF of 4.09E-6/yr and the CDF subtotals by APB are taken from Figure 2.5-7 of NUREG/CR-4551 Vol. 6 Rev.1 Part 1.

(3) The individual APB contributions to total (i.e., internal accident sequences) 50-mile radius dose rate are taken from Table 5.1-3 of NUREG/CR-4551 Vol. 6 Rev.1 Part 1.

(4) The APB 50-mile dose risk is calculated by multiplying the individual APB dose risk contributions (column 4) by the total mean 50-mile radius dose risk of 0.52 person-rem/yr (taken from Table 5.1-1 of NUREG/CR-4551 Vol. 6 Rev.1 Part 1).

(5) The individual APB doses are calculated by dividing the individual APB dose risk by the APB frequencies.

Legend for Table 4.3-2:

CCI = Core Concrete Interaction
CD = Core Damage
CF = Containment Failure
CS = Containment Sprays
SP = Suppression Pool
VB = Vessel Breach

The APBs described above are assigned to one of the EPRI release categories for the CPS assessment. These assignments and their basis are provided in Table 4.3-3. It is noted that no assignment of NUREG/CR-4551 APBs is made for EPRI Release Categories 3a and 3b because, per the EPRI methodology, these doses are calculated using factors of 10 and 100, respectively, of the population dose for EPRI Category 1. Also, EPRI Categories 4, 5, and 6 are not affected by the ILRT frequency and are therefore (per the EPRI guidance) not included in the assignment process.

TABLE 4.3-3
ASSIGNED APB FOR EACH OF THE RELEVANT LEVEL 2 RELEASE CATEGORIES FOR CPS

EPRI RELEASE CATEGORY	ASSIGNED APB	BASIS
1: No Containment Failure	7	The intact containment case with release limited to leakage is represented by APB 7 in the Grand Gulf assessment.
2: Containment Isolation Failure	1	APB 1 is conservatively chosen since the drywell may fail or the Suppression Pool bypassed, leading to an early release scenario. APB 1 results in the highest dose of all the Grand Gulf containment failure APBs (which is indicative of a containment failure with suppression pool or drywell bypass)
7: Phenomena-induced containment failures (LERF)	1	APB 1 w/o containment sprays available is chosen for CPS. APB 1 results in the highest dose of all the Grand Gulf containment failure APBs (which is indicative of LERF)
7: Phenomena-induced containment failures (non-LERF)	1, 3, 4, 5, 6, and 8	For CPS, this release category has both early and late releases. The sequences are binned accordingly, as presented in more detail in Table 4.3-7.
8: Containment Bypass	1	The containment bypass case is selected as APB 1 from the Grand Gulf assessment. APB 1 results in the highest dose of all the Grand Gulf containment failure APBs (which is indicative of containment bypass)

Adjustments to Ex-Plant Consequence Calculations

The next step per the EPRI guidance is to adjust the resulting EPRI Accident Class 1, 2, 7, and 8 population doses from the reference plant to account for differences in reactor power level, population, and allowable containment leak rate (L_a).

The 50-mile radius population used in the Grand Gulf NUREG/CR-4551 consequence calculations is 3.25E+5 persons. This is based on 1980 Census data as documented in NUREG/CR-4551 Vol. 2, Rev. 1, Part 7 [33] Appendix A.3.

TABLE 4.3-4
NUREG/CR-4551 GRAND GULF POPULATION

DISTANCE FROM PLANT		
(KM)	(MILES)	POPULATION
1.6	1.0	34
4.8	3.0	879
16.1	10.0	10,255
32.2	20.0	28,151
48.3	30.0	97,395
64.37	40.0	192,677
80.47	50.0	325,285

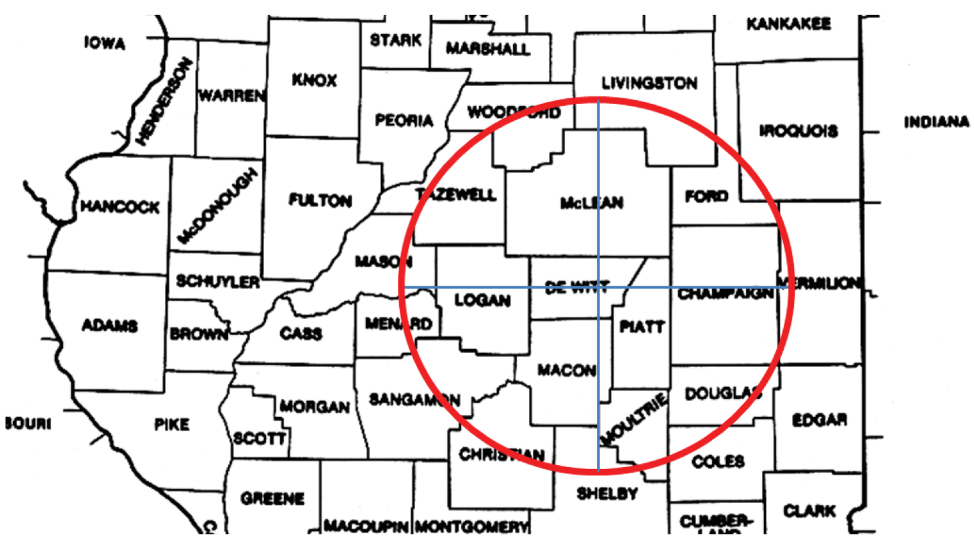
The 50-mile radius population dose for CPS is based on the 2030 population estimate projection using SECPOP 4.2 population data [31] for 2000 and 2010 and assuming the population growth rate from 2000 to 2010 continues for the next two decades. The SECPOP 4.2 code uses census data to calculate population counts for user defined sector segments. The EPRI methodology does not specify population projection to a future date. For this risk assessment, the CPS population was conservatively projected to the year 2030 to represent potential future average population density increases (although current trends indicate a population decrease).

TABLE 4.3-5
SECPOP CODE POPULATION ESTIMATES

RADIUS	SECPOP			
	2000	2010	2020(1)(2)	2030(1)(2)
0 – 10	12,334	12,219	12,105	11,992
0 – 20	57,626	61,143	64,875	68,834
0 – 30	351,649	374,458	398,746	424,610
0 – 40	538,318	574,660	613,455	654,870
0 – 50	768,541	813,071	860,181	910,021

(1) The 2020 and 2030 estimates are made assuming the 5.79% population increase experienced in the 50-mile radius region during the decade from July 2000 to July 2010 continues to occur each of the next two decades.

(2) The Illinois Department of Public Health (IDPH) growth projections [30] from 2010 to 2025 for the counties with areas within the 50 mile radius of CPS are shown in Table 4.3-5B below. The combined county growth rate for these counties is 3.7% for the 2010 to 2025 year period. The IDPH the population projection of 3.7% for a 15 year period demonstrates that the SECPOP 4.2 based growth rate of 5.79% per decade is conservative (leading to a higher CPS dose projection). Additionally, 2019 population data [10] was examined to determine the population change since 2010. Population decreased more than 2% from 2010 to 2019, as shown in Table 4.3-5B. Therefore, the dose calculation methodology is corroborated to be conservative. See map and table below.



Illinois Counties Within 50 Miles of CPS Shown in Map Above

TABLE 4.3-5B
POPULATION PROJECTION ESTIMATES

COUNTY	AREA WITHIN 50 MILE RADIUS	2010 POPULATION - IDPH [30]	2020 POPULATION ESTIMATE - IDPH [30]	2025 POPULATION ESTIMATE - IDPH [30]	2019 POPULATION - US CENSUS BUREAU [10]
Champaign	99%	201,370	217,735	225,626	209,689
Christian	40%	34,804	33,152	32,345	32,304
Coles	20%	53,945	56,851	58,405	50,621
De Witt	100%	16,583	15,832	15,495	15,638
Douglas	75%	19,976	19,767	19,709	19,465
Ford	75%	14,074	13,450	13,244	12,961
Iroquois	<5%	29,657	27,687	26,816	27,114
Livingston	40%	38,882	39,390	39,596	35,648
Logan	100%	30,272	30,380	30,441	28,618
Macon	100%	110,757	105,401	103,126	104,009
Mason	25%	14,627	12,841	12,074	13,359
McLean	100%	169,838	188,341	197,855	171,517
Menard	40%	12,708	12,867	12,913	12,196
Moultrie	90%	14,846	14,715	14,706	14,501
Piatt	100%	16,722	16,205	16,000	16,344
Sangamon	35%	197,822	203,501	207,194	194,672
Shelby	25%	22,339	21,496	21,118	21,634
Tazewell	75%	135,439	136,051	136,436	131,803
Vermillion	<5%	81,588	77,965	76,441	75,758
Woodford	50%	38,664	40,350	41,360	38,459
Total	--	1,255,013	1,283,977	1,300,900(1)(2)	1,226,310(3)

Notes to Table 4.3-5B:

(1) This total is used to determine average county growth over 15-year period. Since many counties have population outside the 50-mile radius, the population within 50 miles of CPS is significantly less than the totals shown above.

(2) IDPH Projected Population Increase from 2010-2025 =

$$(1{,}300{,}900 \div 1{,}255{,}013 - 1) \div 100\% = 3.66\%$$

(3) Actual population change from 2010-2019 = $(1{,}226{,}310 \div 1{,}255{,}013 - 1) \div 100\% = -2.29\%$

The ratio of the population surrounding CPS (Table 4.3-5), to that in the Grand Gulf analysis results in a factor increase of:

9.10E+5 persons / 3.25E+5 persons = 2.80

The Grand Gulf reactor power level used in the NUREG/CR-4551 consequence calculations is 3833 MWt [33]. The current CPS reactor power level is 3473 MWt [18]. Therefore, the ratio of the CPS reactor power to that used in the Grand Gulf analysis results in a multiplication factor of:

3473 MWt / 3833 MWt = 0.91

The containment leakage used in the NUREG/CR-4551 consequence calculations for Grand Gulf is 0.5 %wt/day [29]. The current CPS allowable leakage is 0.65 %wt/day [18]. Because the leakage rates are a function of the containment volume, these plant characteristics are also needed:

- Grand Gulf Containment Volume [29] = 1.40E+6 ft^3
- CPS Containment Volume [18] = 1.51E+6 ft^3

Therefore, the ratio of the CPS allowable leakage and containment volume to that used in the Grand Gulf analysis results in a multiplication factor of:

(0.65% * 1.51E+6) / (0.5% * 1.40E+6) = 1.40

As stated previously, this final adjustment factor is only applied to the intact containment case. Table 4.3-6 provides a summary of each of the adjustment factors used for each APB to estimate the population doses for CPS that can be used in this assessment.

TABLE 4.3-6
CPS ADJUSTED 50-MILE RADIUS POPULATION DOSE

APB #	GRAND GULF 50-MILE RADIUS DOSE (PERSON-REM)[1]	POPULATION ADJUSTMENT FACTOR	REACTOR POWER ADJUSTMENT FACTOR	CONTAINMENT LEAK RATE ADJUSTMENT FACTOR	CPS POPULATION DOSE ADJUSTED 50-MILE RADIUS DOSE (PERSON-REM)
1	2.15E+05	2.8	0.91	N/A	5.48E+05
2	1.45E+05	2.8	0.91	N/A	3.69E+05
3	1.99E+05	2.8	0.91	N/A	5.07E+05
4	1.56E+05	2.8	0.91	N/A	3.97E+05
5	1.26E+05	2.8	0.91	N/A	3.21E+05
6	1.31E+05	2.8	0.91	N/A	3.34E+05
7	7.61E+02	2.8	0.91	1.40	2.71E+03
8	5.43E+04	2.8	0.91	N/A	1.38E+05

Note to Table 4.3-6:

(1) The NUREG/CR-4551 evaluation of Grand Gulf is used as input to the assessment of population dose for CPS. Refer to Table 4.3-2.

Population Dose Risk Calculations

The next step is to take the frequency information from Table 4.2-4 for each relevant EPRI release category class in Table 4.1-1 and associate a representative population dose from Table 4.3-6 for each release category based on the APB assignments made in Table 4.3-3. As discussed in more detail below, EPRI class 7 is further refined based on the CPS Level 2 PRA, as identified in Table 4.3-7, and allocated frequency and APB doses as identified in Table 4.3-8. Table 4.3-9 lists the population dose risk organized by EPRI release category for CPS, including the delineation of LERF and non-LERF frequencies for Class 7. Note that the population dose risk (Column 4 of Table 4.3-8 and Table 4.3-9) was found by multiplying the release category frequency (Column 2 of Table 4.3-8 and Table 4.3-9) by the associated population dose (Column 3 of Table 4.3-8 and Table 4.3-9). Also note that only the applicable EPRI release categories at this point are shown in the tables (i.e., the Class 3 frequencies are derived later and the Class 4, 5, and 6 frequencies are not used in the EPRI methodology for the ILRT extension risk assessment).

Application of Clinton PRA Model Results to NUREG/CR-4551 Dose Results

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the current Clinton PRA Level 2 model are categorized by accident classes that differ from the NUREG/CR-4551 APB classification scheme. Therefore, an assignment process is required to apply the NUREG/CR-4551 dose results. This subsection provides a description of the process used.

The basic process used was to review 128 top sequences of the Clinton Level 2 model (which provide more than 99.4% of Level 2 release frequency) and to assign each sequence into one of the collapsed APBs from NUREG/CR-4551. The CPS Level 2 model (i.e., containment event tree structure) contains significantly more information about the accident sequences than in the collapsed APBs in NUREG/CR-4551, and this assignment process required simplification of CPS accident progression information and assumptions related to categorizations of certain items. The relevant assumptions used for these assignments are summarized in Table 4.3-7. Other containment event tree nodes are included in the Clinton Level 2 model, but these were not used in (or did not contribute to) to the APB assignment performed here for the ILRT assessment. Additionally, these bin assignments are all related to EPRI Class 7 and therefore influence the total base case population dose estimated for CPS but do not influence the change in dose calculated for the ILRT extension risk assessment.

Class 7 Sequences Dose Risk Adjustments

EPRI Class 7 consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs. For this analysis, the associated radionuclide releases are based on the application of the Level 2 containment end states to the APBs from NUREG/CR-4551 as described in Section 4.2. The Class 7 Sequences are divided into two categories. Class “7 LERF” is defined to consist of LERF sequences (excluding the Break Outside Containment (BOC) and Interfacing Systems LOCA (ISLOCA) Class V sequences) and Class “7 non-LERF” is defined to consist of non-LERF sequences. The second category (non-LERF) is further divided into Class 7a, 7b, 7c, 7d, 7e, and 7f for assignments of NUREG/CR-4551 APB Bins 1, 3, 4, 5, 6, and 8. The failure frequency and population dose for each specific APB is shown in Table 4.3-8. The total release frequency and total dose for the Non-LERF Accident Class 7a, 7b, 7c, 7d, 7e, and 7f are then used to determine a weighted average person-rem for use as the representative EPRI Class 7 non-LERF dose in the subsequent analysis. Note that the total frequency and dose associated from this EPRI class does not change based on the ILRT interval, since Class 7 involves containment failure.

TABLE 4.3-7
CPS LEVEL 2 NODAL ASSUMPTIONS FOR APB ASSIGNMENTS

ACCIDENT CLASS	CPS PRA CONTAINMENT EVENT TREE NODE	ASSUMPTION
1 and 3	RX – Core Melt Arrested in Vessel	A success at this node signifies that there is no vessel breach. The sequences following this path are generally grouped in APB 8. However, there are cases in which Engertic Containment Failure (CX) occur. In those cases, these scenairios are assumed to result in a high early release and are categorized as APB 1. Additionally Supp. Pool Failure below the water line (WW) is assumed to result in a high late release and is assigned APB 3. Failure at this node means the core leaves the vessel. APB assignments are based on subsequent nodes.
	CZ - No Energetic DW Failure CX – No Energetic Containment Failure DL – Drywell Isolation SP - Suppression Pool scrubbing	If there is energetic DW failure (CZ) and energetic containment failure (CX), these are assumed to be high early releases and APB 1 (highest dose) is assigned. If the drywell is not isolated or Suppression Pool Scrubbing fails, it is assumed that an un-scrubbed release to containment occurs as soon as the vessel is breached. If Containment Spray (CS) fails the sequence is categorized as APB 1 (the highest dose). If Containment Spray is successful, APB 2 is assigned. If the drywell is isolated and Suppression Pool scrubbing (SP) is successful; Energetic Containment (CX) failure or Containment Isolation (IS) failure occur, early containment failure is assumed and APB 3 is assigned. If Energetic Containment (CX) failure or Containment Isolation (IS) failure do not occur, late containment failure is assumed and categorized as APB 5. An exception to the above rules applies to IBL sequences. Class IBL is defined as “Late” Station Blackout events with core damage at greater than 4 hours. Early injection is present and the drywell is likely not failed early allowing for Suppression Pool scrubbing. Therefore, IBL sequences are categorized as APB 5.
	SI - Late RPV injection before Containment Failure	RPV Injection before containment failure lowers the radioactive release. If not preceded by an energetic containment (CX) failure the sequence is categorized as APB 4.
	WW – Suppression Pool Failure Above the Water Line	If the Supp. Pool fails below the water line (WW); this is assumed to result in a high early release and is categorized as APB 1.
	VC – Containment Vent	Sequences with successful containment vent are typicaly assigned to APB 6. However, if vessel breach does not occur because injection is available after core damage and remains available after containment venting, then APB 8 is assigned.

TABLE 4.3-7
CPS LEVEL 2 NODAL ASSUMPTIONS FOR APB ASSIGNMENTS

ACCIDENT CLASS	CPS PRA CONTAINMENT EVENT TREE NODE	ASSUMPTION
2	RX – Core Melt Arrested in Vessel	For accident class 2, RX is always assumed failed.
	WW – Suppression Pool Failure Above the Water Line	If Supp. Pool fails below the water line (WW fails) it is assumed to result in a high late release and is categorized as APB 5. Accident sequences IIE (late GE declaration) with WW failure are assumed early and are categorized as APB1, unless venting is successful (Class IIV) in which case APB 6 is assigned.
	CZ - No Energetic DW Failure DL – Drywell Isolation SP - Suppression Pool scrubbing	For all Class II sequences with early GE declaration, containment failure will occur in the late time frame. Therefore, these are assigned to APB 5. For Class IIE sequences with late GE declaration, containment failure is assumed to occur in the early time frame. If the drywell is not isolated (CZ or DL fail) or Suppression Pool Scrubbing fails (SP fails), it is assumed that an un-scrubbed release to containment occurs as soon as the vessel is breached. These are assigned to ABP 1. If the drywell is isolated (CZ and DL success) and Suppression Pool scrubbing (SP) is successful; no Energetic Containment (CX) failure or Containment Isolation (IS) failure occur, then early containment failure is assumed and APB 4 is assigned.
	VC – Containment Vent	Sequences with successful containment vent are typicaly assigned to APB 6. However, if vessel breach does not occur because injection is available after core damage and remains available after containment venting, then APB 8 is assigned.
4	RX – Core Melt Arrested in Vessel	For accident class 4, RX is always assumed failed.
	CZ - No Energetic DW Failure SP - Suppression Pool Scrubbing WW - Suppression Pool Failure Above the Water Line	If there are no energetic failures of the Drywell, Suppression Pool Scrubbing or Wetwell failure, the sequence is assigned APB 4. If any do fail, APB 1 is assigned.
5	N/A	No collapsed bin is available for containment bypass scenarios. The closest match to a bypass scenario is assumed to be a vessel breach with early drywell and containment failure APB 1. Bin 1 is assigned as it represents the largest release.

TABLE 4.3-8
ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (CPS LEVEL 2 MODEL)

ACCIDENT CLASS (APB NUMBER)	RELEASE FREQUENCY / YR	POPULATION DOSE (50 MILES) PERSON-REM (1)	POPULATION DOSE RISK (50 MILES) (PERSON-REM / YR) (2)
7 LERF (APB 1)	1.27E-07	5.48E+05	6.94E-02
7 non-LERF			
7a (APB 1)	2.89E-08	5.48E+05	1.59E-02
7b (APB 3)	6.54E-09	5.08E+05	3.32E-03
7c (APB 4)	5.72E-08	3.97E+05	2.28E-02
7d (APB 5)	1.36E-06	3.21E+05	4.36E-01
7e (APB 6)	2.90E-07	3.34E+05	9.68E-02
7f (APB 8)	1.62E-08	1.38E+05	2.25E-03
Class 7 non-LERF Total	1.71E-06	3.28E+05 (3)	5.77E-01

Notes to Table 4.3-8:

(1) Population dose values obtained from Table 4.3-6 based on the Accident Progression Bin.

(2) Obtained by multiplying the Release Frequency value from the second column of this table by the Population dose value from the third column of this table.

(3) The weighted average population dose for Class 7 non-LERF is obtained by dividing the total population dose risk by the total release frequency of categories 7a, 7b, 7c 7d, 7e, and 7f.

TABLE 4.3-9
CPS POPULATION DOSE AND DOSE RISK ORGANIZED BY EPRI RELEASE CATEGORY

EPRI RELEASE CATEGORY	FREQUENCY/YR	POPULATION DOSE (PERSON-REM)	POPULATION DOSE RISK (PERSON-REM/YR)
1: No Containment Failure	9.61E-07	2.71E+03	2.61E-03
2: Containment Isolation Failure	4.39E-07	5.48E+05	2.41E-01
7: Phenomena-induced containment failures (LERF)	1.27E-07	5.48E+05	6.94E-02
7: Phenomena-induced containment failures (non-LERF)	1.76E-06	3.28E+05	5.79E-01
8: Containment Bypass	3.88E-08	5.48E+05	2.13E-02
Total:	**3.33E-06**		**9.13E-01**

4.4 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 breach and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT 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 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the EPRI guidance [3], as the mean failure estimated from the available data (i.e., 2 “small” failures that could only have been discovered by the ILRT in 217 tests leads to a 2/217=0.0092 mean value). For Class 3b, consistent with EPRI guidance [3], a non-informative prior distribution is assumed for no “large” failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

The EPRI methodology contains information concerning the potential that the calculated ΔLERF values for several plants may fall above the “very small change” guidelines of the NRC Regulatory Guide 1.174. This information includes a discussion of conservatisms in the quantitative guidance for ΔLERF. EPRI describes ways to demonstrate that, using plant-specific calculations, the ΔLERF is smaller than that calculated by the simplified method. The following is from the EPRI guidance:

> “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 CPS (as detailed in Section 5) means that the Class 7 phenomena-induced containment failures LERF sequences, and Class 8 containment bypass sequences are subtracted from the CDF that is applied to Class 3b, as these sequences always result in LERF. Also, Class 7 phenomena-induced containment failures non-LERF sequences that are Class IBL and Class II are treated as never resulting in LERF due to their late timing and 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 IBL ("Late" Station Blackout) sequences are events with core damage at greater than 4 hours(1). Class II (Loss of Decay Heat Removal) are events with core damage caused by late containment failure (beyond 24 hours)(2). Class IIE sequences where the General Emergency (GE) is postulated to be declared late and there is a potential for LERF are conservatively retained.

Consistent with the EPRI methodology [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). This interval change would lead to a non-detection probability that is a factor of 5.23 (7.85/1.5) higher for the probability of a leak that is detectable only by ILRT testing, given a 15.7-year vs. a 3-yr interval.

(1) A MAAP 4.0.5 analysis CL06008 Title: Accident Class IBL – SBO with RCIC calculates RPV breach in 8.5 hours [40]. The PRA assigns a high probability (i.e., 95%) of a General Emergency (GE) declaration in accordance with Emergency Action Level (EAL) MG1 at 1 hr based on assessment that restoration of power to both divisions vital buses will not occur within 4 hrs [32]. An early GE declaration scenario is a Class IBL and always results in a late release. The PRA assigns the scenario to Class IBE (i.e., 5%) if the GE is declared late resulting in an early release. Class IBE scenarios are included as a contributor to EPRI accident class 3b. With 7+ hours between GE declaration and vessel breach, the IBL PRA scenario is always a late scenario and Class IBL is excluded from contributing to a 3b scenario.

(2) MAAP 4.0.5 Analysis CL110510 Title: LOOP With Loss of Containment Heat Removal credits LPCS without containment heat removal [40]. Containment fails at 34.5 hours in this scenario. Class II Events only include events where a General Emergency (GE) is declared "Early." A "Late" declaration is classified as Class IIE and is included as an EPRI Class 3b scenario. The PRA assigns a high probability (i.e., 95%) a GE would be declared "Early" at approximately 4 hours before containment failure [32]. Assuming injection is lost at the time of containment failure, core damage would occur ~4.4 hours after containment failure and ~8 hours after the GE declaration. Therefore, Class II events are always "Late" events and do not contribute to LERF. A late GE declaration (i.e., Class IIE) is assigned a low probability (i.e., 5%) in the PRA.

EPRI Methodology

This analysis uses the approach outlined in the EPRI Methodology [3]. The steps of the methodology 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 [3].
2. Develop plant-specific population dose rates (person-rem 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 from fifteen years to 15.7 years.
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 and compare this change with the acceptance guidelines of RG 1.174 [4].
5. Determine the impact on the CCFP.
6. Evaluate the impact of external events.

The first three steps of the methodology deal with calculating the change in dose. The change in dose is the historical principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The fourth step in the methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174 [4]. Because there is no change in CDF for CPS, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The fifth step of the methodology calculates the change in containment failure probability, referred to as the CCFP. The NRC has identified a CCFP of less than 1.5% as the acceptance criteria for extending the Type A ILRT test intervals as the basis for showing that the proposed change is consistent with the defense in depth philosophy [7]. As such, this step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174. Step 6 takes into consideration the additional risk due to external events and investigates the impact on results due to varying the assumptions associated with the liner corrosion rate and failure to visually identify pre-existing flaws.

4.5 IMPACT OF EXTENSION ON DETECTION OF STEEL CORROSION THAT LEADS TO LEAKAGE

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [5], consistent with the EPRI methodology [3]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The Clinton primary containment is a pressure-suppression BWR/Mark III containment type that also includes a steel-lined reinforced concrete structure.

The liner sections at Clinton are completely welded together and anchored into the concrete. There is no air space between the liner and the concrete structure. The corrosion/oxidation effects associated with water being in contact with the carbon steel liner and the concrete reinforcing bars are minimized due to the lack of available oxygen between the concrete and the liner. Furthermore, the liner is intended to be a membrane and constitutes a leak-proof boundary for the containment. The liner is nominally 0.25-inch to 0.50-inch thick depending on location and has been oversized to serve as form-work for concrete pouring during construction.

Because concrete was poured against the containment liner significant leakage from containment is not expected even if through-liner corrosion should occur.

The concrete side of the liner is not accessible and cannot be directly inspected by visual means. The large majority of the inside of the containment liner is fully exposed to the containment atmosphere and is accessible for inspection. There is a high likelihood that through-wall defects would be detected through the visual examinations performed. Portions of the inside of the containment liner that are not accessible include the liner below the suppression pool surface and portions of the liner that are obstructed from view by equipment (e.g., piping, cable trays, ductwork) and structural elements (e.g., intermediate concrete floors) next to the containment wall. However, it is estimated that 80% of the inside of the containment liner that is exposed to air is accessible for inspection. The portion of the liner below the suppression pool surface is demonstrated to be low leakage since it is capable of retaining suppression pool water. There are leak test channels at the containment liner seams in the suppression pool area to drain any water that leaks through the suppression pool liner. Therefore, leakage through the suppression pool liner is detectable.

The areas of the containment liner above the suppression pool surface that cannot be inspected are judged to be no more susceptible to degradation than those portions that are accessible. Corrosion identified in areas that are accessible for inspection could indicate that an investigation of similar areas that are not readily accessible may be required. Therefore, any widespread corrosion phenomena would be investigated, and corrective action would be taken. This does not completely preclude the possibility of undetected localized corrosion occurring in areas that are not accessible; however, industry experience has shown a fairly low incidence rate for through-liner corrosion. Furthermore, localized breaches of the containment liner are not likely to lead to significant containment breaches since a leakage path through the reinforced concrete structure would also have to be present. A corrosion sensitivity study has been performed that estimates the impact on the ILRT risk assessment results based upon the above factors.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel structure. This likelihood 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 containment cylinder and dome
- The historical 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

Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures (see Table 4.5-1, Step 1).
- The two corrosion events over a 5.5-year data period are used to estimate the flaw probability in the Calvert Cliffs analysis and are assumed to be applicable to the CPS containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner. It is noted that two additional events have occurred in recent years (based on a data search covering approximately 9 years documented in Reference [26]). In November 2006, the Turkey Point 4 containment building liner developed a hole when a sump pump support plate was moved. In May 2009, a hole approximately 3/8" by 1" in size was identified in the Beaver Valley 1 containment liner. For risk evaluation purposes, these two more recent events occurring over a 9-year period are judged to be adequately represented by the two events in the 5.5-year period of the Calvert Cliffs analysis incorporated in the EPRI guidance (See Table 4.5-1, Step 1).
- Consistent with the Calvert Cliffs analysis, the steel 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 4.5-1, Steps 2 and 3).
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a flaw exists in the steel was estimated as 1.1% for the cylinder and dome region, 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 ILRT target pressure of 37 psig. Consistent with the Calvert Cliffs analysis, probabilities of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis (See Table 4.5-1, 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 containment walls (See Table 4.5-1, Step 4).
- In the Calvert Cliffs analysis, it is noted that approximately 85% of the interior wall surface is accessible for visual inspections. The amount at Clinton is approximately 80%, which is similar. Consistent with the Calvert analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a 5% likelihood of a non-detectable flaw are used. This results in a total undetected flaw probability of 10%, which is assumed in the base case analysis. (See Table 4.5-1, Step 5.) Additionally, it should be noted that to date, all liner corrosion events have been detected through visual inspection.

- 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. This is a particularly conservative assumption for Clinton because it is unlikely that any releases would not be scrubbed in the Mark III containment pool.
- Unlike the Calvert Cliffs design, the Clinton drywell has a steel liner. However, due to the conservative treatment of the containment failures (see previous bullet), the impact of non-detection of corrosion will only be applied to the ILRT extension. The NEI/EPRI characterization of Category 3b as a LERF contributor is considered extremely conservative for a Mark III. Inclusion of drywell liner non-detection failures due to steel corrosion would only increase the conservatism.

TABLE 4.5-1
STEEL CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME	CONTAINMENT BASEMAT
1	Historical Steel Flaw Likelihood Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	Events: 2 2/(70 * 5.5) = 5.2E-3	Events: 0 (assume half a failure) 0.5/(70 * 5.5) = 1.3E-3
2	Age Adjusted Steel 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 / Failure Rate 1 / 2.1E-3 15.7 / 1.6E-2 15.7 year average = 6.71E-3	Year / Failure Rate 1 / 5.0E-4 15.7 / 3.9E-3 15.7 year average = 1.68E-3
3	Flaw Likelihood at 3, 10, and 15 years Uses age adjusted 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) 9.65% (1 to 15 years) 11.23% (1 to 15.7 years)	0.18% (1 to 3 years) 2.42% (1 to 15 years) 2.81% (1 to 15 years)
4	Likelihood of Breach in Containment Given Steel Flaw	1%	0.1%
5	**Visual Inspection Detection Failure Likelihood** Use 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.

TABLE 4.5-1
STEEL CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME	CONTAINMENT BASEMAT
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**0.00071% (at 3 years)** =0.71% * 1% * 10% **0.00965% (at 15 years)** =9.65% * 1% * 10% **0.0112% (at 15.7 years)** =11.2% * 1% * 10%	**0.00018% (at 3 years)** =0.18% * 0.1% * 100% **0.00242% (at 15 years)** =2.42% * 0.1% * 100% **0.00281% (at 15.7 years)** =2.81% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage that is subsequently added to the EPRI Class 3b contribution is the sum of Step 6 for the containment cylinder and dome, and the containment basemat:

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

At 15 years: 0.00965% + 0.00242% = 0.0121%

At 15.7 years: 0.00112% + 0.00281% = 0.0140%

4.6 IMPACT OF DWBT INTERVAL EXTENSION OF RELEASE CATEGORIES

Clinton used a plant-specific MAAP 4.0 model to determine the effects of the increased drywell leakages that could lead to higher containment pressure. The MAAP results, along with other considerations, found that containment failure induced by containment pressurization aggravated by the drywell bypass leakage change is highly unlikely. The relatively small changes postulated due to the DWBT interval extension make no appreciable change in the containment pressurization compared to its ultimate capability. The containment overpressure challenges due to the loss of containment heat removal capability are already accounted for in the Clinton PRA. As such, the perturbation on these sequences caused by slight changes in the drywell bypass area is considered a negligible contributor to CDF.

CPS assigned the equivalent EPRI category and LERF characterization as shown in Table 4.1-4. No credit for the availability of containment spray was taken in the CPS analyses. All EPRI class 1 and 3a were categorized as Non-LERF while 3b was categorized as LERF. The CPS MAAP runs did demonstrate that Class 3b, although treated as LERF, would result in releases significantly below the LERF threshold for CSI release. A similar approach to the previous CPS method to account for the DWBT will be used for this assessment.

TABLE 4.6-1
CPS [19] DWBT AND ILRT LEAKAGE COMBINATION ACCIDENT CLASSES

LEAKAGE COMBINATIONS	DW BYPASS LEAKAGE	CONTAINMENT LEAKAGE	EPRI CLASSIFICATION ASSIGNMENT
AA'	$1DWL_b$	$1L_a$	1 (Non-LERF)
AB'	$1DWL_b$	$10L_a$	3a (Non-LERF)
AC'	$1DWL_b$	$35L_a$(1)	3b (LERF)
BA'1	$10DWL_b$	$1L_a$	1 (Non-LERF)
BB'1	$10DWL_b$	$10L_a$	3a (Non-LERF)
BC'1	$10DWL_b$	$35L_a$	3b (LERF)
CA'1	$35DWL_b$(1)	$1L_a$	3a (Non-LERF)
CB'1	$35DWL_b$	$10L_a$	3b (LERF)
CC'1	$35DWL_b$	$35L_a$	3b (LERF)

Note to Table 4.6-1:

(1) Note that $35L_a$ was used in the prior assessments, but per the updated EPRI guidance as approved by the NRC, $100L_a$ is now used for EPRI Class 3b.

Consistent with the prior assessments, the probability for each combination in Table 4.6-1 is determined by multiplying the conditional probabilities for DWBT and ILRT category by each other. Section 4.6.1 provides an analysis of available Mark III DWBT data to estimate the likelihood of the different DW bypass leakage categories.

4.6.1 DWBT Data Analysis

Table 4.6-2 summarizes the available DWBT results for the Mark III containment types previously reported in Attachment 1, Table 5 of Reference [27] updated with the latest CPS test results. In the prior CPS DWBT extension analysis [19], 300 SCFM was used as the reference leakage for the risk assessment. This will also be used in this assessment for the base drywell leakage rate, DWL_b. Therefore, the analysis is performed using the leakage characteristic of the "as found" state of the drywell. This recognizes both the historical results of the DWBT and the fact that Clinton continuously monitors the DW leakage. CPS is committed to trending this monitored information and noting any adverse trends (which there have been none). Based on these results and the continuous on-line monitoring, it is considered appropriate to use the conservatively high leakage rate of 300 scfm (DWL_b)[1] as the baseline leakage characteristic of a 3/10-year DWBT frequency. This is conservative but is not as large as the Technical Specification allowable. The rationale for using a conservative but more realistic value than the Technical Specification leakage for the drywell is that the last six DWBTs show that the drywell leakage is below 31 scfm (see Table 4.6-2), which is more than two orders of magnitude below the Technical Specification limit (3654 scfm @ 3psig) [18]. The conservative analysis characterization of the DWBT using 300 scfm bounds even the initial drywell leakage (January 1986 test leakage from Table 4.6-2), which had defective electrical penetrations. These defective electrical penetrations were subsequently repaired.

Table 4.6-2 shows industry bypass test results provided in a previous CPS RAI response [27] and updated with the latest (Feb-08) CPS test data. Additional data for recent tests at other sites may be available; however, this information is adequate to show Clinton performance relative to the industry.

(1) A realistic estimate would be closer to 30 scfm. This conservatism affects the population dose estimates.

TABLE 4.6-2
MARK III DRYWELL BYPASS TEST RESULTS

SITE	TEST DATE	LEAKAGE RATE (SCFM)	ACTUAL LEAKAGE / 300 SCFM
Clinton	Jan-86	273	0.91
	Nov-86	20.8	0.07
	Apr-89	18.8	0.06
	Mar-91	21.9	0.07
	May-92	18	0.06
	Nov-93	30.2	0.10
	Feb-08	20.18	0.07
Grand Gulf	Nov-85	2315	7.72
	Nov-86	1568	5.23
	Dec-87	1500	5.00
	Apr-89	1631	5.44
	Nov-90	1591	5.30
	May-92	618	2.06
	Nov-93	869	2.90
Perry	Aug-87	124	0.41
	Jul-89	123	0.41
	Dec-90	797	2.66
	May-92	253	0.84
	Jun-94	2450	8.17
	Jul-94	111	0.37
River Bend	Dec-87	602	2.01
	May-89	141	0.47
	Nov-90	345	1.15
	Aug-92	754	2.51
	Jun-94	421	1.40

Figure 4.6-1 shows a scatter plot of the data in Table 4.6-2 compared to the reference assumed base leakage value, DWL_b, of 300 SCFM. (Note that the assumed base drywell leakage value of 300 SCFM is less than the allowable drywell bypass leakage for CPS of 3654 SCFM at 3.0 psid) [18]. Two of the test data are above $6DWL_b$ and all test data is below $10DWL_b$. The 300 SCFM base case drywell leakage therefore represents a conservative assumption but is used for consistency with the previously accepted ILRT/DWBT extension requests for CPS.

The Technical Specification allowable leakage for the drywell is not used because of the on-line monitoring that is established by the past DWBT. Use of the Technical Specification limit would mischaracterize the Clinton drywell integrity and would make the decision not risk-informed. Therefore, the DW leakage is characterized in the analysis to be 1 times, 10 times, or 100 times a conservative characterization of the drywell leakage, which is referred to in this analysis as DWL_b.

This leads to the specification of the drywell leakage rates consistent with the EPRI ILRT methodology:

Minimal leakage case	300 SCFM @ 3 psid (DWL_b)
10 DWL_b case	3000 SCFM @ 3 psid
100 DWL_b case	30,000 SCFM @ 3 psid

These represent conservative characterizations of the "as found" drywell bypass leakage. The 100 DWL_b case leakage rate of 30,000 SCFM @ 3 psid is less than the maximum allowable rate of 36,540 SCFM @ 3 psid.[(1)]

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is $1.0L_a$ (or 1.0 DWL_b for the drywell).

(1) USAR section 6.2.6.5.1 notes that a leakage rate of 3,654 scfm is 10% of the maximum allowable leakage rate. (3,654 scfm * 10 = 36,540 scfm)

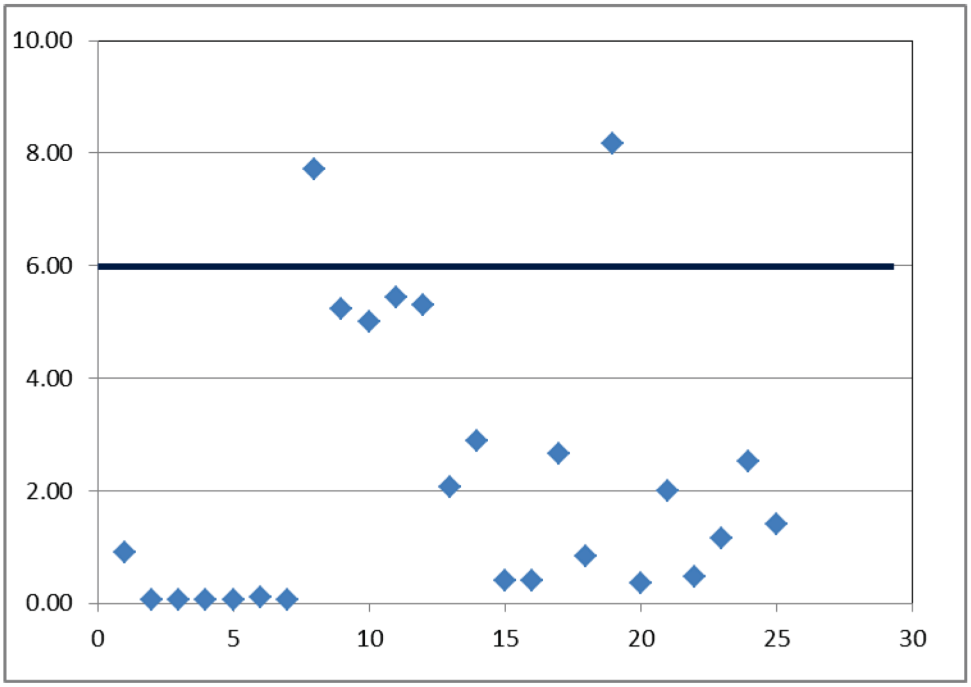


FIGURE 4.6-1
MARK III DWBT RESULTS COMPARED TO 300 SCFM

X Axis = Test # from Table 4.6-2

Y Axis = Test Leakage ÷ 300 SCFM

Only 2 of the data points are measurably in the 6-10 range, and all are below the 10 DWL_b leakage rate assumed for the intermediate category in this assessment.

<u>Drywell Probability of Small and Large Drywell Failures</u>

The base case probabilities for containment (Wetwell) failure probability are applied to the drywell small and large failure events as shown in Table 4.6-3. This is consistent with the approach used in the previous ILRT LARs [19, 36]. Applying these failure probabilities is appropriate for the following reasons:

- In the older BWR containment designs (i.e., Mark I and II), the drywell enclosure is also part of the containment enclosure. Therefore, the data used in the NEI/EPRI approach is reflective of drywell failures. The body of plant experience used considered the older BWR containment designs. Therefore, the NEI/EPRI data is reflective of typical BWR drywell failure mechanisms.

- The CPS containment and drywell designs are similar in many of their construction details. A comparison of the containment and drywell design features is provided in Table 4.1-2. As this comparison shows, the basic designs are much the same and therefore would be expected to have much the same leakage failure mechanisms.
- As noted in Section 4.6.1, Clinton has the ability to continuously monitor the DW leakage. As noted in Section 4.6.1, small airline leaks cause the drywell to pressurize at a rate of approximately 0.03 psi/hr. The operators vent the drywell approximately once per day. It is unlikely that the instrument air leaks will diminish during operation. Therefore, if the drywell pressurization rate went to zero, this would indicate a small drywell leak may have caused this drop in the pressurization rate.
- As shown in Figure 4.6-1, no industry event has exceeded the $10L_a$ threshold.

Based on the above discussion, the EPRI ILRT containment leakage probabilities documented in Section 4.4 are applied to the DW and WW as shown in Table 4.6-3.

TABLE 4.6-3
DW AND WW LEAKAGE PROBABILITIES

DW LEAKAGE SIZE (L_b)	DW LEAKAGE PROBABILITY (BASE)	WW LEAKAGE SIZE (L_a)	WW LEAKAGE PROBABILITY
$1L_b$	0.9885(1)	$1L_a$	0.9885(1)
$10L_b$	0.0092	$10L_a$	0.0092
$100L_b$	0.0023	$100L_a$	0.0023

(1)The probability of assumed normal drywell leakage ($1L_a$) is [1-(Prob. of $10L_b$ + Prob. of $100L_b$)].

These values are therefore used for the base case assessment to represent the DW bypass leakage behavior. Increases to these values are assumed to occur for the different test intervals consistent with the ILRT methodology.

The combined DW Leakage probability and WW Leakage probability are used in the analysis. The base case combined probabilities are shown in the Table 4.6-4:

TABLE 4.6-4
DW AND WW COMBINED LEAKAGE PROBABILITIES

CASE	DW LEAKAGE	CTMT LEAKAGE	DW LEAK PROB	CTMT LEAK PROB	COMBINED PROB	EPRI CLASS
AA'	1	1	0.99	0.99	0.98	1
AB'	1	10	0.99	0.0092	0.0091	3a
AC'	1	100	0.99	0.0023	0.0023	3b
BA'1	10	1	0.0092	0.99	0.0091	1
BB'1	10	10	0.0092	0.0092	8.5E-5	3a
BC'1	10	100	0.0092	0.0023	2.1E-5	3b
CA'1	100	1	0.0023	0.99	0.0023	3a
CB'1	100	10	0.0023	0.0092	2.1E-5	3b
CC'1	100	100	0.0023	0.0023	5.3E-6	3b
Combined Probabilities (Sum)						
					0.9862	1
					0.0115	3a
					0.00232	3b

<u>Containment Overpressure</u>

In the case of accident sequences that are the result of the long-term loss of containment heat removal, containment pressurization and eventual failure are assumed to result in a loss of core coolant injection systems. The CPS PRA models long-term loss of containment heat removal and the resultant loss of core coolant injection systems.

As part of the 2003 LAR [19], an assessment of the possibility that Clinton overpressure containment failures may increase in frequency due to the extension of the DWBT interval was performed by examining those sequences with the highest potential to cause such containment pressure increases. The USAR was reviewed to identify that the limiting condition was a 2” primary system LOCA in the drywell. Using this information and the identified allowable leak areas, several confirmatory MAAP cases were performed to demonstrate the containment challenges for varying bypass flow areas.

The assessment documented the following:

- The containment pressurization due to a LOCA is insensitive to relatively large variations in the DW Bypass area and does not exceed 20 psia except for the "worst case" postulated condition of a 2" LOCA and maximum Technical Specification Bypass.
- The pressure suppression capability of the containment is robust.
- The large volume in the outer containment minimizes the effects of changes in the drywell bypass flow area.
- Any effects of the containment pressurization due to drywell bypass leakage can be effectively terminated by:
  a) RPV depressurization which is directed by the EOPs on exceeding the pressure suppression pressure or
  b) Containment sprays which are directed by the EOPs upon exceeding relatively low containment pressures

     Both of these operating crew actions can be completed over many hours and therefore their success probability is high.
- Subsequent peaks of 30-40 psia in the containment pressure are due to hydrogen combustion events.

  The conclusion from this investigation is that containment failure induced by containment pressurization aggravated by the drywell bypass leakage change is highly unlikely. The relatively small changes postulated due to the DWBT interval extension make no appreciable change in the containment pressurization compared to its ultimate capability. The containment overpressure challenges due to the loss of containment heat removal capability are already accounted for in the Clinton PSA. As such, the perturbation on these sequences caused by slight changes in the drywell bypass area are considered negligible contributors to CDF.

The pressurization issue was addressed in the Safety Evaluation Report (SER) that was part of the NRC letter [11] approving the one-time 15-year interval extension. The SER noted the following:

> "During a small-break loss-of-coolant accident, potential leak paths between the drywell and containment airspace could result in excessive containment pressure if the steam flow into the airspace would bypass the vapor suppression capabilities of the pool. The potential leakage paths between the drywell and the containment are: 1) piping and electrical penetrations; 2) the drywell equipment hatch; and 3) the drywell personnel air lock. The staff found that 1) the electrical penetrations are unlikely to leak significantly, and the design drywell bypass leakage rate is so large that,

> even if the valves in many of the pipes were left open, the design limit would not be exceeded; and 2) both the equipment hatch and drywell air lock have double compression seals and are leak tested in accordance with TSs."

Based on the significant margin found in the 2003 LAR MAAP runs and the deterministic arguments noted above, there is no change in CDF due to the small increases in drywell bypass leakage associated.

5.0 RESULTS

The application of the approach based on EPRI Guidance [3] leads to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.0-1 lists these accident classes.

TABLE 5.0-1
ACCIDENT CLASSES

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures
3b	Large Isolation Failures
4	Small Isolation Failures (Failure to seal –Type B)
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	Containment Bypass
CDF	**All CET End states (including very low and no release)**

The analysis examined CPS-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the categorization 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 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, if applicable. (EPRI 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 Class 6 sequences). Consistent with the EPRI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypass (EPRI Class 8 sequences), large containment isolation failures (EPRI Class 2 sequences), and small containment isolation “failure-to-seal” events (EPRI 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.

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 5.0-1.

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

Step 3 Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval from 15 years to 15.7 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 CCFP.

Step 6 Evaluate the impact of external events.

It is noted that the calculations were generally performed using a spreadsheet such that the presented numerical results may differ very slightly as compared to values if calculated by hand.

5.1 STEP 1 – QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

The CPS PRA Level 2 Model [32] is used to develop the initial set of internal events release categories for use in this analysis. As described in Section 4.3, the release categories were assigned to the EPRI classes as shown in Table 4.3-3. This application combined with the CPS dose risk (person-rem/yr) as shown in Table 4.3-9 forms the basis for estimating the increase in population dose risk.

For the assessment of the impact on the risk profile due to the ILRT extension, the potential for pre-existing leaks is included in the model. These pre-existing leak events are represented by the Class 3 sequences in EPRI TR-1018243 [3]. Two failure modes were considered for the Class 3 sequences, namely Class 3a (small breach) and Class 3b (large breach).

The determination of the frequencies associated with each of the EPRI categories listed in Table 5.0-1 is presented next. Since the Class 1 frequency is determined based on remaining contribution not assigned to other classes, the discussion appears in reverse order starting with EPRI Class 8 and ending with EPRI Class 1. However, EPRI Class 2 is discussed prior to Class 3 since its value is used in the final determination of the Class 3 frequencies.

Class 8 Sequences

This group represents sequences where containment bypass occurs. The failure frequency for Class 8 sequences is 3.88E-08/yr, as documented in Table 4.2-4.

Class 7 Sequences Dose Risk Adjustments

Class 7 consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs. For this analysis, the associated radionuclide releases are based on the application of the Level 2 end states to the Accident Progression Bins from NUREG/CR-4551 as described in Section 4.2.

The Class 7 Sequences are divided into 2 categories, which consist of LERF sequences (excluding the BOC and ISLOCA Class V sequences) and non-LERF sequences. The second category (non-LERF) is further divided into Bins 1, 3, 4, 5, 6, and 8 from NUREG/CR-4551. These non-LERF sequences are grouped into Accident Classes 7a, 7b, 7c, 7d, 7e, and 7f as documented in Table 4.3-8. The failure frequency and population dose for each specific APB are shown in 4.3-8. As shown in Table 4.3-8, the population dose person-rem for Class 7 LERF sequences is based on the largest APB value in

NUREG/CR-4551, while population dose person-rem for Class 7 non-LERF sequences is based on a weighted average of six APB bins.

Class 6 Sequences

These are sequences that involve core damage with a failure-to-seal containment leakage due to failure to isolate the containment. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the EPRI guidance, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 5 Sequences

This group represents containment isolation failure-to-seal of Type C test components. Because these failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 4 Sequences

This group represents containment isolation failure-to-seal of Type B test components. 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 this analysis.

Class 2 Sequences

This group consists of large containment isolation failures. The failure frequency for Class 2 sequences is 4.39E-07/yr, as documented in Table 4.2-4. Note that this frequency is not affected by the ILRT interval change.

Class 3 Sequences

This group represents pre-existing leakage in the containment structure. The containment leakage for these sequences can be either small (in excess of design allowable but $<10L_a$) or large. In this analysis, a value of $10L_a$ was used for small pre-existing flaws and $100L_a$

for large flaws, consistent with the EPRI methodology [3]. The probability of Class 3a and Class 3b leakages are combined leakage probabilities from Table 4.6-4 to account for the impact of drywell leakage on the ILRT interval extension risk metrics. As described in Section 4.4, additional consideration is made to not apply these failure probabilities to those cases that are already classified as LERF (i.e., Class 7 LERF and Class 8 LERF contributions), or would never lead to a LERF (EPRI Class 7 non-LERF contribution consisting of Class II and Class IBL sequences).

Class_3a = 0.0115 * [CDF – (EPRI Class 7 LERF + EPRI Class 8 + Class II + Class IBL)]

= 0.0115 * [3.33E-06 – (1.27E-07 + 3.88E-08 + 6.92E-07 + 7.01E-07)]

= 2.03E-08/yr

Class_3b = 0.00232 * [CDF – (EPRI Class 7 LERF + EPRI Class 8 + Class II + Class IBL)]

= 0.00232 * [3.33E-06 – (1.27E-07 + 3.88E-08 + 6.92E-07 + 7.01E-07)]

= 4.11E-09/yr

For this analysis, the associated containment leakage for Class 3a and Class 3b is $10L_a$ and $100L_a$, respectively, which is consistent with the EPRI methodology [3].

Class 1 Sequences

This group represents the frequency when the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 9.37E-07/yr for CPS and is determined by subtracting all containment failure end states, including the EPRI/NEI Class 3a and 3b frequencies calculated above, from the total CDF.

Class 1 = CDF – (EPRI Classes)

= 3.33E-06 – (4.39E-07 (class 2) + 2.03E-08 (3a) + 4.11E-09 (3b) + 1.27E-07 (7 LERF) + 1.76E-06 (7-Non-LERF) + 3.88E-08 (Class 8))

= 9.37E-07/yr

For this analysis, the associated maximum containment leakage for this group is $1L_a$, consistent with an intact containment evaluation. Note that the value for this Class reported in Table 5.1-1 is slightly lower than that reported in Tables 4.2-3 since the 3a and 3b frequencies are now subtracted from Class 1.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to release of radionuclides to the public have been derived in a manner consistent with the definition of accident classes defined in EPRI TR-1018243 [3] and are shown in Table 5.1-1.

TABLE 5.1-1
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (CPS BASE CASE)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)
1	No Containment Failure	9.37E-07
2	Large Isolation Failures (Failure to Close)	4.39E-07
3a	Small Isolation Failures	2.03E-08
3b	Large Isolation Failures	4.11E-09
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 LERF	Failures Induced by Phenomena (LERF)	1.27E-07
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.76E-06
8	Containment Bypass	3.88E-08
CDF	**All CET End states (including intact case)**	**3.33E-06**

5.2 STEP 2 – DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the weighted average person-rem doses to the population within a 50-mile radius from the plant. The releases are based on the assessment provided in Section 4.3 for CPS (see Table 4.3-9 of this analysis). The results of applying these releases to the EPRI containment failure classifications are summarized as follows:

Class 1 = 2.71E+03 person-rem (at 1.0L_a)

Class 2 = 5.48E+05 person-rem

Class 3a = 2.71E+03 person-rem x 10L_a = 2.71E+04 person-rem

Class 3b = 2.71E+03 person-rem x 100L_a = 2.71E+05 person-rem

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 LERF = 5.48E+05 person-rem

Class 7 non-LERF = 3.28E+05 person-rem

Class 8 = 5.48E+05 person-rem

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [3] for all EPRI classes are provided in Table 5.2-1, which includes the values previously presented in Table 4.3-9 as well as the Class 3a and 3b population doses calculated above.

TABLE 5.2-1
CPS POPULATION DOSE
FOR POPULATION WITHIN 50 MILES

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)
1	No Containment Failure ($1L_a$)	2.71E+03
2	Large Isolation Failures (Failure to Close)	5.48E+05
3a	Small Isolation Failures	2.71E+04
3b	Large Isolation Failures	2.71E+05
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	NA
7 LERF	Failures Induced by Phenomena (LERF)	5.48E+05
7 non-LERF	Failures Induced by Phenomena (non-LERF)	3.28E+05
8 LERF	Containment Bypass	5.48E+05

The above population dose, when multiplied by the frequency results presented in Table 5.1-1, yields the CPS baseline mean dose risk for each EPRI accident class. These results are presented in Table 5.2-2.

TABLE 5.2-2
CPS ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 3 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION (2)		CHANGE DUE TO CORROSION (PERSON-REM/YR)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1(1)	No Containment Failure	2.71E+03	9.37E-07	2.54E-03	9.37E-07	2.54E-03	-4.27E-08
2	Large Isolation Failures (Failure to Close)	5.48E+05	4.39E-07	2.41E-01	4.39E-07	2.41E-01	--
3a	Small Isolation Failures	2.71E+04	2.03E-08	5.51E-04	2.03E-08	5.51E-04	--
3b	Large Isolation Failures	2.71E+05	4.11E-09	1.12E-03	4.13E-09	1.12E-03	4.27E-06
7 LERF	Failures Induced by Phenomena (LERF)	5.48E+05	1.27E-07	6.94E-02	1.27E-07	6.94E-02	--
7 non-LERF	Failures Induced by Phenomena (non-LERF)	3.28E+05	1.76E-06	5.79E-01	1.76E-06	5.79E-01	--
8	Containment Bypass	5.48E+05	3.88E-08	2.13E-02	3.88E-08	2.13E-02	--
CDF	**All CET end states**		**3.33E-06**	**0.914**	**3.33E-06**	**0.914**	**4.23E-06**

Notes to Table: 5.2-2:

(1) Characterized as $1L_a$ 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.

(2) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

5.3 STEP 3 – EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 15-TO-15.7 YEARS

The next step is to evaluate the risk impact of extending the test interval from its current 15-year value to 15.7 years. To do this, an evaluation must first be made of the risk associated with the 15-year interval.

Risk Impact Due to 15-year Test Interval

As previously stated, ILRT 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). The risk contribution is changed based on the EPRI guidance as described in Section 4.4 by a factor of 5 compared to the base case values. The results of the calculation for a 15-year interval are presented in Table 5.3-1.

Risk Impact Due to 15.7-Year Test Interval

The risk contribution for a 15.7-year interval is calculated in a manner similar to the 15-year interval. The difference is in the increase in probability of not detecting a leak in Classes 3a and 3b for the ILRT Type A tests. For this case, the value used in the analysis is a factor of 5.23 compared to the 3-year interval value, as described in Section 4.4, and a factor of 1.047 compared to the 15-year interval value calculated above. The results for this calculation are presented in Table 5.3-2.

TABLE 5.3-1
CPS ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION [(2)]		CHANGE DUE TO CORROSION EXTENSION (PERSON-REM/YR)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1[(1)]	No Containment Failure	2.71E+03	8.39E-07	2.28E-03	8.39E-07	2.28E-03	-5.68E-07
2	Large Isolation Failures (Failure to Close)	5.48E+05	4.39E-07	2.41E-01	4.39E-07	2.41E-01	--
3a	Small Isolation Failures	2.71E+04	1.01E-07	2.75E-03	1.01E-07	2.75E-03	--
3b	Large Isolation Failures	2.71E+05	2.06E-08	5.58E-03	2.08E-08	5.64E-03	5.68E-05
7 LERF	Failures Induced by Phenomena (LERF)	5.48E+05	1.27E-07	6.94E-02	1.27E-07	6.94E-02	--
7 non-LERF	Failures Induced by Phenomena (non-LERF)	3.28E+05	1.76E-06	5.79E-01	1.76E-06	5.79E-01	
8	Containment Bypass	5.48E+05	3.88E-08	2.13E-02	3.88E-08	2.13E-02	--
CDF	**All CET end states**		**3.33E-06**	**0.921**	**3.33E-06**	**0.921**	**5.63E-05**

Notes to Table 5.3-1:

(1) Characterized as $1L_a$ 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.

(2) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

TABLE 5.3-2
CPS ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 15.7 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION (2)		CHANGE DUE TO CORROSION EXTENSION (PERSON-REM/YR)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1(1)	No Containment Failure	2.71E+03	8.33E-07	2.26E-03	8.33E-07	2.26E-03	-6.61E-07
2	Large Isolation Failures (Failure to Close)	5.48E+05	4.39E-07	2.41E-01	4.39E-07	2.41E-01	--
3a	Small Isolation Failures	2.71E+04	1.06E-07	2.88E-03	1.06E-07	2.88E-03	--
3b	Large Isolation Failures	2.71E+05	2.15E-08	5.84E-03	2.18E-08	5.90E-03	6.61E-05
7 LERF	Failures Induced by Phenomena (LERF)	5.48E+05	1.27E-07	6.94E-02	1.27E-07	6.94E-02	--
7 non-LERF	Failures Induced by Phenomena (non-LERF)	3.28E+05	1.76E-06	5.79E-01	1.76E-06	5.79E-01	
8	Containment Bypass	5.48E+05	3.88E-08	2.13E-02	3.88E-08	2.13E-02	--
CDF	**All CET end states**		**3.33E-06**	**0.921**	**3.33E-06**	**0.921**	**6.55E-05**

Notes to Table 5.3-2:

(1) Characterized as $1L_a$ 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.

(2) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

5.4 STEP 4 – DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY

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 CDF below 1E-6/yr and increases in LERF below 1E-7/yr, and small changes in LERF as below 1E-6/yr. Because the ILRT interval extension does not impact CDF, the relevant metric is LERF.

For CPS, 100% of the frequency of Class 3b sequences can be used as a 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 15-year test interval from Table 5.3-1, the Class 3b frequency is 2.08E-08/yr; based on a 15.7-year test interval from Table 5.3-2, the Class 3b frequency is 2.18E-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 15 to 15.7 years (including corrosion effects) is 9.80E-10/yr. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is within Region III of Figure 4 of Reference [4] (very small changes in LERF) when comparing the 15.7-year results to the 15-year results.

5.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the 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 methodology [3]. The NRC has previously accepted similar calculations [7] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The following table shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects in which the flaw rate is assumed to double every five years.

CCFP 1 IN 15 YRS	CCFP 1 IN 15.7 YRS	ΔCCFP$_{15.7\text{-}15}$
71.76%	71.79%	0.03%

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

The change in CCFP of approximately 0.03% as a result of extending the test interval to 15.7 years from 15 years is judged to be insignificant.

5.6 SUMMARY OF INTERNAL EVENTS RESULTS

Table 5.6-1 summarizes the internal events results of this ILRT extension risk assessment for CPS. The internal events risk results associated with a change in the test interval from 1-in-15 years to 1-in-15.7 years all are below the acceptance criteria defined in Section 1.3, namely:

1. Change in LERF = 9.80E-10/yr, which is less than 1.0E-7/yr for the “very small” risk increase as defined in RG 1.174.
2. Change in population dose rate is 3.78E-4 person-rem/yr (0.04%), which is less than 1.0 person-rem/year or 1% of the total population dose.
3. Change in CCFP is 0.03%, which is less than 1.5%.

TABLE 5.6-1
CPS ILRT CASES:
BASE AND 15.7 YR EXTENSION
(INCLUDING AGE ADJUSTED STEEL CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 1 IN 15 YEARS		EXTEND TO 1 IN 15.7 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	2.71E+03	8.39E-07	2.28E-03	8.33E-07	2.26E-03
2	5.48E+05	4.39E-07	2.41E-01	4.39E-07	2.41E-01
3a	2.71E+04	1.01E-07	2.75E-03	1.06E-07	2.88E-03
3b	2.71E+05	2.08E-08	5.64E-03	2.18E-08	5.90E-03
7 LERF	5.48E+05	1.27E-07	6.94E-02	1.27E-07	6.94E-02
7 non-LERF	3.28E+05	1.76E-06	5.79E-01	1.76E-06	5.79E-01
8	5.48E+05	3.88E-08	2.13E-02	3.88E-08	2.13E-02
Total		3.33E-06	0.921	3.33E-06	0.921
ΔDose Rate		---		3.78E-04	
Δ3b LERF		---		9.80E-10	
CCFP %		71.76%		71.79%	
ΔCCFP %		---		0.03%	

5.7 STEP 6 – CONTRIBUTIONS FROM EXTERNAL HAZARDS

Since the risk acceptance guidelines in RG 1.174 are intended for comparison with a full-scope assessment of risk, including internal and external events, a bounding analysis of the potential impact from external events and other hazard groups is presented here because the external event models do not have a full Level 2 model. Appendix A provides a technical adequacy assessment of the Seismic CDF and LERF and the Fire PRA.

Internal Fire Risk [8]

The current CPS Fire PRA model (CL117BF0) is based upon the current FPIE PRA model (i.e., CL117B), both of which were initially updated as part of the 2017 periodic update. The total Fire PRA CDF reported is 7.75E-05/yr; the Fire PRA LERF is 5.30E-06/yr [8]. To reduce conservatism in the ILRT extension analysis, the methodology of subtracting existing LERF from CDF is also applied to the Fire PRA model. The following shows the calculation for Class 3b:

$$\begin{aligned} Freq_{class3b15yr} &= 5 * P_{class3b} * (CDF - LERF) \\ &= 5 * 0.00232 * (7.75E\text{-}5 - 5.30E\text{-}6) = 8.41\text{E-}7 \end{aligned}$$

$$\begin{aligned} Freq_{class3b15.7yr} &= 5.23 * P_{class3b} * (CDF - LERF) \\ &= 5.23 * 0.00232 * (7.75E\text{-}5 - 5.30E\text{-}6) = 8.80\text{E-}7 \end{aligned}$$

Seismic Risk [38]

A quantifiable seismic PRA model for Clinton has not yet been approved for general use in risk applications. However, a calculation for seismic CDF and LERF estimates for the CPS TSTF-505 (RICT) Program, including a review of available industry SSC Fragility Information, was performed [38]. This analysis yielded a seismic CDF of 6.4E-6/yr and a seismic LERF of 1.6E-6/yr. Subtracting seismic LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$\begin{aligned} Freq_{class3b15yr} &= 5 * P_{class3b} * (CDF - LERF) \\ &= 5 * 0.00232 * (6.4E\text{-}6 - 1.6E\text{-}6) = 5.59\text{E-}8 \end{aligned}$$

$$\begin{aligned} Freq_{class3b15.7yr} &= 5.23 * P_{class3b} * (CDF - LERF) \\ &= 5.23 * 0.00232 * (6.4E\text{-}6 - 1.6E\text{-}6) = 5.85\text{E-}8 \end{aligned}$$

Other External Events

In addition to internal fires and seismic events, the CPS IPEEE Submittal [34] analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Hazards

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

Accordingly, these other external event hazards are not included explicitly in this section and are reasonably assumed not to impact the results or conclusions of the ILRT interval extension risk assessment.

External Event Contributor Summary

The external event contributions to Class 3b frequencies are then combined to obtain the total external event contribution to Class 3b frequencies. The LERF increase is conservatively assumed to be the change in Class 3b frequency.

TABLE 5.7-1

CPS 3B (LERF) AS A FUNCTION OF ILRT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS

	3B FREQUENCY (1-PER-15 YEAR ILRT)	3B FREQUENCY (1-PER-15.7 YEAR ILRT)	LERF INCREASE(1)
External Events	8.97E-07	9.38E-07	4.13E-08
Internal Events	2.08E-08	2.18E-08	9.80E-10
Combined	9.18E-07	9.60E-07	4.22E-08

Thus, the total increase in LERF (measured from the baseline 1-per-15 year ILRT interval to the proposed 1-per-15.7 year frequency) due to the combined internal and external events contribution is estimated as 4.22E-08/yr, which meets the guidance for “very small” change in risk, as it is less than 1.0E-7/yr [4].

5.8 16-YEAR ILRT INTERVAL SENSITIVITY

While it is expected the 15.7-year ILRT interval provides ample time for the reasons for the extension request to be mitigated, a sensitivity is performed with a 12-month one-time extension interval (16-year total interval). Table 5.8-1 summarizes the internal events results of this ILRT extension risk assessment for CPS. The internal events risk results associated with a change in the test interval from 1-in-15 years to 1-in-16 years all are less than the acceptance criteria defined in Section 1.3.

TABLE 5.8-1
CPS ILRT CASES:
BASE AND 16 YR EXTENSION
(INCLUDING AGE ADJUSTED STEEL CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 1 IN 15 YEARS CDF (1/YR)	BASE CASE 1 IN 15 YEARS PERSON-REM/YR	EXTEND TO 1 IN 16 YEARS CDF (1/YR)	EXTEND TO 1 IN 16 YEARS PERSON-REM/YR
1	2.71E+03	8.39E-07	2.28E-03	8.31E-07	2.25E-03
2	5.48E+05	4.39E-07	2.41E-01	4.39E-07	2.41E-01
3a	2.71E+04	1.01E-07	2.75E-03	1.08E-07	2.94E-03
3b	2.71E+05	2.08E-08	5.64E-03	2.22E-08	6.02E-03
7 LERF	5.48E+05	1.27E-07	6.94E-02	1.27E-07	6.94E-02
7 non-LERF	3.28E+05	1.76E-06	5.79E-01	1.76E-06	5.79E-01
8	5.48E+05	3.88E-08	2.13E-02	3.88E-08	2.13E-02
Total		3.33E-06	0.921	3.33E-06	0.921
ΔDose Rate		---		5.43E-04	
Δ3b LERF		---		1.41E-09	
CCFP %		71.76%		71.80%	
ΔCCFP %		---		0.04%	

Using the same methodology as in Section 5.7, the external event contributions to Class 3b frequencies are combined to obtain the total external event contribution to Class 3b frequencies with a sensitivity 16-year interval.

TABLE 5.8-2

CPS 3B (LERF) AS A FUNCTION OF ILRT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS

	3B FREQUENCY (1-PER-15 YEAR ILRT)	3B FREQUENCY (1-PER-16 YEAR ILRT)	LERF INCREASE(1)
External Events	8.97E-07	9.57E-07	5.97E-08
Internal Events	2.08E-08	2.22E-08	1.41E-09
Combined	9.18E-07	9.79E-07	6.11E-08

Thus, the total increase in LERF (measured from the baseline 1-per-15 year ILRT interval to the postulated 1-per-16 year sensitivity frequency) due to the combined internal and external events contribution is estimated as 6.11E-08/yr, which meets the guidance for “very small” change in risk, as it is less than 1.0E-7/yr [Reference 4].

6.0 CONCLUSIONS

Based on the results from Section 5, the following conclusions regarding the assessment of the plant risk are associated with temporarily extending the Type A ILRT test frequency to 15.7 years:

- Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines “very small” changes in risk as resulting in increases of CDF below 1E-6/yr and increases in LERF below 1E-7/yr. “Small” changes in risk are defined as increases in CDF below 1E-5/yr and increases in LERF below 1E-6/yr. Since the ILRT extension has no impact on CDF for CPS, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT interval with corrosion included is 9.80E-10/yr (see Table 6.1-1), which falls within the “very small” change region of the acceptance guidelines in Reg. Guide 1.174.
- The change in dose risk for changing the Type A ILRT interval from once-per-15 years to once-per-15.7-years, measured as an increase to the total integrated dose risk for all accident sequences, is 3.78E-04 person-rem/yr using the EPRI guidance with the corrosion included (see Table 5.6-1). NEI 94-01 [1] states that a “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. The results of this calculation meet these criteria.
- The increase in the conditional containment failure probability from the 15-year interval to the 15.7-year interval including corrosion effects using the EPRI guidance (see Table 5.6-1) is 0.03%, which is less than the acceptance criteria of 1.5% identified in the NRC SER on the issue [7] as discussed in Section 1.3.
- To determine the potential impact from external events, an additional assessment from the risk associated with external events was performed. As shown in Table 5.7-1, the total increase in LERF due to internal events and external events is 4.22E-08/yr, which falls within the “very small” change region of the acceptance guidelines in Reg. Guide 1.174.

Therefore, increasing the ILRT interval on a one-time basis to a one-in-15.7-year frequency is not considered to be significant since it represents only a small change in the CPS risk profile.

<u>Previous Assessments</u>

The NRC in NUREG-1493 [6] has previously concluded the following:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 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. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

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

7.0 REFERENCES

1. Nuclear Energy Institute, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01*, Revision 3-A, July 2012.
2. Electric Power Research Institute, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,* EPRI TR-104285, August 1994.
3. Electric Power Research Institute, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325*. EPRI TR-1018243, October 2008.
4. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174*, Revision 3, January 2018.
5. Letter from Mr. C. H. Cruse (Constellation Nuclear, Calvert Cliffs Nuclear Power Plant) to U.S. Nuclear Regulatory Commission, *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Accession Number ML020920100, March 27, 2002.
6. U.S. Nuclear Regulatory Commission, *Performance-Based Containment Leak-Test Program*, NUREG-1493, September 1995.
7. U.S. Nuclear Regulatory Commission, *Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option Of 10 CFR Part 50, Appendix J" and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment Of Extended Integrated Leak Rate Testing Intervals" (TAC No. MC9663)*, Accession Number ML081140105, June 25, 2008.
8. CPS Fire PRA Summary and Quantification Notebook, CL-PRA-021.11, Revision 2, February 2020.
9. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RAS-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.
10. United States Census Bureau QuickFacts, accessed January 2021, https://www.census.gov/quickfacts/.
11. U. S Nuclear Regulatory Commission, Subject: *CLINTON POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. M137675),* Issuance of License Amendment 160 and Safety Evaluation of One-time Extension from 10 to 15 Years, Accession Number ML033360470, January 8, 2004.
12. ERIN Engineering and Research, *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™*, EPRI TR-105189, Final Report, May 1995.
13. Oak Ridge National Laboratory, *Impact of Containment Building Leakage on LWR Accident Risk*, NUREG/CR-3539, ORNL/TM-8964, April 1984.
14. Pacific Northwest Laboratory, *Reliability Analysis of Containment Isolation Systems*, NUREG/CR-4220, PNL-5432, June 1985.

15. U.S. Nuclear Regulatory Commission, *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 (Containment Integrity Check)*, NUREG-1273, April 1988.

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

17. Sandia National Laboratories, *Evaluation of Severe Accident Risks: Grand Gulf, Unit 1,* Main Report NUREG/CR-4551, SAND86-1309, Volume 6, Revision 1, Part 1, December 1990.

18. Exelon, *Clinton Updated Safety Analysis Report (USAR),* Rev. 21, March 2020.

19. Letter from Keith R. Jury (AmerGen Energy Company, LLC, for Clinton Power Station) to U.S. Nuclear Regulatory Commission, *Request for Amendment to Technical Specifications 3.6.5.1, "Drywell" and 5.5.13, "Primary Containment Leakage Rate Testing Program",* Accession Number ML030370524, January 29, 2003.

20. Letter from Jerry C. Roberts (Entergy Operations, for Grand Gulf Nuclear Station) to U.S. Nuclear Regulatory Commission, *License Amendment Request One-time Extension of the Integrated Leak Rate Test and Drywell Bypass Test Interval*, Accession Number ML031400345, May 12, 2003.

21. Letter from Rick J. King (River Bend Regulatory Assurance) to U.S. Nuclear Regulatory Commission, License Amendment Request One-time Extension of the Integrated Leak Rate Test and Drywell Bypass Test Interval, Accession Number ML040540445, February 16, 2004.

22. U.S. Nuclear Regulatory Commission, *Reactor Safety Study*, WASH-1400, October 1975.

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

24. *CPS PRA Summary Notebook (2017 PRA Update)*, CPS PSA-013, Revision 6, February 2020.

25. *CPS PRA Quantification Notebook*, CPS PSA-014, Revision 9, February 2020.

26. Letter from P. B. Cowan (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, *Response to Request for Additional Information – License Amendment Request for Type A Test Extension*, Accession Number ML100560433, February 25, 2010.

27. Letter from Keith R. Jury (AmerGen Energy Company, LLC, for Clinton Power Station) to U.S. Nuclear Regulatory Commission, Additional Information Supporting the Request for Amendment to Technical Specifications 3.6.5.1, "Drywell" and 5.5.13, "Primary Containment Leakage Rate Testing Program", Accession Number ML032671333, September 15, 2003

28. Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.

29. Sandia National Laboratories, Evaluation of Severe Accident Risks: Grand Gulf, Unit 1, Appendices NUREG/CR-4551, SAND86-1309, Volume 6, Revision 1, Part 2, December 1990.
30. Illinois Department of Public Health (IDPH) Population Projections - Illinois, Chicago and Illinois Counties by Age and Sex: July 1, 2010 to July 1, 2025 (2014 edition).
31. SECPOP 4.2, Sector Population, Land Fraction, and Economic Estimation Program, Sandia National Laboratories.
32. *CPS PRA Detailed Level 2 Evaluation Notebook*, CPS PSA-015, Revision 4, February 2020.
33. Sandia National Laboratories, *Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, MACCS Input,* NUREG/CR-4551, SAND86-1309, Volume 2, Revision 1, Part 7, December 1990.
34. Clinton Power Station Individual Plant Examination For External Events, September 1995.
35. 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.
36. Letter from Patrick Simpson (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program," for Permanent Extension of Type A and Type C Leak Rate Test Frequencies, January 25, 2016.
37. Letter from Jennivine Rankin (U.S. Nuclear Regulatory Commission) to Bryan Hanson (Exelon Generation Company, LLC), Clinton Power Station, Unit No. 1 – Issuance of Amendment Regarding Permanent Extension of Type A and Type C Leak Rate Test Frequencies (CAC No. MF7290), September 26, 2017.
38. *External Hazards Assessment for Clinton Power Station*, CL-MISC-026, Revision 0, March 2020.
39. Report # 032362-RPT-12, Clinton PRA Finding and Suggestion Level Fact and Observation Independent Assessment, Rev. 0, January 2019.
40. *CPS PRA Deterministic Calculations Notebook*, CPS PSA-007, Revision 2, December 2011.
41. Report # 032434-RPT-01, Clinton Power Station Fire PRA Finding Level Fact and Observation Closure by Independent Assessment, Rev. 0, February 2020.
42. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 3, December 2020.
43. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Rev. 1, June 2010.
44. NEI 05-04, "Process for Performing PRA Peer Reviews Using the ASME PRA Standard," Rev. 2, November 2008.

45. Clinton Power Station Fire PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, Boiling Water Reactor Owners Group (BWROG), August 2018.
46. Letter from Patrick Simpson (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," April 2020.
47. Clinton Power Station 2009 PRA Peer Review Report, Boiling Water Reactor Owners Group (BWROG), April 2010.

APPENDIX A

PRA TECHNICAL ADEQUACY

A.1 PRA TECHNICAL ADEQUACY INTRODUCTION

This appendix provides information on the technical adequacy of the Clinton Power Station (CPS) Probabilistic Risk Assessment (PRA) Full Power Internal Events (FPIE) model (including internal flooding) and the Fire PRA model.

Exelon employs a multi-faceted approach to establishing and maintaining the technical adequacy and fidelity of PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews.

Section A.2 describes requirements related to the scope of the CPS PRA models. Section A.3 addresses the technical adequacy of the FPIE PRA for this application. Section A.4 similarly addresses the technical adequacy of the Fire PRA for this application.

All PRA models described below have been peer reviewed, and the review and closure of F&Os from the peer review have been independently evaluated to confirm that the associated model changes did not constitute a model upgrade. Currently, there are two (2) open FPIE F&Os that are linked to Supporting Requirements (SRs) assessed as "Not Met" or Capability Category (CC) I. There are no open Fire F&Os (i.e., all F&Os linked to SRs assessed as "Not Met" or "CC I" have been resolved as confirmed by the independent review team). The resolved F&Os and the basis for resolution are documented in the F&O Closure Review reports [39, 41].

A.2 REQUIREMENTS RELATED TO SCOPE OF CPS PRA MODELS

The PRA models discussed in this appendix have been assessed against RG 1.200 [42]. Both the CPS Internal Events PRA model (including internal flooding) and the CPS Fire PRA model are at-power models. The models are capable of quantifying Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Note that this portion of the CPS PRA model does not incorporate the risk impacts of external events. The treatment of seismic risk and other external hazards for this application are discussed in Section A.5.

A.3 SCOPE AND TECHNICAL ADEQUACY OF CPS INTERNAL EVENTS AND INTERNAL FLOODING PRA MODEL

The information provided in this section demonstrates that the CPS FPIE PRA model (including internal flood) meets the expectations for PRA scope and technical adequacy as
presented in RG 1.200. The CPS PRA model for internal events received a formal industry peer review in October 2009 [47]. The CPS (including internal flooding) Peer Review was performed using the NEI 05-04 process [44], the ASME/ANS PRA Standard and Regulatory Guide 1.200. The Peer Review found that 78.5% of the SRs evaluated met Capability Category II or better. There were fifty-six (56) SRs that were assessed as "Not Met" and twelve (12) SRs that were assessed as meeting only Capability Category I. Of the 68 SRs which were assessed as not meeting Capability Category II or better, seven (7) were related to Internal Flooding SRs. Several of the F&Os associated with the open SRs were related to documentation issues.

The 2009 FPIE Peer Review F&Os were addressed during several periodic PRA updates and the resolutions to the F&Os were reviewed by independent review teams in two separate F&O Closures (in December 2018 and November 2019) that included FPIE & Fire PRA F&Os [39, 41]. The independent review teams concluded that for the FPIE PRA, one F&O was dispositioned as "partially resolved" and one F&O was dispositioned as "open." All other F&Os representing a gap to meeting CC II for all SRs were dispositioned as "resolved."

The FPIE PRA Peer Review identified FPIE F&Os associated with SRs assessed as less than CC II. Table A-1 summarizes those F&Os that remain "open" (including those that may be only "partially resolved") at the time of this report. The F&Os discussed in Table A-1 represent the gaps to meeting Capability Category II for the FPIE PRA model.

As documented in Table A-1, only two FPIE F&Os remain open. An assessment with respect to the impact on this application is also provided for each open F&O. The ILRT extension evaluation only requires an assessment of CC-I [1]. Based on the assessments provided in Table A-1, it is concluded the CPS Internal Events PRA (including internal flooding) is of adequate technical capability to support this ILRT extension application.

TABLE A-1

CLINTON FPIE / INTERNAL FLOODING PRA PEER REVIEW OPEN FACTS AND OBSERVATIONS (POST F&O CLOSURE)

F&O ID	ORIGINATING SR(S)	F&O DETAILS	BASIS FOR SIGNIFICANCE	POSSIBLE RESOLUTION	STATUS	DISPOSITION FROM F&O CLOSURE REVIEW	MAINTENANCE VS. UPGRADE	IMPACT TO ILRT EXTENSION APPLICATION
1-32	LE-E1 QU-C1 QU-C2 HR-H3 LE-E4 QU-A5 HR-G7	CPS-PSA-004 Section 5.2 discusses the use of screening values used for HEPs in order to identify cutsets with dependent HEPs. However, only twelve of the over 100 basic events modeling post-initiator operator actions are listed in Table 5.2-1 as using screening values to identify dependency. Of these, six use a value of 1.0E-02 and one uses a value of 1.0E-03. The remaining five use a value of 0.1. It appears that all other HEPs are quantified with their nominal values. Use of such low probability values is likely to result in combinations of dependent HEPs being omitted by truncation values. Use of a sufficiently high value for HEPs is required by SR QU-C1 and not using a sufficiently high value would result in an inadequate assessment of dependent HEPs. (This F&O originated from SR HR-G7)	An inadequate process to identify combinations of operator actions can result in significantly underestimating CDF and LERF.	Solve with all post-initiator HEPs set to 1.0 and identify all combinations of operator action-related HEPs. Perform dependency analyses for all combinations.	Partially Resolved	<u>Clinton Assessment:</u> Section 5.3 and Appendix K of the Human Reliability Analysis Notebook (CL-PRA-004) summarizes HRA Dependency Analysis methodology and results. For CDF and LERF, the FPIE model was quantified with all post-initiator HEPs set to 0.1 or higher at the truncation levels of 5E-9/yr (CDF) and 5E-10/yr (LERF). These truncation levels were selected because they capture all risk-significant post-initiator operator actions. Using the HRA Calculator Dependency Module, all dependent combinations were reviewed for proper dependency levels and order. Once reviewed, a floor value of 1E-06 or 5E-07 may be imposed on the dependent joint HEP depending on the timing of the operator actions. The final FPIE model quantification uses the 0.1 or higher seed values for all post-initiator HEPs and the adjusted dependent joint HEP is recovered using a post-processing recovery file. <u>Independent Review Team Assessment:</u> A check of the CAFTA RR Database indicates that the post-initiator HEPs were set to 0.1 (or greater) prior to dependency analysis. The value of 0.1 can be acceptable depending upon what truncation level is used for the dependency analysis and whether all multiple independent HFEs are recovered by combination HFEs and Joint HEPs. The resolution of this Finding is correlated to Finding 1-34.	<u>Clinton Assessment:</u> Maintenance: Methodology and tools consistent with previous PRA updates. <u>Independent Review Team Assessment:</u> Since no new methods were applied and existing methods were not applied in a different context, this constitutes model maintenance.	This issue has minimal impact on the ILRT extension application since all risk-significant HRA dependencies are captured through the current methodology and results. A review of the CDF & LERF cutsets was performed to determine if any HRA dependent combinations exist without escalated dependent joint HEPs (i.e., they assume zero dependence and thus the HEPs are unaltered). Separately, a review of the combinations concluded that a majority of the unanalyzed dependent combinations are related to time-phased actions (i.e., early vs. late) where no additional dependency need be assigned between the actions because the time-phased calculations already reflect the impacts of those dependencies. A few legitimate dependent combinations were identified upon further review; however, increasing the dependent joint HEPs for these groups does not substantially impact the overall risk results. Further justification for the chosen truncation level used in the HRA Dependency Analysis is required in a future model update. **Therefore, this open item is primarily a documentation issue.**

TABLE A-1

CLINTON FPIE / INTERNAL FLOODING PRA PEER REVIEW OPEN FACTS AND OBSERVATIONS (POST F&O CLOSURE)

F&O ID	ORIGINATING SR(S)	F&O DETAILS	BASIS FOR SIGNIFICANCE	POSSIBLE RESOLUTION	STATUS	DISPOSITION FROM F&O CLOSURE REVIEW	MAINTENANCE VS. UPGRADE	IMPACT TO ILRT EXTENSION APPLICATION
1-34	LE-E1 QU-C1 QU-C2 HR-H3 LE-E4 QU-A5 HR-G7	Solving the PRA models with some HEPs at nominal can result in cutsets with multiple operator actions being truncated out or with the combined probability of all operator actions much below the 1E-6 or 5E-7 floor that the HRA notebook says is used. The peer review team quantified the PRA model with post-initiator HEPs set to 0.1 and identified a significant number of cutsets containing combinations of basic events representing operator action failure. These combinations were reviewed and a large number of combinations identified in this review were not included in the CPS HRA dependency evaluation. (This F&O originated from SR HR-G7)	The solution method used likely under predicts the risk values. This under prediction could be significant based on the total number of operator actions included in the CPS model.	Solve the PRA model with operator action failure probability values set to a high value.	Open	Clinton Assessment: See discussion for F&O 1-32. Independent Review Team Assessment: The CL-PRA-004 Rev. 6 document was reviewed. The final model cutsets were re-imported into the existing HRA DAF files (for FPIE CDF only), using a copy of the HRAC database with all 1.0 HEPs removed and the inhibit ADS also removed per the analyst notes for that HFE. This process was used to determine if there are combinations of HFEs occurring in the final results with all HEPs set to nominal values and no combination event applied. 318 new combinations were identified (in addition to the 216 that were originally identified and implemented), several of which had FV values above 5E-03 as calculated by the HRAC (which is not a true risk metric but a good approximation). For example, 1FWOPFLWCTRL-H-- and 1FWOPMANINIT-H-- appear as a combination together and have a dependency level of HD, confirmed in the HRA Calculator via override notes, however when this pair of HFEs appears together it is not recovered with a combination event. This combination has a an FV value of 2.9E-01 as calculated by the HRAC (again, not a true risk metric but a good approximation). This suggests it is likely risk significant when dependencies are accounted for, and additional unanalyzed combinations may also be present when dependencies are accounted for. The review team's concern is that potentially risk significant combinations of HFEs are not captured through the current approach, due to the chosen truncation level for the dependency identification (5E-9 / 5E-10 for CDF/LERF) in conjunction with the elevated HEP level chosen (0.1). This could under predict risk results as stated in the original F&O, and is supported by the observations noted above. It is noted that the example combination above did appear in the 1E-9 / 5E-11 identification cutsets that were included in the dependency files, but not used.	Clinton Assessment: Maintenance: Methodology and tools consistent with previous PRA updates. Independent Review Team Assessment: Maintenance - modeling error, approach will not change.	See discussion for F&O 1-32.

A.4 SCOPE AND TECHNICAL ADEQUACY OF CPS FIRE PRA MODEL

The CPS Fire PRA (FPRA) Peer Review [45] was performed in April 2018 using the NEI 07-12 Fire PRA peer review process [43], the ASME/ANS PRA Standard, ASME/ANS RA-Sa-2009 [9] and Regulatory Guide 1.200 [42]. The purpose of this review was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used.

The 2018 CPS FPRA Peer Review was a full-scope review of the CPS at-power FPRA against all technical elements in Part 4 of the ASME/ANS PRA Standard, including the referenced internal events Supporting Requirements (SRs). The Peer Review found that 96.9% of the SRs evaluated met Capability Category (CC) II or better. There were five (5) SRs that were assessed as "Not Met" and eight (8) SRs that were assessed as meeting only CC I. Many of the F&Os, leading to open SRs, were related to documentation issues.

The 2018 FPRA Peer Review F&Os were addressed in subsequent FPRA updates and the resolutions to the F&Os were reviewed by independent review teams in two separate F&O Closures (in December 2018 and November 2019) that included FPIE & FPRA F&Os [39, 41]. The independent review teams concluded that for the FPRA, all F&Os have been dispositioned as "resolved." Therefore, there are no open F&Os to discuss for this application.

Given the resolution of all F&Os related to SRs assessed with less than a CC II, it is concluded that the CPS FPRA is of adequate technical capability to support the ILRT extension application.

A.5 SCOPE AND TECHNICAL ADEQUACY OF CPS SEISMIC CDF AND LERF ESTIMATES

Estimates of Seismic CDF and Seismic LERF we derived for use in the CPS TSTF-505 program [38]. The Seismic CDF and Seismic LERF derived for use in the TSTF-505 program reflect best estimates using available information and potentially conservative assumptions; the methodology and technical adequacy are detailed in the CPS TSTF-505 LAR [46]. The technical adequacy of the Seismic CDF and Seismic LERF are qualitatively found to be adequate to support the conclusions found in Section 6.0 of this document.

A.6 SUMMARY

A PRA technical adequacy evaluation was performed consistent with the requirements of RG 1.200. This evaluation combined with the details of the results of this analysis demonstrate with reasonable assurance that the proposed one-time extension to the ILRT interval for CPS Unit 1 to 15.7 years satisfies the risk acceptance guidelines in RG 1.174 and that the associated PRA models are technically adequate.

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Cc: Nicely, Ken M.:(Exelon Nuclear) <ken.nicely@exeloncorp.com>
Subject: RE: CL-LAR-012 Revision 2 Signatures for Approval

By copy of this email, I approve the Rev.2 version

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Subject: RE: CL-LAR-012 Revision 2 Signatures for Approval

I, Justin Sattler, Jensen Hughes PRA Engineer, approve CL-LAR-012 R2 as the PRA Preparer.

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Subject: CL-LAR-012 Revision 2 Signatures for Approval

I, Victoria Warren, Fleet risk Management Engineer, approve CL-LAR-012 R2 as the PRA Reviewer.

Victoria Warren
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Subject: RE: CL-LAR-012 Revision 2 Signatures for Approval

I, Charles Standridge, Jensen Hughes PRA Engineer, approve CL-LAR-012 R2 as the PRA Reviewer.

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I, Matt Johnson, Jensen Hughes Lead PRA Engineer, approve CL-LAR-012 R2 as the PRA Reviewer.

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