Maria Korsnick
Vice President

1503 Lake Road
Ontario, New York 14519-9364
585.771.3494
585.771.3943 Fax
maria.korsnick@constellation.com

Constellation Energy
R.E. Ginna Nuclear Power Plant

March 10, 2005

Ms. Donna M. Skay
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Application for Amendment to Facility Operating License
Proposed License Amendment to Provide a One-time
Integrated Leak Rate Test (ILRT) Interval Extension
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Ms. Skay:

In accordance with the provisions of 10 CFR 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) is submitting a request for an amendment to the Technical Specifications (TS) for the R.E. Ginna Nuclear Power Plant.

This proposed change will revise TS section 5.5.15, Containment Leakage Rate Testing Program, to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate test, which was completed on May 31, 1996. This proposed change is based on and has been evaluated using the "risk informed" guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The "Risk Assessment for R. E. Ginna Nuclear Power Plant Regarding ILRT (Type A) Extension Request" is provided as an attachment to this letter. This risk assessment is based on the dose calculations that were also used to support the Ginna License Renewal evaluation and submittal (Reference NUREG-1437, "*Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 14, Regarding R. E. Ginna Nuclear Power Plant*," dated January 2004).

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Enclosure 1 provides a description of the proposed change and an explanation of the basis for the change. Also contained in Enclosure 1 are typical NRC questions (based on previous industry submittals) and the Ginna LLC response to those questions. Enclosure 2 details the basis for the determination that the proposed change does not involve a significant hazards consideration. Enclosure 3 provides the existing TS page marked up to show the proposed change. Enclosure 4 provides revised (clean) TS pages. Attachment 1 contains the risk assessment for the ILRT (Type A) extension request.

Ginna LLC requests approval of the proposed license amendment by March 1, 2006 to support the planning activities for the outage scheduled for the Fall of 2006, with the amendment being implemented within 90 days.

A similar request was approved for Vogtle Units 1 and 2 in a letter dated January 12, 2004, Clinton Power Station Unit 1 in a letter dated January 8, 2004, Donald C. Cook Units 1 and 2 in a letter dated February 25, 2003, Lasalle Units 1 and 2 in a letter dated November 19, 2003, Indian Point Nuclear Generating Unit No. 3 in a letter dated April 17, 2001 and James A. FitzPatrick in a letter dated September 28, 2004.

There are no new commitments made by this submittal. This submittal contains no proprietary information.

Any questions concerning this submittal should be directed to Thomas Harding, Nuclear Safety and Licensing at (585) 771-3384.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

Enclosures:	1.	Basis for Change Request
	2.	Significant Hazards Consideration Evaluation and Environmental Consideration
	3.	Proposed Technical Specification Change (markup)
	4.	Revised Technical Specification Pages
Attachment:	1.	Risk Assessment for R. E. Ginna Nuclear Power Plant Regarding ILRT (Type A) Extension Request

STATE OF NEW YORK:
: TO WIT:
COUNTY OF WAYNE :

I, Mary G. Korsnick, being duly sworn, state that I am Vice President – R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Mary G. Korsnick

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 10 day of March, 2005.

WITNESS my Hand and Notarial Seal:

Sharon L. Miller
Notary Public

SHARON L MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 2006

My Commission Expires:

12-21-06
Date

xc: Ms. Donna M. Skay (Mail Stop O-8-C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

Mr. Carey W. Fleming
Sr. Counsel - Nuclear Generation
Constellation Generation Group, LLC
750 E. Pratt Street, 17th Floor
Baltimore, MD 21202

Mr. Peter R. Smith
New York State Energy, Research,
and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul D. Eddy
Electric Division
NYS Department of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Enclosure 1
R.E. Ginna Nuclear Power Plant

Basis for Change Request

Basis for Change Request

1.0 PROPOSED CHANGE

R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) is proposing a change to the R.E. Ginna Nuclear Power Plant (Ginna) Technical Specifications (TS). This proposed change will revise TS section 5.5.15, Containment Leakage Rate Testing Program, to reflect a onetime deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate test, which was completed on May 31, 1996. The proposed change involves a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J." The current ten (10) year ILRT for Ginna is due by May 31, 2006, which would require the test to be performed during the 2006 Refueling Outage. The proposed exception would allow the next ILRT for Ginna to be performed within fifteen (15) years from the last ILRT as opposed to the current ten (10) year frequency. The proposed change would revise Section 5.5.15, "Containment Leakage Rate Testing Program" of the Ginna Technical Specifications to add the following statement:

> ..., as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":
>
> > Section 9.2.3: The first Type A test after the May 31, 1996 Type A test shall be performed by May 31, 2011.

This one-time exception will result in the following:

- The Type A Containment ILRT will be performed during the refueling outage currently scheduled for 2011.

- The proposed amendment would provide considerable savings in radiation exposure to personnel, cost, and critical path time during the 2006 refueling outage, by deferring the Type A test for an additional five (5) years.

2.0 BASIS FOR PROPOSED CHANGE

2.1 10 CFR 50, Appendix J, Option B

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 Technical Specifications. The limitation on containment leakage provides assurance that the containment will perform its design function following plant design basis accidents.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Amendment No. 61 of the Ginna TS was issued February 13, 1996, to reflect the adoption of the requirements of 10 CFR Part 50, Appendix J, Option B. This amendment revised Technical Specifications to require Type A, B, and C testing in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specified a method acceptable to the NRC for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based containment leakage rate testing program 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, B, and C leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as found" leakage and maintenance history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak-tightness for five reactor/containment types including a pressurized water reactor (PWR) with a large dry containment. Ginna is a PWR with a large dry containment. NUREG-1493 made the following observations with regard to decreasing the test frequency.

- Reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential 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 the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements (containment bypass or exceeding design pressure requirements dominate risk), the overall effect is very small. NEI 94-01 requires that Type A testing be performed at least once per ten (10) years based upon an acceptable performance history. Acceptable performance history is defined as two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 La. Based upon the acceptable ILRTs for Ginna (April 15, 1993 and May 31, 1996), the current test interval for Ginna is once every ten (10) years, with the next test due to be performed by May 31, 2006.

2.2 Ginna Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its Loss of Coolant Accident (LOCA) configuration. Industry test experience has demonstrated that Type B and C testing detect a large percentage of containment leakage and that the percentage of containment leakage that is detected only by integrated containment leakage testing is very small.

Ginna has undergone nine (9) operational Type A tests. The results of these tests demonstrate that the Ginna containment structure remains an essentially leak-tight barrier and represents minimal risk to increased leakage. These plant-specific results support the conclusions of NUREG-1493. As specified in Ginna Technical Specifications Section 5.5.15, the maximum allowable containment leakage rate L_a, at P_a, is 0.2% of primary containment air weight per day. The Ginna ILRT results are provided below:

Expressed in % of Allowable Leakage (La), < .75 L_a = Tech Spec

L_a = 0.2 % Containment air weight/24 hours @ 60 psig

L_a = 0.1528 % Containment air weight/24 hours @ 35 psig (1)

Ginna Integrated Leak Rate Test History

Date	Leakage	Test Pressure	%L_a	Margin to .75% L_a
1969	0.0387 wt %/day	60.0 psig	19%	75%
1972	0.0620 wt %/day	35.0 psig	40%	46%
1976	0.0440 wt %/day	35.0 psig	28%	62%
1978	0.0490 wt %/day	35.0 psig	32%	58%
1982	0.0197 wt %/day	35.0 psig	12%	83%
1986	0.06407 wt %/day	35.0 psig	42%	44%
1989	0.0463 wt %/day	35.0 psig	30%	60%
1993	0.0540 wt %/day	35.0 psig	35%	53%
1996	0.11967 wt %/day	60.0 psig	59%	21% (2)

(1) The performance of reduced pressure ILRT testing was removed from the Ginna Technical Specifications with the issuance on February 13, 1996 of Amendment 61 to the facility operating license which implemented the amended regulation 10 CFR Part 50, Appendix J, Option B.

(2) The 1996 ILRT was performed in conjunction with the performance of a Structural Integrity Test. The ILRT was performed upon the completion of all containment structure restoration activities following steam generator replacement.

2.3 Description of Containment

The containment consists of the concrete containment structure, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) in accordance with Atomic Industry Forum (AIF) GDC 10 and 49. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat base mat, and a hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. Each weld seam on the inside of the liner has a leak test channel welded over it to allow independent testing of the liner when the containment is open. The liner is also insulated with closed-cell polyvinyl foam covered with metal sheeting up to a point above the spring line and below the containment spray ring headers. The function of the liner insulation is to limit the mean temperature rise of the liner to only 10°F at the time associated with maximum pressure following a DBA.

The containment hemispherical dome is constructed of reinforced concrete designed for all DBA related moments, axial loads, and shear forces. The cylinder wall is prestressed vertically and reinforced circumferentially with mild steel deformed bars. The base mat is a reinforced concrete slab that is connected to the cylinder wall by use of a hinge design which prevents the transfer of imposed shear from the cylinder wall to the base mat. This hinge consists of elastomer bearing pads located between the bottom of the cylinder wall and the base mat, and high strength steel bars which connect the cylinder walls horizontally to the base mat.

The cylinder wall is connected to sandstone rock located beneath the containment by use of 160 post-tensioned rock anchors that are coupled with tendons located in the cylinder wall. This design ensures that the rock acts as an integral part of the containment structure.

The concrete containment structure is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the outside environment to within the limits of 10 CFR 100.

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was originally strength tested at 69 psig (115% of design). The acceptance criteria for this test was 0.1% of the containment air weight per day at 60 psig. Following successful completion of this test, the accident analyses were performed assuming a leakage rate of 0.2% of the containment air weight per day. This leakage rate, in combination with the minimum containment engineered safeguards operating results in offsite doses well within the limits of 10 CFR 100 in the event of a DBA.

The leakage rate of 0.2% of the containment air weight per day is defined in 10 CFR 50, Appendix J, Option B, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.2% per day in the safety analysis at $P_a = 60$ psig.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.
Reference TS Bases 3.6.1

2.4 Containment Leakage Consideration for Operability

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$ except prior to entering MODE 4 for the first time following performance of periodic testing performed in accordance with 10 CFR 50, Appendix J, Option B (see Ginna TS LCO 3.6.1). At that time, the combined Type B and C leakage must be $< 0.6\ L_a$ on a maximum pathway leakage rate (MXPLR) basis, and the overall Type A leakage must be $< 0.75\ L_a$. At all other times prior to performing as found testing, the acceptance criteria for Type B and C testing is $< 0.6\ L_a$ on a minimum pathway leakage rate (MNPLR) basis. In addition to leakage considerations following a design basis LOCA, containment OPERABILITY also requires structural integrity following a DBA.

Compliance with the Limiting Conditions for Operation discussed above will ensure a containment configuration, including personnel and equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.
Reference TS Bases 3.6.1

2.5 Containment Operational Performance

Containment pressure is maintained $\geq$-2.0 psig and $\leq$1.0 psig during plant operation and is monitored by TS surveillance on a frequency of every 12 hours. *Reference TS SR 3.6.4.1*

Containment internal pressure is an initial condition used in the DBA analyses performed to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and Steam Line Break (SLB) Inside Containment, which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The worst case SLB generates larger mass and energy releases than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint.

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). The maximum containment pressure resulting from the worst case SLB does not exceed the containment design pressure, 60 psig.

The containment was also designed for an external pressure load equivalent to -2.5 psig. However, internal pressure is limited to -2.0 psig based on concerns related to providing continued cooling for the reactor coolant pump motors inside containment.
Reference TS Bases 3.6.4

2.6 Renewed Facility Operating License

By letter dated July 30, 2002, Rochester Gas and Electric Corporation submitted the license renewal application (LRA) for Ginna in accordance with 10CFR54. Through the LRA, RG&E requested that the NRC renew the operating license for Ginna (license number DPR-18) for a period of 20 years beyond the expiration of September 18, 2009.

The NRC on May 19, 2004 issued the Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. The renewed facility operating license was issued on the basis of the NRC review of the application dated July 30, 2002, as supplemented by letters submitted to the NRC through January 9, 2004.

The technical basis for issuing the renewed license is set forth in NUREG-1786, "Safety Evaluation Report Related to the License Renewal of the R.E. Ginna Nuclear Power Plant," dated May 2004. The results of the environmental reviews related to the issuance of the renewed license are contained in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 14, Regarding R.E. Ginna Nuclear Power Plant," dated January 2004. The safety evaluation report (SER) documents the technical review of the Ginna LRA by the NRC staff.

It is not the intent of this Technical Specifications amendment request to change or modify the basis or any commitments associated with the License Renewal of the R. E. Ginna Nuclear Power Plant as documented in NUREG-1786 and NUREG-1437.

2.7 Power Uprate Project

Ginna LLC is undergoing a project to uprate the licensed power level of Ginna. This project is being performed ensuring that peak containment pressure for both LOCA and SLB Inside Containment are bounded by the containment design pressure of 60 psig. This project will have no effect on current or future testing performed in accordance with Technical Specification section 5.5.15, Containment Leakage Rate Testing Program.

2.8 Containment Dome Material and Structural Assessment

Two sections of the dome on the Ginna containment building were demolished for the replacement of steam generators in 1996. This construction period provided a unique window of opportunity to perform an in-depth aging investigation on this structure after almost 30 years of service.

A detailed plan was developed to perform on site examination of the containment dome during the construction phase. During this phase visual inspection was conducted to note the actual in-situ condition of the structure at different stages of demolition. Several core samples of concrete were obtained at selected locations before the demolition and also several rebar samples were collected after they were exposed. These samples were evaluated for the effects of aging. The evaluation program for the samples consisted of physical property determinations using the following methods:

- Visual inspection of concrete cores, rebars, and cadwelded rebars;
- Air-void analysis of concrete specimens taken from the cores;
- Petrographic analysis of concrete specimens taken from the cores;
- Chloride content analysis at different locations (depths) along the length of the cores; and
- Compressive strength of the concrete cores.

A visual inspection of the demolished area was conducted. The general condition of the inspected area was excellent. No signs of degradation or damage was detected. The exposed rebars did not exhibit any signs of corrosion or other degradation. The liner plates and welded joints accessible to inspection had no sign of deterioration. In summary this study showed that the concrete dome of the containment building after almost 30 years of being exposed to the environment has not degraded and the effect of aging has been insignificant on this particular structure.
Reference Ginna Technical Report No. 96135-TR-01, "Containment Dome Material and Structural Assessment."

2.9 Common Industry Questions Related to ILRT Extensions

In support of the LAR, Ginna LLC has reviewed the applications of previous licensee requests to see what common questions were asked. The following provide a response to common questions raised by the NRC with respect to these requests.

<u>Question 1:</u>
Since there is no description (or summarization) regarding the containment ISI program being implemented at the plant included in the submittal (reference), provide a description of the ISI methods that provide assurance that in the absence of a containment integrated leak rate testing (ILRT) for 15 years, the containment structural and leak-tight integrity will be maintained.

<u>Response 1:</u>
Containment leak tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the 1992 Edition through the 1992 Addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI. More specifically, subsections IWE and IWL. The ASME Section XI, Subsections IWE/IWL Inservice Inspection Program was implemented at Ginna Station in response to NRC rule making. In 1996, 10CFR50.55(a) was amended to impose the Inservice Inspection (ISI) requirements of ASME Section XI, Subsection IWE for steel containments (Class MC) and steel liners for concrete containments (Class CC) and the examination requirements of Subsection IWL for reinforced and prestressed concrete containments (Class CC). The full scope of subsection IWE includes steel containment shells and their integral attachments, steel liners for concrete containments and their integral attachments; containment hatches and air locks; seals, gaskets and moisture barriers, and pressure-retaining bolting. The scope of Subsection IWL includes reinforced concrete and unbonded post-tensioning systems.

The Containment Program was implemented at Ginna Station in September 1998. This Program outlines the first IWE/IWL Inservice Inspection Interval requirements and was formally included in the ASME Section XI ISI Program. The primary inspection methods are visual examinations (VT-1, VT-3, VT-1C VT-3C) with limited supplemental volumetric and surface examinations as necessary. Tendon anchorages and wires are visually examined. Tendon wires are tested for verification that minimum mechanical properties requirements are met. Tendon corrosion protection medium is analyzed for alkalinity content and soluble ion concentrations. Prestressing forces are measured in selected sample tendons. The first IWE/IWL ISI Interval ends in September 2008.

The first interval inspections were completed in September 2001. All accessible concrete surfaces of the Containment Structure were visually examined. All indications were recorded photographically and dispositioned by engineering evaluation. The material condition of the Containment Structure was judged to be excellent. No evidence of significant degradation was found.

Continued implementation of the IWE/IWL Inservice Inspection Program will provide ongoing confirmation that the effects of aging for Containment Structure concrete components remain inactive at Ginna Station and that their intended functions will be maintained during the period of both the extended ILRT test interval and the extended operation of the station.

Consistent with NUREG-1801, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program includes inspections and leak rate tests which would indicate the presence of significant degradation due to loss of material from all applicable corrosion mechanisms. Additionally, plant operating experience has shown that borated water spills in containment have the potential to impact the containment liner. Accordingly, the Boric Acid Corrosion Program is also credited with assessing and managing loss of material in the containment liner (procedure IP-IIT-7).

Furthermore, Regulatory Guide 1.163 position C.3 require licensees to conduct visual inspections of the accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration. These requirements will not be changed as a result of the extended ILRT interval. Inspections in accordance with the requirements of Regulatory Guide 1.163 position C.3 have been conducted during the 10 year interval starting with the completion of the last ILRT and associated SIT on May 31, 1996 and were completed on the following dates:

- November 14, 1997,
- April 8, 1999,
- October 13, 2000

The next inspection is scheduled to be performed during the 2005 refueling outage. The subsequent inspection is scheduled for the 2008 refueling outage which ensures that Ginna is continuing to meet the intent of Regulatory Guide 1.163 position C.3.

In addition, Appendix J, Type B local leak tests performed to verify the leak tight integrity of containment penetration air locks, seals, and gaskets are not affected by the change to the Type A test frequency. Likewise the Appendix J, Type C local leak tests, which are performed to verify the leak tight integrity of containment isolation valves, are not affected by the change to the Type A test frequency.

In addition, a Structures Monitoring Program was developed in response to the requirements of the Maintenance Rule (10CFR50.65) and the License Renewal Rule (10CFR54). The program is implemented in accordance with approved plant procedures. One of the primary elements of the program is monitoring the material condition of concrete structures, including inspection of all accessible interior and exterior surfaces. Evidence of concrete degradation such as spalling, cracking, leaching, rust bleed, etc., are documented and evaluated for appropriate corrective action. Assessment of inaccessible surfaces of structures and structural components is based on the results of inspections of accessible areas, as well as site-specific environmental conditions and industry operating experience. The program has been enhanced over the years to include recommendations resulting from Institute of Nuclear Power Operations (INPO) and NRC audits/inspections as well as other industry guidance.

Additionally, the Structures Monitoring Program is subject to periodic internal and external assessments to ensure effectiveness and continuous improvement.

Continued implementation of the Structures Monitoring Program provides reasonable assurance that the effects of aging will be adequately managed for the Containment Structure concrete components so that their intended function(s) will be maintained consistent with the current licensing basis (CLB) for both the extended ILRT test interval and the extended operation of the station.

<u>Question 2:</u>
IWE-1240 requires licensees to identify the containment surface areas requiring augmented examinations. Provide the locations of the steel containment (or concrete containment liner) surfaces that have been identified as requiring augmented examination and a summary of the findings of the examinations performed.

<u>Response 2:</u>
Review of plant-specific operating experience and recent maintenance and corrective action documents identified only one nonconforming condition at the moisture barrier (caulking) which protects the inaccessible portion of the containment steel liner from corrosion. This condition was discovered during inservice inspections performed to meet the requirements of ASME Section XI, Subsection IWE in 2000. The insulation was removed and the liner was exposed for visual inspection in two areas. Evidence of minor surface corrosion was present in the area with the nonconforming caulking detail.

Ultrasonic thickness readings were taken in both areas, including locations above and along the interface between the liner and the containment concrete floor. All measured values exceeded the minimum required thickness with considerable margin. The liner was cleaned, re-coated and the moisture barrier restored in accordance with original design specification requirements in both areas.

As a result of this discovery, the configuration of the moisture barrier was inspected around the entire circumference of the containment and verified to be intact with no visible gaps or discontinuities. Additional inspections of the liner were performed during the 2002 refueling outage. Approximately 70 linear feet of the liner were exposed and ultrasonic thickness measurements taken at four different excavated areas below the floor level. These measurements verified that no loss of liner thickness had occurred at these locations. The exposed portion of the liner was again cleaned, re-coated, and the moisture barrier restored in accordance with original design specification requirements.

Additional inspections of the moisture barrier and liner are planned during the second and third periods of the Fourth ISI interval, which commenced on January 1, 2000. The condition of the inaccessible portions of the containment liner may be assessed by evaluation of the condition of the liner at the interface with the concrete floor. Therefore, inspections performed under the ASME Section XI, Subsections IWE/IWL ISI Program will provide reasonable assurance that aging effects for the inaccessible portions of the liner plate can be managed so that the liner plate will continue to perform its intended function consistent with the current licensing basis during the period of extended operation.

Ginna has committed to the performance of visual inspections and ultrasonic testing thickness measurements of the containment liner during the 2005 refueling outage. *Reference NUREG-1786, "Safety Evaluation Related to the License Renewal of R.E. Ginna Nuclear Power Plant, Docket No. 50-244." Paragraph 3.5.2.2.1.4 and Appendix A Items 23 and 28.*

During the 2003 refueling outage boric acid corrosion was found in the 'A' containment sump. The pre-repair UT examination of the 3" X 3" triangular area determined that in one localized area a wall thickness of 0.09 inches existed. All other adjacent areas were in excess of a wall thickness of 0.09 inches. The minimum wall thickness required to provide an acceptable vapor barrier for this portion of the liner plate is 0.06 inches. The area of corrosion was cleaned, UT inspected, repaired, and repainted.

The In-service Inspection IWE/IWL Program has scheduled the "A" sump for inspection during successive outages within the current 10 year interval.

<u>Question 3:</u>
For the examination of penetration seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary (Examination Categories E-D and E-G), relief for the requirements of the code have been requested. As an alternative, it was proposed to examine them during the leak-rate testing

of the primary containment. However, Option B of Appendix J for Type B and Type C testing (as per NEI 94-01 and RG 1.163), and the ILRT extension requested in this amendment for Type A testing provide flexibility in the scheduling of these inspections. Provide your schedule for examination and testing of seals, gaskets, and bolted connections that provide assurance regarding the integrity of the containment pressure boundary.

Response 3:
For the Fourth Interval Inservice Inspection (ISI) Program, there are two relief requests that defer testing to the Containment Leakage Rate Testing Program as an alternative examination. The subject relief requests are as follows:

Relief Request No. 9, Containment Inspection Seals and Gaskets;
Relief Request No. 11, Containment Inspection Bolt Torque or Tension Testing.

Approval to implement the above relief requests was granted from the NRC by letter dated March 21, 2000. The impact of extending the ILRT frequency on each of these relief requests is discussed as follows:

Relief Request No. 9, Containment Inspection Seals and Gaskets:

Code Requirement:

ASME Section XI code, 1992 Edition, 1992 Addenda, IWE-2500, Table IWE-2500-1, Examination Category E-D, Item Numbers E5.10 and E5.20 requires seals and gaskets on airlocks, hatches, and other devices to be visually examined, VT-3, once each interval to assure containment leak-tight integrity.

Alternative Examinations:

The leak testing of seals and gasket joints will be in accordance with 10CFR50, Appendix J. No additional alternative examinations to the visual examination, VT-3, of seals and gaskets will be performed.

Discussion:

Several hundred seals and gaskets are affected by this relief request. The alternative examination is implemented by the performance of Type B and Type C testing as applicable. The performance of Type B and C testing will continue to be performed in accordance with the Containment Leakage Rate Testing Program and at the intervals specified there in. If a seal or gasket is replaced, it will be visually inspected before reassembly or closure. Also, an as-left Appendix J leakage test will be performed after installation to ensure leak tightness.

Relief Request No. 11, Containment Inspection Bolt Torque or Tension Testing:

Code Requirement:

ASME Section XI code, 1992 Edition, 1992 Addenda, IWE-2500, Table IWE-2500-1, Examination Category E-G, Pressure retaining Bolting, Item Number E8.20. Bolt torque or tension testing is required on bolted connections that have not been disassembled and reassembled during the inspection interval.

Alternative Examinations:

The following examinations and tests required by Subsection IWE ensure the structural integrity and the leak-tightness of Class MC pressure retaining bolting, and, therefore, no additional alternative examinations are proposed:

(1) Exposed surfaces of bolted connections shall be visually examined in accordance with the requirements of Table IWE-2500-1, Examination category E-G, Pressure Retaining bolting, Item Number E8.10, and

(2) Bolted connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, All Pressure Retaining Components, Item Number E9.40, and

(3) A general visual examination of the entire containment once each inspection period shall be conducted in accordance with 10CFR50.55a(b)(2)(x)(E).

Discussion:

Verification of torque or tension values on bolted joints that are proven adequate through Appendix J testing and visual inspection is adequate to demonstrate that design function is met. Torque or tension testing is not required on any other ASME Section XI, Class 1, 2 or 3 bolted connections or their supports as part of the Inservice Inspection Program. Also, all penetrations at R. E. Ginna Nuclear Power Plant are seated with pressure (not unseated).

The one-time extension requested by Ginna applies only to the 10 CFR 50, Appendix J, Type A integrated leak rate test that is currently on a 10-year interval pursuant to Appendix J, Option B, Performance Based Requirements. Appendix J, Type B and Type C tests are performed at the intervals required by Appendix J, Option B and will be tested at least once in the 10-year interval. This frequency of testing of seals, gaskets, and containment pressure retaining bolting provides reasonable assurance that the integrity of the containment pressure boundary is maintained during the period of the extension.

<u>Question 4:</u>

The stainless steel bellows have been found to be susceptible to trans-granular stress corrosion cracking and the leakage through them is not readily detectable by Type B testing (see Information Notice 92-20). If applicable, please provide information regarding inspection and testing of the bellows, and how such behavior has been factored into the risk assessment.

<u>Response 4:</u>

There are no penetration bellows at Ginna which perform a containment isolation function. The bellows are single ply, ASTM A240, Type 304 stainless steel. The only function of the bellows is to accommodate lateral and axial pipe displacements. *Reference NUREG-1786, "Safety Evaluation Related to the License Renewal of R.E. Ginna Nuclear Power Plant, Docket No. 50-244," Paragraph 3.5.2.2.1.7.*

Prior to the performance of Type A testing, the penetration bellows are aligned to their associated mechanical manifolds to permit the monitoring of containment primary barrier welds. The pressure gauge for each manifold will be monitoring a group of penetrations. For those manifolds exhibiting pressure build-up during the ILRT, the penetrations served by those manifolds will be individually checked upon completion of the ILRT and the leakage located and the leak rate determined.
Reference procedure RSSP-6.2, "Pressurization Monitoring of Penetrations During the ILRT."

Question 5:
Inspections of some reinforced concrete and steel containment structures have found degradation on the uninspectable (embedded) side of the drywell steel shell and steel liner of the primary containment. These degradations cannot be found by visual (i.e., VT-I or VT-3) examinations unless they are through the thickness of the shell or liner, or 100% of the uninspectable surfaces are periodically examined by ultrasonic testing. Please provide information (additional analyses) addressing how potential leakage under high pressure during core damage accidents is factored into the risk assessment related to the extension of the ILRT.

Response 5:
The attached "Risk Assessment for R. E. Ginna Nuclear Power Plant Regarding ILRT (Type A) Extension Request" provides a sensitivity evaluation considering potential corrosion impacts within the framework of the ILRT interval extension risk assessment. The analysis confirms that the ILRT interval extension has a minimal impact on plant risk. Additionally, a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel liner also indicate that even with very conservative assumptions, the conclusions from the original analysis would not change. That is, the ILRT interval extension is judged to have a minimal impact on plant risk and is therefore acceptable. The attached analysis also clarifies the delta LERF for the original License Bases "three tests in 10 years" and the proposed "one test in 15 years." The analysis also provides a discussion on the effects ILRT interval extension would have on the total LERF (internal and external events) for Ginna.

2.10 Ginna Specific Risk Results

Based on the results from Attachment 1, "Risk Assessment for R.E. Ginna Nuclear Power Plant Regarding ILRT (Type A) Extension Request," the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test from ten years to fifteen years:

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

- Reg. Guide 1.174 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 10^{-6}/yr and increases in

LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years is 8.0E-09/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the Ginna risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk by only 0.15%. Therefore, the risk impact change when compared to other severe accident risks is negligible.

- The change in Conditional Containment Failure Probability (CCFP) with respect to a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years was calculated as less than 1%. This is judged to be insignificant and reflects sufficient defense-in-depth.

Enclosure 2
R.E. Ginna Nuclear Power Plant

Significant Hazards Consideration Evaluation and Environmental Consideration

Enclosure 2
R. E. Ginna Nuclear Power Plant
Integrated Leakage Rate Testing Interval Extension

Significant Hazards Consideration Evaluation and Environmental Consideration

1.0 Significant Hazards Consideration Evaluation

The proposed change to the Ginna Station Technical Specifications as identified and justified in Enclosure 1 has been evaluated with respect to 10 CFR 50.92(c) and shown not to involve a significant hazards consideration as described below.

In 10 CFR 50.92(c), the NRC provides the following standards to be used in determining the existence of a significant hazards consideration:

> The Commission may make a final determination, pursuant to the procedures in §50.91 that a proposed amendment to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility 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 an 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.

Basis for no significant hazards consideration determination:

1.1 The proposed Technical Specification change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to Technical Specification 5.5.15, Containment Leakage Rate Testing Program, involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test.

The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG-1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. The Ginna ILRT test history supports this conclusion. In NUREG-1493 Section 10, Summary of Technical Findings, it is concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk.

The proposed change does not result in an increase in core damage frequency since the containment system is used for mitigation purposes only. Containment Leakage Rate Testing Program local leak rate test requirements and administrative controls such as design change control, ASME Section XI Inservice Inspection (ISI) Program Containment Repair and Replacement Program and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with the ASME Section XI Inservice Inspection (ISI) Program Containment Program, Boric Acid Corrosion Program, inspections in accordance with Regulatory Guide 1.163 position C.3 and the Maintenance Rule serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

1.2 The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to Technical Specification 5.5.15 involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or changes in the methods in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

1.3 The proposed TS change does not involve a significant reduction in a margin of safety.

The proposed change to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the Primary Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

Ginna and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with the ASME Section XI Inservice Inspection (ISI) Program Containment Program, Boric Acid Corrosion Program, inspections in accordance with Regulatory Guide 1.163 position C.3 and the Maintenance Rule serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained. Therefore, the proposed Technical Specification change does not involve a significant reduction in a margin of safety.

2.0 Environmental Consideration

Ginna LLC has evaluated the proposed Technical Specification change and determined that:

1. The change does not involve a significant hazards consideration as documented above.

2. The change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG-1493, that Type B and C containment leakage

tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. NUREG-1493 also states, in part, that the reduction in frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk.

3. The change does not involve a significant increase in individual or cumulative occupational radiation exposures since no new or different type of equipment are required to be installed as a result of this Licensing Amendment Request, and the frequency of the required testing which may result in radiation exposure is reduced.

Accordingly, the proposed Technical Specification change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

Enclosure 3
R.E. Ginna Nuclear Power Plant

Proposed Technical Specification Change (markup)

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the 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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

Insert 1

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage 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. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$.

c. Mini-purge valve acceptance criteria is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

Insert 1

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

> Section 9.2.3: The first Type A test performed after the May 31, 1996 Type A test shall be performed by May 31, 2011.

Enclosure 4
R.E. Ginna Nuclear Power Plant

Revised Technical Specification Pages

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain:

a. The methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by the onsite review function and the approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment Program

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include Containment Spray, Safety Injection, and Residual Heat Removal in the recirculation configuration. The prog ram shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Deleted

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I and 40 CFR 141;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the reactor coolant system cyclic and transient occurrences specified in UFSAR Table 5.1-4 to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8 Steam Generator (SG) Tube Surveillance Program

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program in accordance with the Nuclear Policy Manual. This inspection program shall define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Code, Section XI, as required by 10 CFR 50.55a(g). The program shall include the following:

a. The inspection intervals for SG tubes shall be specified in the Inservice Inspection Program.

b. SG tubes that have imperfections > 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.

c. SG sleeves that have imperfections > 30% through wall, as indicated by eddy current, shall be repaired by plugging.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented. The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

a. Containment Recirculation Fan Cooler System

1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

b. Control Room Emergency Air Treatment System (CREATS)

1. Demonstrate the pressure drop across the combined HEPA filters, the prefilters, the charcoal adsorbers and the post-filters is < 11 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass < 0.05%.

3. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 0.05%, when tested under ambient conditions.

4. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 1.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F), a relative humidity of 95%, and a face velocity of 61 ft/min.

c. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.

2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in waste gas decay tanks. The gaseous radioactivity quantities shall be determined following the methodology in NUREG-0133.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance with proper color; and

b. Within 31 days following addition of the new fuel to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b.1 or Specification 5.5.13.b.2 shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71e.

5.5.14 <u>Safety Function Determination Program (SFDP)</u>

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

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

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

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.15 <u>Containment Leakage Rate Testing Program</u>

A program shall be established to implement the leakage rate testing of the 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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

> Section 9.2.3: The first Type A test performed after the May 31, 1996 Type A test shall be performed by May 31, 2011.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage 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. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$.

c. Mini-purge valve acceptance criteria is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

The provisions of SR 3.0.3 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

Attachment 1

R.E. Ginna Nuclear Power Plant

Risk Assessment for R. E. Ginna Nuclear Power Plant
Regarding ILRT (Type A) Extension Request

Risk Assessment for R. E. Ginna Nuclear Power Plant

Regarding ILRT (Type A) Extension Request

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Risk Assessment for R. E. Ginna Nuclear Power Plant
Regarding ILRT (Type A) Extension Request

1.0 PURPOSE OF ANALYSIS

1.1 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) frequency from ten years to fifteen years for a one time extension for Ginna. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for Ginna. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174 [3]. The proposed change would impact testing associated with the current surveillance test for Type A leakage, Ginna procedure RSSP-6.0, "Containment Integrated Leakage Rate Test [24]. "

1.2 BACKGROUND

10CFR50, Appendix J, Option B, allows individual plants to extend the Type A Integrated Leak Rate Test (ILRT) surveillance test interval from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated leakage performance was less than 1.0 La. Ginna meets these requirements.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rule making to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rule making basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285 [2].

The NRC report, Performance Based Leak Test Program, NUREG-1493 [4], which analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing determined that increasing the containment leak rate from the nominal 1.0 percent per day to 10 percent per day leads to a small increase in total population exposure. In addition, increasing the leak rate to 100 percent per day increases the total population risk by less than 1.0 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. This analysis is being performed to confirm these conclusions based on Ginna specific models and available data.

EPRI TR-104285 (Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals) is a follow-on report to NUREG-1493 that provides a methodology for use in preparing PRA analysis to support a submittal. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct a general visual inspection of the accessible areas of the interior of the containment in accordance with Subsection IWE once each period. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B and Type C local leak tests performed to verify the leak-tight integrity of containment penetration valves, air locks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.3 CRITERIA

The acceptance guidelines in RG 1.174 [3] are used to access 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 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also 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 which helps to ensure that the defense-in-depth philosophy is maintained will also be calculated.

In addition, the total risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter.

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in EPRI TR-104285 [2] and NUREG-1493 [4]. The analysis uses the current Ginna Probabilistic Risk Assessment (PRA) model that includes the results from the Ginna Level 2 analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release).

The four general steps of this risk assessment are as follows:

1) Quantify the baseline risk in terms of frequency events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.

2) Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses (i.e., previously performed SAMA calculations using MACCS2 [22]).

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

4) Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with Regulatory Guide 1.174 [3] and compare with the acceptance guidelines of RG 1.174.

This approach is based on the information and methodology contained in the previously mentioned studies and further is consistent with the following:

- Other industry risk assessments of extending the ILRT test interval. The Ginna assessment uses population dose as one of the risk measures. The other risk measures used in the Ginna assessment are Large Early Release Frequency (LERF), and Conditional Containment Failure Probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.

- EPRI TR-104285 and NUREG-1493. The Ginna assessment uses information from NUREG-1273 [6] regarding the low percentage of containment leakage events that would only be detected by an ILRT as input to calculate the increase in the pre-existing containment leakage probability due to the testing interval extension.

- The approach used in the Indian Point 3 risk-informed submittal for a one-time extension of the Type A test interval. The Ginna evaluation uses similar ground rules and methods to calculate changes in risk metrics [14]. NRC approval was granted to Indian Point 3 on April 17, 2001 (TAC No. MB0178) [21].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The Ginna Level 1 and Level 2 internal events PRA model provides representative results for the analysis. The Ginna Level 1 PRA includes fire, floods, and shutdown events.

- It is appropriate to use the Ginna internal events PRA model as a gauge to effectively describe the risk changes attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire, floods and shutdown events were to be included in the calculations.

- An evaluation of the risk trade-off impact of performing the ILRT during shutdown is addressed using the generic results from EPRI TR 105189 [10].

- Dose results for the containment failures modeled in the PRA can be characterized by the Ginna population dose results from MACCS2 calculations such as performed for SAMA [22].

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.

- The maximum containment leakage for Class 1 sequences is 1.0 L_a. Class 3 accounts for increased leakage due to Type A inspection failures.

- The maximum containment leakage for Class 3a sequences is 10 L_a based on the previously approved methodology [14, 21].

- The maximum containment leakage for Class 3b sequences is 35 L_a based on the previously approved methodology [14, 21].

- The impact on population doses from Interfacing System LOCAs (ISLOCAs) 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 ISLOCA contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this assumption.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal. Containment isolation valves that fail to close during an accident and in response to a containment isolation signal are calculated on a Ginna specific basis and made part of the overall population dose and LERF calculations.

- In general, the text describes values to two significant digits. The actual calculations are performed using spreadsheets.

4.0 INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 GENERAL RESOURCES AVAILABLE

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

1) NUREG/CR-3539 [7]
2) NUREG/CR-4220 [8]
3) NUREG-1273 [6]
4) NUREG/CR-4330 [9]
5) EPRI TR-105189 [10]
6) NUREG-1493 [4]
7) EPRI TR-104285 [2]

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

NUREG/CR-3539 [7]

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

NUREG/CR-4220 [8]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. It is the latter category that is applicable to containment isolation modeling that is the focus of this risk assessment.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events.

NUREG-1273 [6]

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

NUREG/CR4330 [9]

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

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

EPRI TR-105189 [10]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the benefit from extending the ILRT frequency from 3 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for drain down events
- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS-draindown events caused by ILRT/LLRT activities, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in safety due to extending the ILRT test interval.

NUREG-1493 [4]

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2-1.0%) population risk.

NUREG-1493 used information from NUREG-1273 regarding the low percentage of containment leakage events that would only be detected by an ILRT in the calculation of the increase in the pre-existing containment leakage probability due to the testing interval extension. NUREG-1493 makes the following assumptions in this probability calculation:

- The average time that a pre-existing leakage may go undetected increases with the length of the testing interval (and is ½ the length of the test interval).
- Only 3% of all pre-existing leaks can be detected only by an ILRT (i.e., and not by LLRTs).

This same approach that was used in a previously approved ILRT test interval extension submittal [14, 21] is also proposed here for the Ginna ILRT test interval extension risk assessment.

EPRI TR-104285 [2]

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

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage frequencies into eight (8) classes of containment response to a core damage accident.

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Containment isolation failures not identified by LLRT (e.g., isolation failures due to testing or maintenance)
7. Containment failure due to core damage accident phenomena
8. Containment bypass

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

> "These study results show that the proposed CLRT [containment leak rate tests] frequently changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year..."

4.2 PLANT SPECIFIC INPUTS

The information used to perform the Ginna ILRT Extension Risk Assessment includes the following:

- Population Dose Calculations by release category (e.g., MACCS2 code calculation results).
- Ginna PRA Model
- ILRT results to demonstrate adequacy of the administrative and hardware issues. The two most recent Type A ILRT tests for Ginna were successful, so the current Type A test interval is 10 years.

4.2.1 Population Dose Calculations

The population dose is calculated from MACCS2 calculations performed for the Ginna License Renewal Environmental Report [22]. Table 4.2-1 summarizes the calculated population doses at 50 miles for each release category defined in the report.

Table 4.2-1

SUMMARY OF POPULATION DOSE BASED ON GINNA LICENSE RENEWAL ENVIRONMENTAL REPORT [22]

Release Category	Description	Population Dose at 50 miles (person-rem)
1	Intact Containment	2.27E+04
2	Interfacing Systems LOCA	1.76E+07
3	Loss of Containment Isolation	3.38E+06
4	Steam Generator Tube Rupture (Wet)	1.15E+06
5	Steam Generator Tube Rupture (Dry)	4.62E+06
6	Steam Generator Tube Rupture (ARV Cycle)	6.89E+05
7	Late Global Containment Failure	9.39E+05
8	Late Small Containment Failure	4.51E+05
9	Thermally Induced Tube Rupture	4.72E+06
10	High Pressure Reactor Coolant Break	1.36E+06
11	Low Pressure Reactor Coolant Break	1.94E+05

4.2.2 Ginna PRA Model

Revision 4.1 of the Ginna PRA Model was used to quantify frequencies for release categories used in the Ginna License Renewal Environmental Report [22]. For the ILRT Extension, the frequencies for the release categories on Table 4.2-1 were calculated using Revision 4.3 of the Ginna PRA Model. The release category frequencies were then multiplied by the associated population doses to obtain population dose risk. A summary of these calculations is provided in Table 4.2-2. For Revision 4.3 of the Ginna PRA model, the core damage frequency (CDF) is 5.37E-5 per-yr and the Large Early Release Frequency (LERF) is 6.44E-06 per-yr.

Table 4.2-2

SUMMARY OF POPULATION DOSE RISK CALCULATIONS BASED ON GINNA LICENSE RENEWAL ENVIRONMENTAL REPORT [22] AND REVISION 4.3 OF THE GINNA PRA MODEL

Release Category	Description	Frequency	Population Dose at 50 miles (person-rem)	Population Dose Risk at 50 miles (person-rem/yr)
1	Intact Containment	4.73E-05	2.27E+04	1.074
2	Interfacing Systems LOCA	2.25E-07	1.76E+07	3.960
3	Loss of Containment Isolation	4.70E-07	3.38E+06	1.589
4	Steam Generator Tube Rupture (Wet)	4.14E-06	1.15E+06	4.761
5	Steam Generator Tube Rupture (Dry)	0.00E+00	4.62E+06	0.000
6	Steam Generator Tube Rupture (ARV Cycle)	3.80E-10	6.89E+05	0.000
7	Late Global Containment Failure	1.00E-06[1]	9.39E+05	0.939
8	Late Small Containment Failure	1.00E-06[1]	4.51E+05	0.451
9	Thermally Induced Tube Rupture	2.18E-07	4.72E+06	1.029
10	High Pressure Reactor Coolant Break	3.55E-07	1.36E+06	0.483
11	Low Pressure Reactor Coolant Break	1.58E-08	1.94E+05	0.003
Total		**5.37E-05**		**14.289**

1) This frequency is the sum of both global and small containment failures.

4.2.3 Release Category Definition

Table 4.2-3 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI methodology [2].

Table 4.2-3
EPRI 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 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).
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.

4.3 CONDITIONAL PROBABILITY OF ILRT FAILURE (SMALL AND LARGE)

The ILRT can detect a number of failures such as liner breach, failure of certain bellows arrangements, and failure of some sealing surfaces. The proposed ILRT test interval extension may influence the conditional probability associated with the ILRT failure. To ensure that this effect is properly accounted for, the Class 3 Accident Class is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

To calculate the probability that a liner leak will be large (Event CLASS-3B), use was made of the data presented in NUREG-1493 [4]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Because 21L_a does not constitute a large release, no releases have occurred based on the 144 ILRTs reported in NUREG-1493 [4]. The EPRI Interim Guidance [23] reported 1 ILRT failure in 38 ILRTs. This failure does not constitute a large release.

Using the approach provided the EPRI methodology [2], a bayesian update of a non-uniform prior was used to estimate the conditional probability of ILRT failure.

$$\text{Probability}_{3b} = (2n+1)/2N$$

Where:
n = Number of ILRT failures
N = Number of ILRT tests

For n = 0 and N= (144+38) = 182

$$\text{Probability}_{3b} = (2*0 + 1)/ (2*182) = 0.002747$$

To calculate the probability that a liner leak will be small (Event CLASS-3a), use was made of the data presented in NUREG-1493 [4]. The data found in NUREG-1493 states that 144 ILTRs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of 1.0 L_a. However, of these "failures" only 4 were found by an ILRT; the others were found by Type B and C testing on errors in test alignments. Therefore, the number of failures considered for "small releases" are 4-of-144. The EPRI Interim Guidance [23] stated that one failure found by an ILRT was found in 38 ILRTs performed after NUREG-1493.

For n = 5 and N= (144+38) = 182

$\text{Probability}_{3a} = (2*5 + 1)/(2*182) = 0.03$

4.4 IMPACT OF EXTENSION ON LEAK DETECTION PROBABILITY

The NRC in NUREG-1493 [4] has determined from a review of operating experience data that only 3% of the ILRT failures were found which local leakage-rate testing could not and did not detect. Further, in NUREG-1493 it is noted that the leakage rates observed in these few Type A test failures were only marginally above currently prescribed limits and could be characterized by a leakage rate of about two times the allowable.

Also in NUREG-1493 [4], it was assumed that the characteristic magnitude of leakages detectable only by ILRTs would not change, but the probability of leakage would change due to the longer intervals between tests. The change in probability was 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 (3yrs/2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10yrs/2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by

ILRT testing. However, since ILRTs have been demonstrated to improve the residual leak detection by only 3%, the interval change noted above would only lead to about a 10% (3.33 x 3%) non-detection leak probability. It is assumed that Local Leak Rate Test (LLRT) will continue to provide leak detection for the 97% of leakages. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a 15% (7.5/1.5x3%) non-detection probability of a leak. These are obviously approximations assumed by the NRC and EPRI because the current 3 ILRTs in 10 years would have a T/2 = 1.67 years instead of 1.5 years

Therefore, the failure rate of ILRTs for which the LLRTs do not provide adequate backup is 0.03/1.5 year average detection time. Applying a constant failure rate model, the failure probability of ILRTS, P_f, can be estimated as follows:

For 3 Year Interval

$$P_f = \frac{1}{2}\lambda T = \frac{0.03}{1.5\ yr} * \frac{3\ yrs}{2} = 0.03$$

For 10 Year Interval

$$P_f = \frac{1}{2}\lambda T = \frac{0.03}{1.5 yr} * \frac{10\ yrs}{2} = 0.10$$

For 15 Year Interval

$$P_f = \frac{1}{2}\lambda T = \frac{0.03}{1.5 yr} * \frac{15\ yrs}{2} = 0.15$$

EPRI has previously interpreted this to mean that the failure to detect probability values are tabulated as follows:

ILRT FAILURE TO DETECT PROBABILITY

ILRT Interval	EPRI Assessment [2]	IP3 [14]	Constant Failure Rate Model
3 yr	0.03	0.03	0.03
10 yr	0.13	0.13	0.10
15 yr	NA	0.18	0.15

In addition, IP3 [14] has used this same estimate of changes in detection probability in a submittal to extend the ILRT interval on a one-time basis. The IP3 request for a one-time ILRT extension was approved by the NRC on April 17, 2000 (TAC No. MB0178) [21].

The analysis included in this report follows the precedence set by the EPRI report and the IP3 analysis. The use of the constant failure rate model is conservatively represented by the assumed "failure to detect" probabilities used by EPRI and in the IP3 submittal.

5.0 RESULTS

The application of the approach based on EPRI-TR-105189 [10] and a previous risk assessment submittal on this subject [14] has established a clear process for the calculation and presentation of results.

The method chosen to display the results is according to the eight (8) accident classes consistent with these two reports. Table 5-1 lists these accident classes.

The analysis performed examined Ginna specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accident contribution to risk was considered in the following manner:

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

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

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences).

- Accidental sequences involving containment bypass (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile.

However, they are not affected by the ILRT frequency change.

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

Table 5-1

ACCIDENT CLASSES

Accident Classes (Containment Release Type)	DESCRIPTION
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal-Type B)
5	Small Isolation Failures (Failure to seal-Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

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

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the applicable eight accident classes presented in Table 5-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.

Step 3 - Evaluate the risk impact of extending Type A test interval from 10 to 15 years.

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

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

The severe accident sequence frequencies that can result in offsite consequences are evaluated. Revision 4.3 of the Ginna PRA model [25] as documented by Ginna LLC is used in the ILRT evaluation.

This step involves the review of the Ginna License Renewal Environmental Report [22] to establish the mapping of release categories to population dose at 50 miles. Revision 4.3 of the Ginna PRA model was used to re-calculate the release category frequencies in the Ginna License Renewal Environmental Report since the original calculations were based on Revision 4.1. The results of these calculations were provided on Table 4.2-2

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

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks are included in the model. Specifically, a simplified model based on NUREG-1493 results is used to predict the likelihood of having a small/large breach in the containment liner that is undetected by the Type A ILRT test. These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]. The Class 3 leakage includes the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered to ensure proper representation of available data. These are Event Class-3a (small breach) and Event Class-3b (large breach).

After including the respective "large" and "small" liner breach leak rate probabilities (Classes 3a and 3b), the eight severe accidents class frequencies were developed consistent with the definitions in Table 5-1 and described below.

Class 1 Sequences.
This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). For this analysis, the associated maximum containment leakage for this group is 1.0 L_a, consistent with an intact containment evaluation.

The Class 1 frequency is calculated as follows:

$$F_{Class\ 1} = CDF - (F_{Class\ 2} + F_{Class3} + F_{Class4} + F_{Class5} + F_{Class6} + F_{Class7} + F_{Class8})$$

Where:
CDF = Core Damage Frequency
$F_{Class\ 2}$ = Class 2 frequency (per yr)
$F_{Class\ 3}$ = Class 3 frequency (per yr)
$F_{Class\ 4}$ = Class 4 frequency (per yr)
$F_{Class\ 5}$ = Class 5 frequency (per yr)
$F_{Class\ 6}$ = Class 6 frequency (per yr)
$F_{Class\ 7}$ = Class 7 frequency (per yr)
$F_{Class\ 8}$ = Class 8 frequency (per yr)

Class 2 Sequences.
This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. These sequences are dominated by failure-to-close of large containment isolation valves. The frequency per year for these sequences is 4.70E-7/year and is determined by the frequency of Release Category 3 on Table 4.2-2.

Class 3 Sequences.

This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (2.0 L_a to 35 L_a) or large (>35 L_a).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.03 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.002747 [see Section 4.3]

CLASS_3a_FREQUENCY = 0.03 * 5.37E-5/year = 1.61E-6/year

CLASS_3b_FREQUENCY = 0.002747 * 5.37E-5/year = 1.48E-7/year

For this analysis, the associated containment leakage for Class 3a is 10 L_a and for Class 3b is 35 L_a. These assignments are consistent with the Indian Point 3 ILRT submittal [14] which was approved by the NRC [21].

Class 4 Sequences.

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

Class 5 Sequences.

This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the 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 6 Sequences.

This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution.

The low failure probabilities for this class are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the fact that this failure class is not impacted by Type A testing, a screening value is considered appropriate for this low probability failure mode. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology (i.e. $PROB_{class6}>0$), a conservative screening value of 4.0E-4 will be used to evaluate this class as described below.

The frequency per year for these sequences is determined as follows:

$CLASS\_6\_FREQUENCY = PROB_{largeT\&M} * CDF$

Where:

$PROB_{largeT\&M}$ = random large containment isolation failure probability due to valve misalignment is is estimated using NUREG/CR1278

= 4.0E-4

CLASS_6_FREQUENCY = 4.0E-4 * 5.37E-5/year
= 2.15E-8/year

For this analysis the associated containment leakage for this group is represented by the direct release from containment, i.e., Class 2 consequences are assigned.

Class 7 Sequences.
This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., direct containment heating, melt-through, overpressure). The baseline frequency per year for these sequences is 1.39E-6/year and is determined by the sum of the frequencies for Release Categories 7[1], 8[1], 10, and 11 on Table 4.2-2. For 10-yr and 15-yr test intervals, there is a likelihood that a corrosion related containment leakage may not be detected. Therefore, the baseline frequency for Class 7 sequences is increased by a factor of 1.00163 to account for undetected corrosion related containment leakage. See Appendix A for basis and supporting calculations. Note that this factor is conservatively based on a test interval increase from 3 years to 15 years and is used for the 10-year and 15-year cases.

Class 8 Sequences.
This group consists of all core damage accident progression bins in which containment bypass occurs. The frequency per year for these sequences is 4.58E-6/year and is determined by the sum of the frequencies for Release Categories 2, 4, 5, 6, and 9 on Table 4.2-2.

Summary of Accident Class Frequencies
In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definition of Accident Classes defined

in EPRI-TR-104285. Table 5-2 summarizes these accident frequencies by Accident Class.

Table 5-2

RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	4.55E-05	84.69%
2	Large Isolation Failures (Fail to Close)	4.70E-07	0.87%
3A	Small Isolation Failures (Liner Beach)	1.61E-06	3.00%
3B	Large Isolation Failures (Liner Beach)	1.48E-07	0.27%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00%
6	Other Isolation Failures (e.g., dependent failures)	2.15E-08	0.04%
7	Failures induced by Phenomena (early and late)	1.39E-06	2.59%
8	Bypass (Interfacing Systems LOCA)	4.58E-06	8.53%
Total		**5.37E-05**	

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

Plant-specific release analysis was performed to evaluate the person-rem doses to the population, within a 50-mile radius from the plant. The releases are based on MACCS2 calculations for Ginna that were also used to support the Ginna License Renewal evaluation and submittal [22].

From the data section of this calculation, the person-rem (population dose) taken out to 50 miles is based on Ginna specific MACCS2 calculations documented in the Ginna

License Renewal Environmental Report [22] and as shown on Table 4.2-2.

Class 1 = 2.27E+4 person-rem (at 1.0 L_a)

Class 2 = 3.38E+6

Class 3a = 2.27E+4 person-rem x 10 L_a = 2.27E+5 person-rem

Class 3b = 2.27E+4 person-rem x 35 L_a = 7.95E+5 person-rem

Class 4 = Not analyzed (Assigned a zero value)

Class 5 = Not analyzed (Assigned a zero value)

Class 6 = 3.38E+6 person-rem (Assumed a Class 2 release)

Class 7 = 1.04E+06 person-rem (sum of the population dose risk for Release Categories 7, 8, 10, and 11 on Table 4.2-2 divided by the frequency for EPRI Class 7 on Table 5-2)

Class 8 = 2.13E+6 person-rem (sum of the population dose risk for Release Categories 2, 4, 5, 6, and 9 on Table 4.2-2 divided by the frequency for EPRI Class 8 on Table 5-2)

The population dose estimates derived for use in the risk evaluation are summarized in Table 5-3.

Table 5-3

GINNA POPULATION DOSE ESTIMATES FOR POPULATION WITH 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	2.27E+4
2	Large Isolation Failures (Failure to Close)	3.38E+6
3a	Small Isolation Failures (liner breach)	2.27E+5
3b	Large Isolation Failures (liner breach)	7.95E+5
4	Small Isolation Failures (Failure to seal-Type B)	0
5	Small Isolation Failures (Failure to seal-Type C)	0
6	Other Isolation Failures (e.g., dependent failures)	3.38E+6
7	Failures Induced by Phenomena	1.04E+06
8	Bypass (Interfacing System LOCA)	2.13E+6

The above results, when combined with the results presented in Table 5-2, yield the baseline mean consequence measures for each accident class. These results are presented in Table 5-4.

Table 5-4

ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS (I.E., REPRESENTATIVE OF ILRT DATA)

EPRI Accident Class	**Description**	**EPRI Accident Class Frequency (per year)**	**Population Dose at 50 Miles (person-rem)**	**Population Dose Rate at 50 Miles (person-rem/yr)**
1	No Containment Failure	4.55E-05	2.27E+04	1.033
2	Large Isolation Failures (Fail to Close)	4.70E-07	3.38E+06	1.589
3a	Small Isolation Failures (Liner Beach)	1.61E-06	2.27E+05	0.366
3b	Large Isolation Failures (Liner Beach)	1.48E-07	7.95E+05	0.117
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000
6	Other Isolation Failures (e.g., dependent failures)	2.15E-08	3.38E+06	0.073
7	Failures induced by Phenomena (early and late)	1.39E-06	1.04E+06	1.444
8	Bypass (Interfacing Systems LOCA)	4.58E-06	2.13E+06	9.750
Total		**5.37E-05**		**14.371**

The total dose per year is compared with the other sites as shown below:

Plant	Annual Dose (Person-Rem/yr)	Reference
Indian Point 3	14.515	14
Peach Bottom	6.2	15
Crystal River	1.4	16
Ginna	14.371	Table 5-4

Based on the risk values from Table 5-4, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 (i.e., the Class affected by the ILRT interval change) is as follows:

$\%Risk_{BASE}$ = [(CLASS $3a_{BASE}$ + CLASS $3b_{BASE}$)/ $Total_{BASE}$] x 100

Where:

$CLASS3a_{BASE}$ = Class 3a person-rem/year = 3.66E-1 person-rem/year [Table 5-4]

$CLASS3b_{BASE}$ = Class 3b person-rem/year = 1.17E-1 person-rem/year [Table 5-4]

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval
= 14.371 person-rem/yr [Table 5-4]

$\%Risk_{BASE}$ = [(0.366 + 0.117)/14.371] x 100 = [0.483/14.371] x 100

$\%Risk_{BASE}$ = 3.36%

5.3 STEP 3 – EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEAR

According to NUREG-1493 [4], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10-years will increase the average time that a leak detectable only by an ILRT goes undetected from 1.5 years to 5 years. The average time for failure to detect is calculated using the approximation ½ λT where T is the Test interval and λ, the leakage failure rate, is (3%)/1.5 year (see section 4.4). Because ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% undetectable rate in the overall probability of leakage (½ * (3% /1.5 years) * 10 years).

For a 15-yr-test interval, the result is a 15% overall probability of leakage (i.e., ½ * (3% /1.5 yrs) * 15 years). Thus, increasing the ILRT test interval from 10 years to 15 years translates into a 5% increase in the overall leakage probability.

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (i.e. a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 1.1 (The factor of 1.1 is based on a test interval of 1.5 years. This is an approximation since 3 tests in 10 years is really a test interval of 1.66 years (10years/3 tests). If 1.66 is used instead of 1.5, then the factor is 1.07.). Specifically, there is a factor of 1.1 increase in the Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 10 yrs.

Risk Impact of Corrosion Related Leakage due to Increase from 3 to 15-year Test Interval

Increasing the test interval from 3 to 15 years may reduce the chance of detecting corrosion related leakage. The likelihood of not detecting corrosion related leakage due to increased test interval from 3 to 15 years is calculated to be 0.163%. Details of this calculation are provided in Appendix A. The calculation assumes that the total containment surface area below the containment spring-line (i.e., the dome-cylinder interface) can be exposed to corrosion. The increased likelihood of corrosion related leakage is assumed to increase LERF frequency contributions from phenomena-related accident sequences (EPRI Class 7) by a factor of 1.00163. This factor is applied to both 10-year and 15-year test interval calculations.

The results of this calculation are presented in Table 5-5. Based on the Table 5-5 values, the Type A 10-year test frequency percent risk contribution ($\%Risk_{10}$) for Class 3 is as follows:

$$\%Risk_{10} = [(CLASS3a_{10} + CLASS3b_{10}) / Total_{10}] \times 100$$

Where:

$CLASS\ 3a_{10}$ = Class 3a person-rem/year = 4.03E-1 person-rem/year [Table 5-5]

$CLASS\ 3b_{10}$ = Class 3b person-rem/year = 1.29E-1 person-rem/year [Table 5-5]

$TOTAL_{10}$ = Total person-rem/yr for 10-year interval = 14.416 person-rem/yr [Table 5-5]

$\%Risk_{10} = [(0.403 + 0.129) / 14.416] \times 100 = [0.532/14.416] \times 100$

$\% Risk_{10} = 3.69\%$

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 3.69%.

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is as follows:

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$$

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 14.371 person-rem/yr [Table 5-4]

$TOTAL_{10}$ = Total person-rem/yr for 10 yr ILRT interval = 14.416 person-rem/yr [Table 5-5]

$$\Delta\%Risk_{10} = [(14.416 - 14.371) / 14.371] \times 100.0$$

$$\Delta\%Risk_{10} = 0.31\%$$

Therefore, the increase in risk contribution because of the change to the already approved ten-year ILRT test frequency from three-in-ten years to one-in-ten-years is 0.31%.

Table 5-5

ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED EVERY 10 YEARS

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	4.53E-05	2.27E+04	1.029
2	Large Isolation Failures (Fail to Close)	4.70E-07	3.38E+06	1.589
3a	Small Isolation Failures (Liner Beach)	1.77E-06	2.27E+05	0.403
3b	Large Isolation Failures (Liner Beach)	1.62E-07	7.95E+05	0.129
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000
6	Other Isolation Failures (e.g., dependent failures)	2.15E-08	3.38E+06	0.073
7	Failures induced by Phenomena (early and late)	1.39E-06	1.04E+06	1.444
8	Bypass (Interfacing Systems LOCA)	4.58E-06	2.13E+06	9.750
Total		**5.37E-05**		14.416

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is 15 percent or 1.15 consistent with previously approved method [14, 21]. Specifically, there is a factor of 1.15 increase in Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 15 yrs. (See Section 4.4) The results for this calculation are presented in Table 5-6.

Based on the values from Table 5-6, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class 3 is as follows:

$\%Risk_{15} = [(CLASS3a_{15} + CLASS3b_{15}) / Total_{15}] \times 100$

Where:

$CLASS\ 3a_{15}$ = Class 3a person-rem/year = 4.21E-1 person-rem/year [Table 5-6]

$CLASS\ 3b_{15}$ = Class 3b person-rem/year = 1.35E-1 person-rem/year [Table 5-6]

$TOTAL_{15}$ = Total person-rem/yr for 15-year interval
= 14.438 person-rem/yr [Table 5-6]

$\%Risk_{15} = [(0.421 + 0.135)/14.438] \times 100 = [0.556/14.438] \times 100$

$\%Risk_{15} = 3.85\%$

Therefore, the Total 15-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 3.85%.

The percent increase on the total integrated plant risk when the ILRT is extended from 10 years to 15 years is computed as follows:

$\%TOTAL_{10\text{-}15} = [(TOTAL_{15} - TOTAL_{10}) / TOTAL_{10}] \times 100$

Where:

$TOTAL_{10}$ = Total person-rem/year for 10-year interval
= 14.416 person-rem/year [Table 5-5]

$TOTAL_{15}$ = Total person-rem/year for 15-year interval
= 14.438 person-rem/year [Table 5-6]

$\%TOTAL_{10\text{-}15} = [(14.438 - 14.416) / 14.416] \times 100$

$\%TOTAL_{10\text{-}15} = 0.15\%$

Therefore, the risk impact on the total plant risk for these accident sequences, as influenced by Type A testing, is only 0.15%.

The percent risk increase ($\Delta Risk_{15}$) due to a fifteen-year ILRT over the baseline is as follows:

$\%\Delta Risk_{15} = [(Total_{15} - Total_{BASE} / Total_{BASE}] \times 100$

Where:

$TOTAL_{BASE}$ = Total person-rem/year for baseline interval
= 14.371 person-rem/year [Table 5.4]

$TOTAL_{15}$ = Total person-rem/year for 15-year interval
= 14.438 person-rem/year [Table 5-6]

$\%\Delta Risk_{BASE\text{-}15} = [(14.438 - 14.371) / 14.371] \times 100$

$\%\Delta Risk_{BASE\text{-}15} = 0.46\%$

Therefore, the total increase in risk contribution associated with relaxing the ILRT test frequency from three-in-ten-years to one-per-fifteen years is 0.46%

Table 5-6

ANNNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED EVERY 15 YEARS

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	4.52E-05	2.27E+04	1.027
2	Large Isolation Failures (Fail to Close)	4.70E-07	3.38E+06	1.589
3a	Small Isolation Failures (Liner Beach)	1.85E-06	2.27E+05	0.421
3b	Large Isolation Failures (Liner Beach)	1.70E-07	7.95E+05	0.135
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000
6	Other Isolation Failures (e.g., dependent failures)	2.15E-08	3.38E+06	0.067
7	Failures induced by Phenomena (early and late)	1.39E-06	1.04E+06	1.444
8	Bypass (Interfacing Systems LOCA)	4.58E-06	2.13E+06	9.750
Total		**5.37E-05**		14.438

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

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. Class 3b is treated in this analysis as a potential LERF contributor. Class 3a is not treated a "large" release. Therefore, for this evaluation, only Class 3b sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Other accident classes such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF.

Reg. Guide 1.174[3] 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 core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

<u>Baseline (3 Yr Test Interval) LERF</u>

The baseline LERF frequency (with undetected leakage- Class 3B) is calculated as follows:

$$LERF_{BASE} = CDF_{BASE} - F_{CLASS\ 1\_BASE} - F_{CLASS\ 3a\_BASE}$$

Where:

$LERF_{BASE}$ = Base LERF frequency (per-yr)

CDF_{BASE} = Base CDF frequency (per-yr) = 5.37E-05/yr [Table 5.4]

$F_{CLASS\ 1\_BASE}$ = Base Class 1 frequency (per-yr) = 4.55E-05/yr [Table 5.4]

$F_{CLASS\ 3a\_BASE}$ = Base Class 3a frequency (per-yr) = 1.61E-06/yr [Table 5.4]

$LERF_{BASE}$ = 5.37E-05 - 4.55E-05 - 1.61E-06 = 6.613E-06 per-yr

LERF for 10-Yr Test Interval

The LERF frequency for a 10-yr test interval is calculated as follows:

$$LERF_{10} = (CDF_{10} - F_{CLASS\ 1\_10} - F_{CLASS\ 3a\_10})$$

Where:

$LERF_{10}$ = LERF frequency for 10-yr test interval (per-yr)

CDF_{10} = CDF frequency for 10-yr test interval (per-yr)
= 5.37E-05/yr [Table 5.5]

$F_{CLASS\ 1\_10}$ = Class 1 frequency for 10-yr test interval (per-yr)
= 4.53E-05/yr [Table 5.5]

$F_{CLASS\ 3a\_10}$ = Base Class 3a frequency for 10-yr test interval (per-yr)
= 1.77E-06/yr [Table 5.5]

$LERF_{10}$ = 5.37E-05 - 4.53E-05 - 1.77E-06 = 6.630E-06 per-yr

The LERF increase ($\Delta LERF_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is as follows:

$$\Delta LERF_{BASE\text{-}10} = LERF_{10} - LERF_{BASE}$$

$\Delta LERF_{BASE\text{-}10}$ = 6.630E-06/yr – 6.613E-06/yr = 1.7E-08/yr

LERF for 15-Yr Test Interval

The LERF frequency for a 15-yr test interval is calculated as follows:

$$LERF_{15} = (CDF_{15} - F_{CLASS\ 1\_15} - F_{CLASS\ 3a\_15})$$

Where:

$LERF_{15}$ = LERF frequency for 15-yr test interval (per-yr)
CDF_{15} = CDF frequency for 15-yr test interval (per-yr)
= 5.37E-05/yr [Table 5.6]
$F_{CLASS\ 1\_15}$ = Class 1 frequency for 15-yr test interval (per-yr)
= 4.52E-05/yr [Table 5.6]
$F_{CLASS\ 3a\_15}$ = Base Class 3a frequency for 15-yr test interval (per-yr)
= 1.85E-06/yr [Table 5.6]

$LERF_{15}$ = 5.37E-05 - 4.52E-05 - 1.85E-06 = 6.638E-06 per-yr

The LERF increase ($\Delta LERF_{BASE-15}$) due to a 15-year ILRT over the baseline is as follows:

$\Delta LERF_{BASE-15} = LERF_{15} - LERF_{BASE}$
$\Delta LERF_{BASE-15}$ = 6.638E-06/yr – 6.613E-06/yr = 2.44E-08/yr

The LERF increase ($\Delta LERF_{10-15}$) due to a 15-year ILRT over the 10-yr ILRT is as follows:

$\Delta LERF_{10-15} = LERF_{15} - LERF_{10}$
$\Delta LERF_{10-15}$ = 6.638E-06 – 6.630E-06 = 8.0E-09

It should be noted that the calculated changes in LERF for all cases are well below the 1.0E-7/yr screening criterion in Reg. Guide 1.174 and represents a very small change in risk.

5.5 IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

Another parameter that the NRC Guidance Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in 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).

Because the only classes that are increasing are Classes 3a and 3b, the change in CCFP can be calculated by the difference in these classes.

The percent increase in CCFP increase ($\Delta\%CCFP_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is a follows:

$$\Delta\%CCFP_{BASE\text{-}10} = [((F_{CLASS\ 3a\_10} + F_{CLASS\ 3b\_10}) - (F_{CLASS\ 3a\_BASE} + F_{CLASS\ 3a\_BASE})) / CDF] \times 100$$

$$\Delta\%CCFP_{BASE\text{-}10} = [((1.77E\text{-}06 + 1.62E\text{-}07)\ \text{[Table 5-5]} - (1.61E\text{-}06 + 1.48E\text{-}07)) / 5.37E\text{-}05] \times 100\ \text{[Table 5-4]}$$

$$\Delta\%CCFP_{BASE\text{-}10} = 0.33\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{BASE\text{-}15}$) due to a 15-year ILRT over the baseline is a follows:

$$\Delta\%CCFP_{BASE\text{-}15} = [((F_{CLASS\ 3a\_15} + F_{CLASS\ 3b\_15}) - (F_{CLASS\ 3a\_BASE} + F_{CLASS\ 3b\_BASE})) / CDF] \times 100$$

$$\Delta\%CCFP_{BASE\text{-}15} = [((1.85E\text{-}06 + 1.70E\text{-}07)\ \text{[Table 5-6]} - (1.61E\text{-}06 + 1.48E\text{-}07)) / 5.37E\text{-}05] \times 100\ \text{[Table 5-4]}$$

$$\Delta\%CCFP_{BASE\text{-}15} = 0.49\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{10\text{-}15}$) due to a 15-year ILRT over the 10-year ILRT is a follows:

$$\Delta\%CCFP_{10\text{-}15} = [((F_{CLASS\ 3a\_15} + F_{CLASS\ 3b\_15}) - (F_{CLASS\ 3a\_10} + F_{CLASS\ 3b\_10})) / CDF] \times 100$$

$$\Delta\%CCFP_{10\text{-}15} = [((1.85E\text{-}06 + 1.70E\text{-}07)\ \text{[Table 5-6]} - (1.77E\text{-}06 + 1.62E\text{-}07)) / 5.37E\text{-}05] \times 100\ \text{[Table 5-5]}$$

$$\Delta\%CCFP_{10\text{-}15} = 0.16\%$$

This change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

5.6 RESULTS SUMMARY

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

1. The baseline (3-per-10-year frequency) risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 3.36% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 3.69% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.31%.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 3.85% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from the once-per-10-year frequency to once-per-15 years is 0.15%.

6. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.46%.

7. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

8. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 2.44E-08/yr. This is also found to be "very small" using the acceptance guidelines in Reg. Guide 1.174.

9. The risk increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 8.0E-09/yr. This is determined to be a very small using the acceptance guidelines of Reg. Guide 1.174.

10. This change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

11. Other salient results are summarized in Table 5-7.

Table 5-7

SUMMARY OF RISK IMPACT ON TYPE A ILRT TEST FREQUENCY

Risk Metric	Risk Impact (Baseline)	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	3.36% of total integrated value 4.83E-1 person-rem/yr	3.69% of total integrated value 5.32E-1 person-rem/yr	3.85% of total integrated value 5.56E-1 person-rem/yr
Total Integrated Risk	14.371 person-rem/year	14.416 person-rem/year	14.438 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.31%	0.46%
Increase in LERF over Baseline	N/A	1.7E-08/yr	2.44E-08/yr
Percent Increase in CCFP over Baseline	N/A	0.33%	0.49%

6.0 CONCLUSIONS

This section provides the principal conclusions of the ILRT test interval extension risk assessments as reported for the following:

- Previous generic risk assessment by the NRC
- Plant specific Ginna risk assessment for the at-power case
- General conclusions regarding the beneficial effects on shutdown risk

6.1 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from the current 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 liner.

6.2 GINNA SPECIFIC RISK RESULTS

The findings for Ginna confirm the general findings of previous studies on a plant specific basis considering the severe accidents evaluated for Ginna, the Ginna containment failure modes, the Ginna Technical Specification allowed leakage, and the local population surrounding Ginna.

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

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

- Reg. Guide 1.174 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 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years is 8.0E-09/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the Ginna risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk by only 0.15%. Therefore, the risk impact change when compared to other severe accident risks is negligible.

- The change in Conditional Containment Failure Probability (CCFP) with respect to a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years was calculated as less than 1%. This is judged to be insignificant and reflects sufficient defense-in-depth.

6.3 RISK TRADE-OFF

The performance of an ILRT occurs during plant shutdown and introduces some small residual risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real shutdown risk impacts associated with the setup and performance of the ILRT during shutdown operation [10]. While these risks have not been quantified for Ginna, it is judged that there is a positive (yet un-quantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT test interval extension, there are, in fact, positive safety benefits that reduce the already small risk associated with the extension of the ILRT test interval.

7.0 REFERENCES

1) *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 1995.

2) *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

3) *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, July 1998.

4) *Performance-Based Containment Leak-Test Program*, NUREG-1493, September 1995.

5) *Evaluation of Severe Accident Risks: Peach Bottom, Unit 2,* Main Report NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 1990.

6) *Technical Findings and Regulatory Analysis for Genetic Study Issue II.e.43 'Containment Integrity Check',* NUREG-1273, April 1988.

7) *Impact of Containment Building Leakage on LWR Accident Risk,* Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

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

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

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

11) *Individual Plant Examination Peach Bottom Atomic Power Station Units 2 and 3,* Volumes 1 and 2 Philadelphia Electric Company, 1992

12) *ALWR Severe Accident Dose Analysis,* DE-ACOG-87RL11313, March 1989.

13) Patrick D. T. O'Connor, *Practical Reliability Engineering,* John Wiley & Sons, 2nd Edition, 1985.

14) Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 16, 2001.

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

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

17) Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* July 1998.

18) Burns, T. J., *Impact of Containment Building Leakage on LWR Accident Risk,* Oak Ridge National Laboratory, NUREG/CR-3539, April 1984.

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

20) Letter from SNC (H. L. Summer, Jr.) to USNRC dated July 26, 2000.

21) United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

22) NUREG-1437, "*Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 14, Regarding R. E. Ginna Nuclear Power Plant,*" dated January 2004.

23) J. Haugh, John Gisclon, W. Parkinson, Ken Canavan, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Intervals", Rev. 4, EPRI, November 2001

24) R. E. Ginna Procedure RSSP-6.0, "Containment Integrated Leakage Rate Test."

25) R. E. Ginna Nuclear Power Plant Probabilistic Safety Assessment Final Report, Revision 4.3

APPENDIX A

Effect of Age-Related Degradation on Risk Informed/Risk Impact Assessment for Extending Containment Type A Test Interval

A.1.0 PURPOSE

The purpose of this calculation is to assess the effect of age-related degradation of the containment on the risk impact for extending the Ginna Integrated Leak Rate Test (ILRT) or Containment Type A test) interval from ten to fifteen years.

A.2.0 INTENDED USE OF ANALYSIS RESULTS

The results of this calculation will be used to indicate the sensitivity of the risk associated with the extension in the ILRT interval to potential age-related degradation of the containment shell to support obtaining NRC approval to extend the Integrated Leak Rate Test (ILRT) interval at Ginna from 10 years to 15 years. This calculation actually evaluates the impact of extending the interval from 3 years to 15 year.

A.3.0 TECHNICAL APPROACH

The present analysis shows the sensitivity of the results of the assessment of the risk impact of extending the Type A test interval for Ginna to age-related liner corrosion.

The prior assessment included the increase in containment leakage for EPRI Containment Failure Class 3 leakage pathways that are not included in the Type B or Type C tests. These classes (3a and 3b) include the potential for leakage due to flaws in the containment shell. The impact of increasing the ILRT interval for these classes included the probability that a flaw would occur and be detected by the Type A test that was based on historical data. Since the historical data includes all known failure events, the resulting risk impact inherently includes that due to age-related degradation.

The present analysis is intended to provide additional assurance that age-related liner corrosion will not change the conclusions of the prior assessment. The methodology used for this analysis is similar to the assessments performed for Calvert Cliffs Nuclear Power Plant (CCNPP – Reference A1), Comanche Peak Steam Electric Station (CPSES – Reference A2), D. C. Cook (CNP – Reference A3) and St. Lucie (SL – Reference A4) in responses to requests for additional information from the NRC staff. The CCNPP, CPSES and CNP extension request submittals have been approved by the NRC.

In general, concrete containments with steel liners such as that at Ginna have a potential for corrosion. Because of this, the analysis is carried out separately for those portions of the containment not in potential contact with foreign material and those portions in potential contact with the foreign material. This is considered more appropriate than the cylinder, dome and the basement portions utilized in prior analyses.

As in Reference A1, this calculation uses the following steps with Ginna values utilized where appropriate:

Step1 – Determine corrosion-related flaw likelihood.

> Historical data will be used to determine the annual rate of corrosion flaws for the containment.

Step 2 – Determine age-adjusted flaw likelihood.

> The historical flaw likelihood will be assumed to double every 5 years. The cumulative likelihood of a flaw is then determined as a function of ILRT interval.

Step 3 – Determine the change in flaw likelihood for an increase in inspection interval.

> The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests is then determined from the results of Step 2.

Step 4 – Determine the likelihood of a breach in containment given a flaw.

For there to be a significant leak from the containment, the flaw must lead to a gross breach of the containment. The likelihood of this occurring is determined as a function of pressure and evaluated at the Ginna ILRT pressure.

Step 5 – Determine the likelihood of failure to detect a flaw by visual inspection.

The likelihood that the visual inspection will fail to detect a flaw will be determined considering the portion of the containment that is uninspectable at Ginna as well as an inspection failure probability.

Step 6 – Determine the likelihood of non-detected containment leakage due to the increase in test interval.

The likelihood that the increase in test interval will lead to a containment leak not detected by visual examination is then determined as the product of the increase in flaw likelihood due to the increased test interval (Step 3), the likelihood of a breach in containment (Step 4) and the visual inspection non-detection likelihood (Step 5). The results of the above for the two regions of the containment are then added to get the total increased likelihood of non-detected containment leakage due to age-related corrosion resulting from the increase in ILRT interval.

The result of Step 6 is then used, along with the results of the prior risk analysis in the body of this analysis to determine the increase in LERF as well as the increase in person-rem/year and conditional containment failure probability due to age-related liner corrosion.

A.4.0 INPUT INFORMATION

1. General methodology and generic results from the Calvert Cliffs assessment of age-related liner degradation (Reference A1).

2. The Ginna ILRT test pressure of 60.0 psig (Reference A5).

3. Ginna containment failure pressure of 145 psia based on liner tearing. (Reference A6).

4. The surface area of the containment potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in the areas of limited access is 41,313.34 ft^2. This is based on the total inside surface area of the containment below the spring-line (i.e., interface between dome section and cylinder section). The inside diameter of the containment is 105 feet and height of the cylinder section is approximately 99 feet.

5. The number of containments, either free-standing steel shell or concrete with steel liners is 104 and the average area of steel potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in the areas of limited access is 61,900 ft^2 (Reference A11).

A.5.0 REFERENCES

A1. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," Constellation Nuclear letter to USNRC, March 27, 2002.

A2. "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, Respond to Request for Additional Information Regarding License Amendment Request (LAR) 01-14 Revision to Technical Specification (TS) 5.5.16 Containment Leakage Rate Testing Program," TXU Energy letter to USNRC, June 12, 2002.

A3. "Donald C. Cook Nuclear Plants Units 1 and 2, Response to Nuclear Regulatory Commission Request for Additional Information Regarding the License Amendment Request for a One-time Extension of Integrated Leakage Rate Test Interval," Indiana Michigan Power Company, November 11. 2002.

A4. "St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Proposed License Amendments, Request for Additional Information Response on Risk-Informed One Time Increases in Integrated Leak Rate Test Surveillance Interval," Florida Power & Light Company letter to USNRC, December 13, 2003.

A5. R. E. Ginna Procedure RSSP-6.0, "Containment Integrated Leakage Rate Test."

A6. R. C. Mecredy (RG&E) letter to the Document Control Desk (NRC), "Generic Letter 88-20, Level 2 Probabilitstic Safety Assessment." August 30, 1997.

A7. S. E. Phillippi, "Calculation of Inspectable And Uninspectable Containment Vessel Surface Areas," SCIENTECH, INC. Analysis File 17547-0001-A2, Rev. 0, March 24, 2003.

A8. "Containment Liner Through Wall Defect due to Corrosion," Licensing Event Report, Ler-NA2-99-02, North Anna Nuclear Power Plant Station Unit 2.

A9. "Brunswick Steam Electric Plant, Units 1 and 2, Dockets 50-325 and 50-324/License Nos. DPR=71 and DPR-62, Response tp Request for Additional Information Regarding Request for License Amendments – Frequency of Performance Based Leakage Rate Testing," CP&L letter to USNRC, February 5, 2002.

A10. "IE Information Notice No. 86-99; Degradation of Steel Containments." USNRC, December 8. 1986.

A11. E. R. Schmidt, "Calculation of Industry Average Containment Surface Area Subject to Age-Related Corrosion Due to Foreign Material," Analysis File 17547-0001-A4, Rev. 0, November 14, 2003.

A.6.0 MAJOR ASSUMPTIONS

1. As indicated in the NRC's Request for Additional Information (References A3 and A4, for example) there have been 4 instances of age-related corrosion leading to holes in steel containment liners or shells. Three of these instances (Cook - Reference A3, North Anna – Reference A8 and Brunswick – Reference A9) were in concrete containments with steel liners and due to foreign material imbedded in the concrete in contact with the steel liner. The fourth instance (Oyster Creek) – Reference A10) was in a freestanding steel containment and occurred in areas where sand fills the gap between the steel shell and the surrounding concrete and was attributed to water accumulating in this sand. This data is therefore considered to represent a corrosion induced failure rate only for the area of the Ginna containment steel shell which can not be visually inspected and has the potential to be in contact with foreign material or where foreign material may be trapped. For the other areas where the containment steel shell is not likely to be in contact with foreign material, the corrosion induced failure rate should be substantially lower and

taken to be that based on no observations of corrosion induced failure of the containment steel shell in these regions.

2. The historical data of age-related corrosion leading to holes in the steel-containment has occurred primarily (3 out of 4 instances) for steel lined concrete containments. For these containments the surface area in contact with the concrete comprises essentially the entire surface area of the containment. For Ginna, the surface area at risk is assumed to be the entire surface area below the spring-line (i.e., interface between the dome section and the cylindrical section) which is in contact with the concrete, can not be visually inspected and has the potential for contact with foreign material. For an internal diameter of 105 feet and cylinder height of approximately 99 feet, the inside surface area is 41, 313 square feet. Since the greater the surface area in contact with the concrete, the greater the chance of foreign material being in contact with the steel containment liner. Therefore, the containment failure rate due to corrosion will be taken to be proportional to the uninspectable surface area in contact with the concrete. The containment failure rate due to corrosion will be taken to be that for the industry times the ratio of the surface area at risk for Ginna to the average area at risk for the industry.

3. The visual inspection data is conservatively limited to 5.5 years reflecting the time from September 1996, when 10 CFR 50.55a started requiring visual inspection, through March 2002, the cutoff date for this analysis. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to September 1996 and after the cutoff date. The two instances of corrosion identified by visual inspections discussed previously in this LAR (Enclosure 1 section 2.8, and section 2.9 Question 2) showed that there have been no identified failures of the liner as the result of corrosion issues. (Step 1)

4. As in Reference A1, the containment flaw likelihood is assumed to double every 5 years. This is included to address the increased likelihood of corrosion due to aging. (Step 2)

5. The likelihood of a significant breach in the containment due to a corrosion induced localized flaw is a function of containment pressure. At low pressures, a breach is very unlikely. Near the nominal failure point, a breach is expected. As in Reference A1, anchor points have a 0.1% chance of cracking near the flaw at 20 psia and 100% chance at the failure pressure (liner tearing failure pressure of 145 psia for Ginna from Reference A6) are assumed with logarithmic interpolation between these two points. (Step 4)

6. In general, the likelihood of a breach in the lower head region of the containment occurring, and this breach leading to a large release to the atmosphere, is less than that for the cylindrical portion of the containment. The assumption discussed in item 5 above is, however, conservatively applied to the lower head region of the containment, as well as to the cylindrical portions.

7. All non-detected containment overpressure leakage events are assumed to be large early releases.

8. The interval between ILRTs at the original frequency of 3 tests in 10 years is taken to be 3 years.

9. Concrete acts to protect steel in contact with it. We feel that there is little likelihood of corrosion occurring in the floor liner plates. We also feel that there is little likelihood of corrosion occurring in the containment dome liner that can not be identified by visual inspection. The interior of the containment dome is not insulated. The basis for this is provided by the evaluation of the condition of the dome liner in contact with concrete

during construction activities associated with Steam Generator replacement. This evaluation showed that the dome liner of the containment building after almost 30 years of service has not degraded and the effect of aging has been insignificant. *Reference Enclosure 1, Section 2.8.*

A.7.0 IDENTIFICATION OF COMPUTER CODES:

None used.

A.8.0 DETAILED ANALYSIS:

A.8.1 Step 1 – Determine a corrosion-related flaw likelihood.

As discussed in Assumptions 1, 2 and 3, the likelihood of through wall defects due to corrosion for the areas of the containment potentially contacted by foreign materials is based on 4 data points in 5.5 years.

[4 failures*(41,313.34ft^2 / 61,900ft^2 / (104 plants*5.5 years/plant) = 4.67E-03 per year]

For the areas of the containment where foreign material is not likely to contact the containment the defect likelihood is taken to be that for no observed failures using a non-informative prior distribution.

Failure Frequency = [# of failures (0) + ½] / (Number of unit years (104*5.5)]
= 8.74E-04 per year.

A similar area-at-risk correction as above for the area in contact with concrete is not appropriate for the area where foreign material is not likely to contact the containment since the majority of the steel liner or shell for all plants has at least one side of the surface subject to this reduced corrosion (and none has been observed).

A.8.2 Step 2 – Determine age-adjusted liner flaw likelihood.

Reference A1 provides the impact of the assumption that the historical flaw likelihood will double every 5 years on the yearly, cumulative and average likelihood that an age-related flaw will occur. For a flaw likelihood of 5.2E-03 per year, the 15 year average flaw likelihood is 6.27E-03 per year for the cylinder/dome region. This result of Reference A1 is generic in nature, as it does not depend on any plant specific inputs except the assumed historical flaw likelihood.

For the present assumption of 4 historical failures in 104 plants, the 15 year average flaw likelihood is 89.8% (4.67E-03/5.2E-03 = 0.898 or 89.8%) of the above value (6.27E-03) or 5.63E-03 per year, and in accordance with Assumption 1, is applicable to the region of the containment potentially in contact with foreign material.

Similarity, for the region of the containment not potentially in contact with foreign material, the 15 year average flaw likelihood is 16.8%(8.74E-04/5.2E-03 = 0.168) of the above value (6.27E-03) or 1.05E-03 per year.

A.8.3 Step 3 – Determine the change in flaw likelihood for an increase in inspection interval.

The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is determined from the result of Step 2 in Reference A1 to be 8.7% for the cylinder/dome region based on assumed historical flaw likelihood and the resulting 6.27E-03/year 15 year average flaw likelihood. This result of Reference A1 is generic in nature, as it does not depend on any plant specific, inputs except the assumed historical flaw likelihood.

For the present assumption of 4 historical failures in 104 plants, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 89.8% (as in Step 2) of that given in Reference A1 (0.898*8.7%) or 7.81% and in accordance with Assumption 1 is applicable to only the region of the containment potentially in contact with foreign material.

Similarly, for the region of the containment not potentially in contact with foreign material, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 16.8% (as in Step 2) of that given in Reference A1 or 1.46%.

A.8.4 Step 4 – Determine the likelihood of a breach in containment given a liner flaw.
The likelihood of a breach in containment occurring is determined as a function of pressure as follows:

For a logarithmic interpolation on likelihood of breach

LOG (likelihood of breach) = m (pressure) + a

Where m = slope
a = intercept

The values of m and a are determined from solution of the two equations for the values of 0.1% at 20 psia and 100% of containment failure pressure of 145 psia (Reference A6).

Log 0.1 = m*20 + a

Log 100 = m*145 +a

or

m = (Log 100 – Log 0.1)/(145-20) = 0.024

and

a = Log 0.1 – 0.024*20 = -1.48

The upper end of the range of Ginna ILRT pressure of 60.0 psig (Reference A5) gives the highest likelihood of breach.

At 74.7 psia (60.0 + 14.7), the above equation gives

Log (likelihood of breach) = 0.024*74.7 – 1.48 = 0.3128

Likelihood of breach = $10^{0.3128}$ = 2.05%

In accordance with Reference A1, the above value is for the cylinder/dome portions of the containment. For this analysis, this value is also assumed to be applicable to the region of the containment potentially in contact with foreign material.

A.8.5 Step 5 – Determine the likelihood of failure to detect a flaw by visual inspection

A 10% failure rate for that portion of the containment that is visually inspectable is assumed.

A.8.6 Step 6 – Determine the likelihood of non-detected containment leakage due to the increase in test interval.

The likelihood of non-detected containment leakage in each region due to age-related corrosion of the liner considering the increase in ILRT interval is then given by:

The increased likelihood of an undetected flaw because of the increased ILRT interval (Step3)	*	The likelihood of a containment breach given a liner flaw (Step 4)	*	The likelihood that visual inspection will not detect the flaw (Step 5)

= 1.46% * 0.0205*0.10 = 0.003% for the regions not potentially contacted by foreign material.

= 7.81% * 0.0205*1.0 = 0.16% for the regions potentially contacted by foreign material.

The total is then the sum of the values for the two regions or

Total Likelihood of Non-Detected Containment Leakage = 0.003% + 0.16%
= 0.163% for the ILRT interval increase from 3 years to 15 years.