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10 CFR 50.90

TMI-14-123

October 30, 2014

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

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: License Amendment Request - Modify Reactor Coolant System Pressure Isolation Check Valve Technical Specification Maximum Allowable Leakage Limits

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (Exelon) requests the following amendment to the Technical Specifications, Appendix A, of Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI).

The proposed amendment would modify the Technical Specification (TS) Table 3.1.6.1, "PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS," maximum allowable leakage limits.

Attachment 1 provides the Evaluation of Proposed Changes. Attachment 2 provides the Proposed Technical Specification Marked-Up Pages. Attachment 3 provides a Simplified Schematic Diagram of Emergency Core Cooling Systems. Attachment 4 provides Reactor Coolant System Pressure Isolation Valve (RCS PIVs) Through-Leakage Testing History.

The proposed change has been reviewed by the TMI Plant Operations Review Committee and approved in accordance with Nuclear Safety Review Board procedures.

Exelon requests approval of the proposed amendment by October 16, 2015, in order to support the TMI Fall 2015 Refueling Outage. Once approved, the amendment shall be implemented within 30 days.

There are no regulatory commitments contained in this request.

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

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

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of October 2014.

Respectfully,

James B

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

Attachments:
1) Evaluation of Proposed Technical Specification Changes
2) Proposed Technical Specification Marked-Up Pages
3) Simplified Schematic Diagram of Emergency Core Cooling Systems
4) Reactor Coolant System Pressure Isolation Valve Through-Leakage Testing History

cc: USNRC Regional Administrator, Region I
USNRC Project Manager, TMI-1
USNRC Senior Resident Inspector, TMI-1
Director, Bureau of Radiation Protection - PA Department of Environmental Protection

ATTACHMENT 1

EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

SUBJECT: Modify Reactor Coolant System Pressure Isolation Check Valve Technical Specification Maximum Allowable Leakage Limits

CONTENTS

Modify Reactor Coolant System Pressure Isolation Check Valve Technical Specification Maximum Allowable Leakage Limits

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI).

The proposed changes will revise the Technical Specifications (TS) Table 3.1.6.1, "PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS," by deleting footnotes (a)1 through (a)4 involving additional restrictions to the Maximum Allowable Leakage limits of 5 gallons per minute (gpm). In addition, the proposed changes will correct two typographical errors and one clerical error.

Currently, the Maximum Allowable Leakage limit for these Reactor Coolant System Pressure Isolation Valves (RCS PIVs) is ≤ 5.0 gpm. The Maximum Allowable Leakage limit is further restricted with Footnote (a) that describes additional incremental surveillance testing acceptance criteria prior to reaching the 5 gpm limit, which is based in part on the remaining margin between the 5 gpm limit and the last measured leakage. This proposal requests deletion of the Footnotes (a)1 through (a)4, making it consistent with Improved Standard Technical Specification (ITS) surveillance requirements (NUREG 1430 SR 3.4.14.1). (Reference 1)

Exelon Generation Company, LLC (Exelon) requests approval of the proposed changes within approximately one year (by 10/16/15) of this submittal to support our T1R21 Refuel Outage in Fall 2015. Once approved, the amendment shall be implemented to support outage activities and within 30 days.

2.0 DETAILED DESCRIPTION

Technical Specification Changes

TS Table 3.1.6.1, "PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS," Maximum Allowable Leakage Footnotes (a)1 through (a)4 below will be deleted:

Footnote:
(a)
1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.
2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are considered unacceptable.

The following typographical errors are being corrected:

1. Table 3.1.6.1, Train A, CF-VSA has a typographical error and is being corrected to CF-V5A.

2. In TS 3.1.6.10.a, the acronym LIPS is being corrected to LPIS.

The following clerical error is being corrected:

1. In Table 3.1.6.1 delete "(≤5.0 GPM for all valves)" phrase that is on the same typing line that begins with "Low Pressure Injection."

2. In table 3.1.6.1 revise valve list to read:

CF-V5A	≤ 5.0 GPM
DH-V22A	≤ 5.0 GPM
CF-V5B	≤ 5.0 GPM
DH-V22B	≤ 5.0 GPM

<u>Core Flooding System Description</u>

The Core Flooding System provides core protection continuity for intermediate and large RCS pipe failures as described in TMI UFSAR section 6.1.2.1. This system automatically floods the core when the RCS pressure drops below 600 psig. The Core Flooding System is self-contained, self-actuating, and passive in nature. The combined coolant volume in the two tanks (CF-T-1A, 1B) is sufficient to re-cover the core hot spot, assuming no liquid remains in the reactor vessel following the LOCA. The discharge pipe from each core flooding tank is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the core flooding tanks contains an electric motor operated stop valve adjacent to the tank and two inline nominal 14-inch check valves in series (CF-V4A, 5A and CF-V4B, 5B). The CF-V5A and 5B check valves are considered RCS PIVs. The stop valves at the core flooding tank outlet are fully open, with electric power removed, during reactor power operation.

During power operation when the RCS pressure is higher than the Core Flooding System pressure, the two series check valves between the flooding nozzles and the core flooding tanks prevent high pressure reactor coolant from entering the core flooding tanks.

The driving force to inject the stored borated water into the reactor vessel is supplied by pressurized nitrogen which occupies approximately one third of the core flooding tank volume. Connections are provided for adding both borated water and nitrogen during power operation so that the proper level and pressure may be maintained.

Each core flooding tank is protected from overpressurization by a relief valve installed directly on the tank. The size of these relief valves is based upon a maximum water makeup rate (100 gpm) to the core flooding tank. The relief valve at the top of each core flooding tank has a set pressure of 700 psig air at 130.0 °F. The relief valve capacity is 1,296 scfm air at 60 °F. Redundant level and pressure indicators for each tank are provided on the main control console. High and low pressure and level are alarmed in the main control room.

Decay Heat Removal System Description

The Decay Heat Removal System is designed to maintain core cooling for larger break sizes and is described in TMI UFSAR Section 6.1.2.1, ECCS Operation. This system provides low pressure injection independent of and in addition to the high pressure injection provided by the Makeup and Purification System.

Low pressure injection will be accomplished through two separate flow paths, each including one pump and one heat exchanger and terminating directly in the reactor vessel through core flooding nozzles located on opposite sides of the vessel.

The initial injection of water by the Decay Heat Removal System involves pumping water from the borated water storage tank into the reactor vessel. With all engineered safeguards pumps operating, and assuming the maximum break size, this mode of operation lasts for a minimum of approximately 25 minutes. When the borated water storage tank reaches a low level, the operator will take action to open the suction valves from the Reactor Building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the Reactor Building sump.

Items to be inspected for leaks to atmosphere will be pump seals, valve packing, flange gaskets, heat exchangers, and safety valves. The check valves of the low pressure injection system (DH-V22A, B) are leak tested per Technical Specification 4.2.7. The DH-V22A and B check valves are considered RCS PIVs.

Decay Heat Removal System relief valves are provided to protect the low pressure piping and components from overpressure. The relief valves are set to protect system components consistent with their design pressures. The relief valve in each flow path has a set pressure of 520 psig liquid at 60.0 °F and a capacity of 32 gpm liquid at 70.0 °F.

RCS PIVs (CF-V5A, B and DH-V22 A, B) are those check valves in the primary RCS that isolate the boundary between the high pressure primary coolant system and connected low pressure piping systems. These valves are included in a Simplified Schematic Diagram of Emergency Core Cooling Systems (see Attachment 3).

3.0 TECHNICAL EVALUATION

PIV leakage testing was originally established by the NRC in response to concerns regarding the intersystem loss-of-coolant accident (ISLOCA), which was identified in the Reactor Safety Study of 1975, WASH-1400 (Reference 2). An ISLOCA event at TMI would involve the failure of two in-series RCS PIVs, which would subject a low pressure system outside of containment to full primary coolant system pressure. The low pressure system would consequently rupture, resulting in a LOCA that would bypass containment, thereby jeopardizing the ability for long-term reactor core cooling. NUREG-0103 Rev 4 Standard Technical Specifications for Babcock and Wilcox PWRs (Fall 1980) initially limited RCS PIV leakage to 1 gpm.

TS LCO 3.1.6.10 and TS Table 3.1.6.1 provide requirements for the Maximum Allowable Leakage for RCS PIVs, including the limiting condition for operation, action requirements and surveillance requirements. TS Table 3.1.6.1 limits Maximum Allowable Leakage for each RCS PIV to ≤ 5 gpm. For TMI, the additional leakage surveillance testing acceptance criteria embodied in Table 3.1.6.1 Footnote (a) evolved from NRC TMI Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, dated April 20, 1981 (Reference 3). These additional requirements provided further attention to the rate of valve leakage degradation in the 1 to 5 gpm range.

Although this specification provides a limit on allowable RCS PIV leakage, and additional rate of valve leakage degradation limit, its main purpose is to prevent overpressure failure of the low-pressure portions of connecting systems. The Maximum Allowable Leakage limit is an indication that the PIVs between the RCS and the connecting lower pressure system are degraded or degrading. Excessive RCS PIV leakage could lead to overpressure of the low-pressure piping or components, potentially resulting in an ISLOCA outside of containment.

The NRC, through its approval of Babcock and Wilcox Improved Standard Technical Specifications (ITS), NUREG-1430, has revised the leakage rates enforced in the TMI April 20, 1981 Order and endorsed an RCS PIV leakage rate limit without additional incremental surveillance testing acceptance criteria (i.e., Footnotes (a)1 through (a)4). The revised ITS RCS PIV leakage limit is 0.5 gpm per nominal inch of valve diameter with a maximum limit of 5 gpm. This change tightened the leakage requirement for smaller valves and relaxed it for larger ones. CF-V5A, B are nominal 14-inch diameter valves and the DH-V22A, B are nominal 10-inch diameter valves, and therefore have a Maximum Allowable Leakage rate of 5 gpm using ITS criteria. The propose deletion of the incremental surveillance testing acceptance criteria is consistent with ITS, which contains no required additional incremental surveillance testing acceptance criteria as valve leakage progresses to the maximum allowable leakage of 5 gpm.

NUREG-0677, May 1980 (Reference 4) evaluated various RCS PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the RCS PIVs can substantially reduce the probability of an ISLOCA. There was no consideration of additional incremental surveillance testing acceptance criteria prior to reaching a predetermined maximum allowable leakage in the probability calculations for a check valve leak failure and rupture failure calculations for the specific configuration of two series check valves. Therefore, the probability of an ISLOCA is unaffected.

The proposed change does not affect the current Inservice Testing (IST) Program. The RCS PIVs are Class 1 Category A/C valves and are tested in accordance with ASME OM Code, Edition 2004 through the 2006 addenda, which is the latest edition incorporated by reference in Paragraph (b) of 10 CFR 50.55a for TMI. ASME OM Code ISTC-3630, "Leakage Rate for Other Than Containment Isolation Valves," requires testing every 2 years. TMI TS require leakage testing every refuel outage or 9 months for forced outages. The requirements for water leakage rates are 0.5D gal/min or 5 gal/min, whichever is less, at function pressure differential, where D is nominal valve size in inches. TMI TS require the Maximum Allowable Leakage to be ≤ 5.0 gpm.

A potential concern related to deleting the PIV allowable leakage rates incremental surveillance testing requirements is that the low pressure systems isolated by the PIVs may not have sufficient pressure relief capacity to cope with the incremental increased leakage (up to 5 gpm limit). In-leakage exceeding the pressure relief capacity of a low pressure system could lead to its overpressurization and rupture. The maximum allowable leakage of 5 gpm remains the same and is well within the pressure relief valve capacity of the Core Flooding and Decay Heat Removal Systems. These relief valves are tested in accordance with ASME O&M Code.

In addition, the proposed change will have the additional benefit of reducing the frequency of maintenance that places the plant in a higher risk condition during outage conditions for the repair of the check valves. The higher risk condition is caused by the reduction in RCS water inventory during the repair of the check valves. There is also a corresponding dose benefit. Approximately 125 mrem dose was received during the 2003 repair of CF-V5B.

The history of the leakage results from the outage testing and maintenance of the RCS PIVs is provided in Attachment 4. As allowed by the TS, these tests are performed at reduced RCS pressure, with at least 150 psi pressure differential across each of the valves. The leakage results are adjusted up to the maximum accident pressure differential across each of the valves as described in the ASME OM Code, which applies a multiplier that is the square root of the pressure differential ratio. This adjustment is conservative in that it does not recognize that the valves may seat tighter at higher differential pressures. A brief summary of the trends is provided below:

- CF-V5A has maintained a leak rate generally below approximately 0.5 gpm since August 1981.
- CF-V5B has been found to have leakage. It most recently underwent seat repair in 2003, but its leakage has increased over the last few outage tests to approximately 2.6 gpm. The current trend has been identified and being tracked in the Corrective Action Program.
- DH-V22A/B have had no measureable leakage since 1999, except in 2010, when each valve was found with less than 1 gpm. The step increase in 2007 on the chart results from adding a conservatism to the test result to account for uncertainty due to the volume of the test tubing when there is no measureable leakage. Thus, the 0.68 gpm value is a bounding value representing only the maximum possible leakage rate for each of these valves.

Since the valves are paired in series, CF-V5A and DH-V22A protect the “A” train of LPI, and CF-V5B and DH-V22B protect the “B” train of LPI. The actual leakage from the RCS will be based on the tighter of the two valves in series. Thus, based on IST leakage testing, it is currently < 0.22 gpm for one train and < 0.68 gpm for the other train.

RCS leakage is trended throughout the operating cycle to identify adverse RCS leakage trends. This trending is to ensure early detection and prevention of leakage from Reactor Primary Systems. The RCS Leakage Monitoring and Action Plan (Reference 5) is a fully developed statistical program sensitive to changes in total RCS mass that is measured daily. The daily measurement is commonly referred to as the RCS “mass balance.” This is done in compliance with the Pressurized Water Reactor Owners Group (PWROG) Operations Subcommittee “Needed” Recommendations to address NRC concerns related to RCS leakage trending. The RCS total leakage mass balance calculated quarterly baseline, or mean, values for the current operating cycle have been less than 0.02 gpm. This includes RCS PIV leakage. Any leakage past the CF-V-5 and DH-V-22 valve pairs would be evident in the total RCS leak rate. Since the total RCS leakage is so low, there is little or no leakage past the paired PIVs protecting the LPI system piping. The 0.02 gpm value is several orders of magnitude less than the subject PIV limit. Therefore, leakage from the RCS through the subject PIVs during current cycle operation would be indicated in the RCS total leakage mass balance trend prior to approaching the 5 gpm limit.

The deletion of the additional incremental surveillance testing acceptance criteria represents a reduction in testing margin prior to exceeding the 5 gpm Maximum Allowable Leakage. The 5 gpm leak represents the point in valve leakage rate deterioration where valve inspection and maintenance is required. The reduction in testing margin is more than compensated by existing programs in place for addressing RCS leakage. The IST program requires timely evaluations of significant adverse trends in valve leakage. The RCS Leakage Monitoring Program provides daily monitoring of RCS leakage with Action Levels starting at a detected increase of either 0.1 gpm total leakage, or nine consecutive daily values greater than the baseline, or mean, value (currently 0.0126 gpm). The Corrective Action Program requires equipment inspection or testing that reveals more than the expected amount of age-related degradation to be addressed via the initiation of an issue report.

In regards to the typographical and clerical errors, the valve number is CF-V5A, not CF-VSA and the correct acronym for Low Pressure Injection System is LPIS, not LIPS. When the NRC TMI Order (Reference 3) was implemented in 1981, the leakage limit of ≤ 5 gpm was intended to be applied on a per valve basis. There was no intent to limit total RCS PIV leakage for all valves, or a particular LPIS train, to the ≤ 5.0 gpm limit. There is no discussion in the NRC Order's Safety Evaluation Report or the referenced Franklin Research Center Technical Evaluation Report that supports applying the ≤ 5.0 gpm limit to total RCS PIV leakage or a particular LPIS train. A review of the same 1981 NRC Orders for similar designed Babcock and Wilcox plants was conducted and the review concluded that all leakage limits were applied on a per valve basis (Reference 6). The inclusion of this leakage limit (≤ 5.0 GPM for all valves) on the same typing line as the system title, "Low Pressure Injection," and the same typing lines that begin with "Train A" and "Train B" was a clerical/administrative error when implementing the final wording for the revised TS page 3-15b.

In summary, the proposed change satisfies the basis for RCS PIV allowable leakage requirements; would not challenge the pressure relief capacities of connected low pressure systems; does not increase the probability of an ISLOCA; and is consistent with ITS leakage testing requirements for Babcock and Wilcox plants (NUREG-1430) and the leakage requirements specified in the ASME OM Code.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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

Codes:

10 CFR 50.2

...*Reactor coolant pressure boundary* means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

- (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
- (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
- (iii) The reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping...

10 CFR 50.55a(c)

(c) *Reactor coolant pressure boundary*. (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III [4, 5] of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

(2) Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, *Provided*:

- (i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or
- (ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

(3) The Code edition, addenda, and optional ASME Code cases to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, subject to the following conditions:

- (i) The edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section;
- (ii) The ASME Code provisions applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition;
- (iii) The ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 edition; and
- (iv) The optional Code cases applied to a component must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(4) For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit.

10 CFR 50 Appendix A, General Design Criterion (GDC) 54

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

10 CFR 50 Appendix A, General Design Criterion (GDC) 55

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:
(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

ASME OM Code, Edition 2004 Edition through 2006 Addenda

ISTC-3630 Leakage Rate for Other Than Containment Isolation Valves

Relevant Guidance:

Regulatory Guide 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage, Rev 1

...Intersystem leakage may not be detectable through the above-mentioned systems; therefore, plants should employ other alarm and leakage monitoring methods. Methods should include monitoring the activity of water flowing through the containment boundary into the connected systems, as well as measuring airborne radioactivity where such systems vent outside the containment boundary. Another method of obtaining indications of uncontrolled or undesirable intersystem flow is to perform a water inventory balance, which is designed to provide appropriate information (such as abnormal water levels in tanks and abnormal flow rates)....

NUREG 1430, "Standard Technical Specifications, Babcock and Wilcox Plants," Revision 4

WASH -1400, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix V, October 1975.

10CFR50.2, 10CFR50.55a(c), and GDC 55 of 10CFR50, Appendix A define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB) which separate the high pressure RCS from an attached low pressure system. During their service lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV leakage LCO TS 3.1.6 allows leakage through these valves in amounts that do not compromise safety.

4.2 PRECEDENT

License Amendment No.154 for Virgil C, Summer Nuclear Station, Unit 1 - Increased Allowable Operational Leakage rate for Reactor Coolant System Pressure Isolation Valves (TAC No. MB2238, dated February 14, 2002 (ML020560626)

> This amendment revised V. C. Summer TS 3.4.6.2f by increasing the allowable operational leakage rate for 23 of the 35 RCS PIVs listed in TS Table 3.4-1. This change implemented a size-dependent allowable leakage rate of 0.5 gallon per minute per nominal inch of valve diameter, up to a maximum of 5 gallons per minute per valve per NUREG-1431. The associated surveillance requirements did not contain an additional leakage testing criteria based on the remaining testing margin to maximum allowable leakage of 5 gpm.

License Amendment Nos. 259 and 250 for Sequoyah Nuclear Plant Units 1 and 2 - Issuance of Amendments Regarding Enhancement of Reactor Coolant Leakage Detection and Operational Leakage Consistent with Standard Technical Specifications (TAC Nos. MA6760 and MA6761) (TS 98-10), dated August 4, 2000 (ML003738637)

> This amendment, in part, for RCS PIVs implemented a size-dependent allowable leakage rate of 0.5 gallon per minute per nominal inch of valve diameter, up to a maximum of 5 gallons per minute per valve per NUREG-1431. The associated surveillance requirements did not contain an additional leakage testing criteria based on the remaining testing margin to maximum allowable leakage of 5 gpm.

4.3 NO SIGNIFICANT HAZARDS CONSIDERATION

Exelon has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed changes will not alter the way any structure, system, or component (SSC) functions, and will not alter the manner in which the plant is operated. In addition, the proposed amendment will not impact the ability of any SSC to mitigate an accident as currently evaluated in the UFSAR.

   This proposed change deletes certain Reactor Coolant System Pressure Isolation Valve (RCS PIV) allowable leakage surveillance testing criteria in consideration of the safety significance and design capabilities of the plant and current industry testing and maintenance practices. The proposed change is consistent with Improved Standard Technical Specification (ITS) NUREG 1430, Standard Technical Specifications, Babcock and Wilcox Plants," Revision 4, and current RCS PIV leak testing practices. The maximum allowable leakage rate of 5 gpm remains unchanged; only the leakage testing incremental testing acceptance criteria below the 5 gpm limit is being deleted. Since the testing frequency and maximum allowable leakage remains unchanged, the probability or consequence of an interfacing system loss-of-coolant accident (ISLOCA) is unaffected. There are no changes to the ASME OM Code leakage testing requirements and methods for this class of valves. Additionally, two typographical errors and one clerical error are being corrected which are administrative in nature.

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

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

   Response: No.

   The proposed revision is not a result of changes to plant equipment, system design, or operating practices. The modified LCO requirement will allow some relaxation of the leak testing method acceptance criteria for the RCS PIVs, consistent with NUREG-1430. Since the functions of the associated systems will continue to perform without change, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. Further, the proposed changes do not introduce any new failure modes. Additionally, two typographical errors and one clerical error are being corrected which are administrative in nature.

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

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

   Response: No.

   The proposed revision to the RCS PIV leakage testing acceptance criteria will not result in changes to system design or setpoints that are intended to ensure timely identification of plant conditions that could be precursors to accidents or potential degradation of accident mitigation systems. Since testing frequency and maximum allowable leakage for the RCS PIVs remain unchanged, the margin associated with the identification of RCS PIV degradation is not significantly reduced. The confidence in the ability of the fission product barriers (fuel cladding, RCS boundary, containment) to limit the level of radiation dose to the public remains the same. Additionally, two typographical errors and one clerical error are being corrected which are administrative in nature.

Since the setpoints and design features that support the margin of safety are unchanged, and actions for inoperable systems continue to provide appropriate time limits and compensatory measures, the proposed changes will not significantly reduce the margin of safety.

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

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

4.4 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NUREG 1430, "Standard Technical Specifications, Babcock and Wilcox Plants," Revision 4

2. Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix V, October 1975

3. NRC TMI Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, dated April 20, 1981(ML003764606, ML003764617, ML003764625, ML003764633)

4. NUREG-0677, The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes, May 1980

5. ER-AP-331-1003, "RCS Leakage Monitoring and Action Plan," Revision 7

6. NRC Orders for Modification of License Concerning Primary Coolant System Pressure Isolation Valves:

   - Arkansas Nuclear One -1, Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981 (ML021220493)

   - Crystal River-3, Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981 (ML02063007)

   - Davis Besse, Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves April 20, 1981 (ML021160283)

   - Oconee-1, 2, 3, Order for Modification of Licenses Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981 (ML011990044)

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION MARKED-UP PAGES

(Unit 1)

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3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7, except that such losses when added to leakage shall not exceed 30 gpm. If leakage plus losses exceeds 30 gpm, the reactor shall be placed in HOT SHUTDOWN within 24 hours of detection.

3.1.6.10 Operating conditions of POWER OPERATION, STARTUP and HOT SHUTDOWN apply to the operational status of the high pressure isolation valves between the primary coolant system and the low pressure injection system.

INSERT

LPIS

a. During all operating conditions in this specification, all pressure isolation valves listed in Table 3.1.6.1 that are located between the primary coolant system and the LIPS shall function as pressure isolation devices except as specified in 3.1.6.10.b. Valve leakage shall not exceed the amount indicated in Table 3.1.6.1.(a)

b. In the event that integrity of any high pressure isolation check valves specified in Table 3.1.6.1 cannot be demonstrated, reactor operation may continue provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition. (b)

c. If Specification 3.1.6.10.a or 3.1.6.10.b cannot be met, an orderly shutdown shall be accomplished by achieving HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within an additional 30 hours.

Bases

Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactive contamination and required cleanup or, in the case of reactor coolant, it could develop into a still more serious problem and, therefore, the first indications of such leakage will be followed up as soon as practical. The unit's makeup system has the capability to makeup considerably more than 30 gpm of reactor coolant leakage plus losses.

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks.

(a) For the purpose of this specification, integrity is considered to have been demonstrated by meeting Specification 4.2.7.

(b) Motor operated valves shall be placed in the closed position and power supplies deenergized.

TABLE 3.1.6.1

PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS

DELETE

INSERT

System	Valve No.	Maximum(a) Allowable Leakage
Low Pressure Injection		(≤5.0 GPM for all valves)
Train A	CF-V~~3~~A 5	~~(~~≤5.0 GPM ~~for all valves)~~ ≤5.0 GPM
	DH-V22A	
Train B	CF-V5B	~~(~~≤5.0 GPM ~~for all valves)~~ ≤5.0 GPM
	DH-V22B	

INSERT

MODIFY

Footnote:

(a)

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

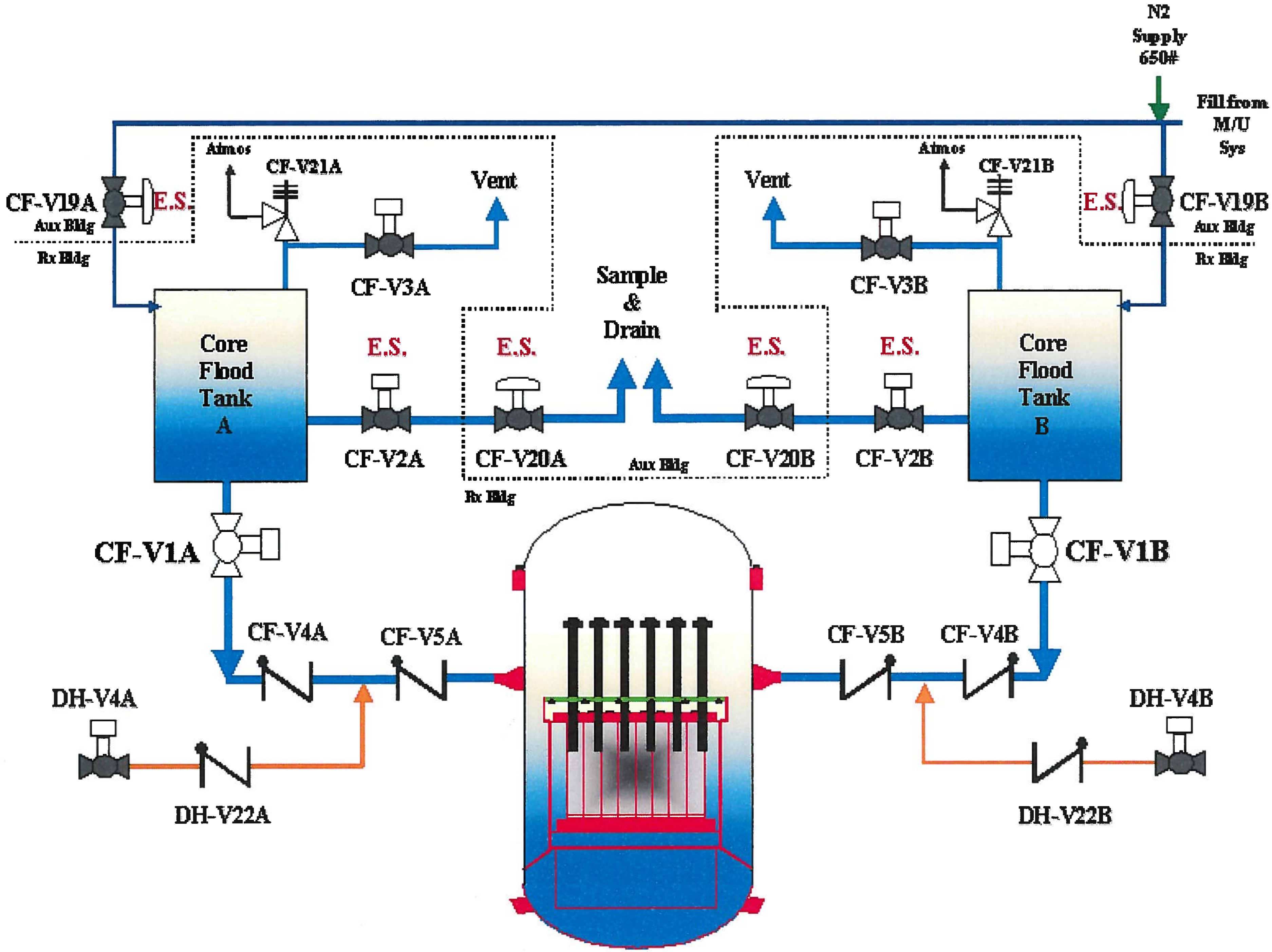
4. Leakage rates greater than 5.0 gpm are considered unacceptable.

DELETE

~~Order Dated 4/20/81~~
~~Amendment No. 141~~

ATTACHMENT 3

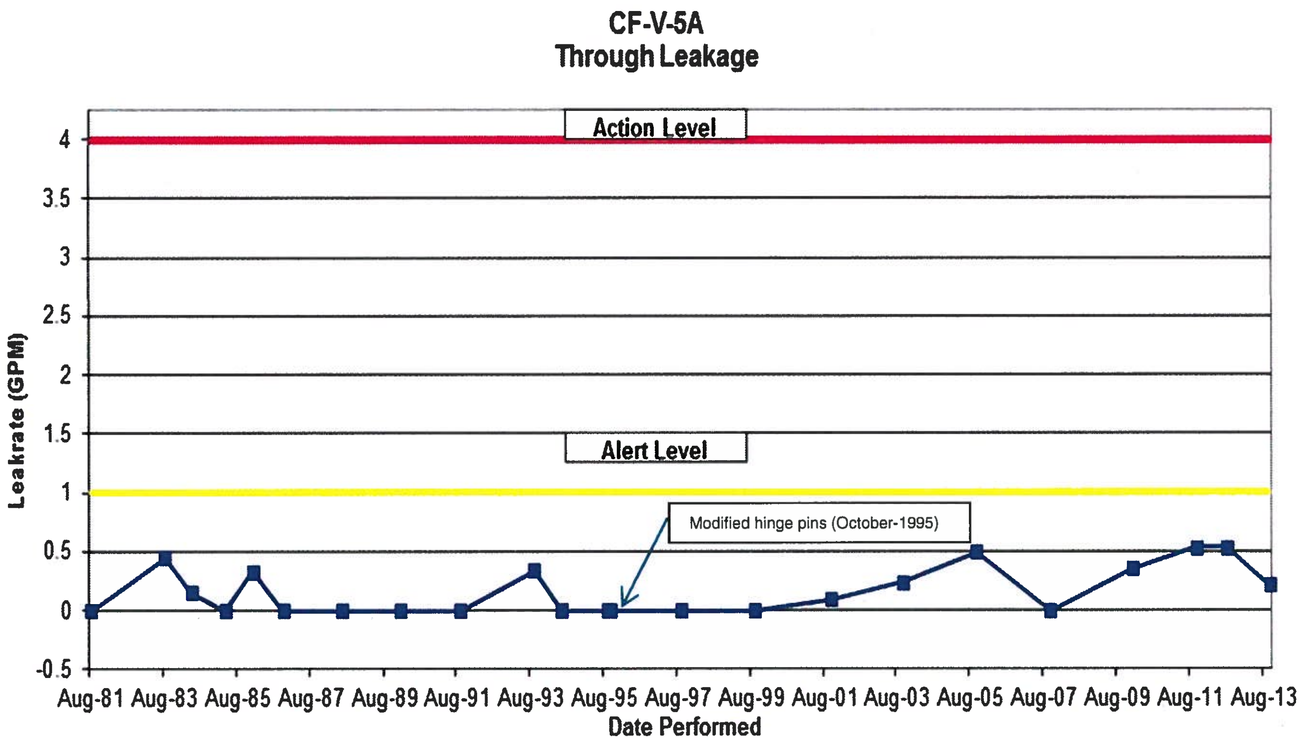
SIMPLIFIED SCHEMATIC DIAGRAM OF EMERGENCY CORE COOLING SYSTEMS



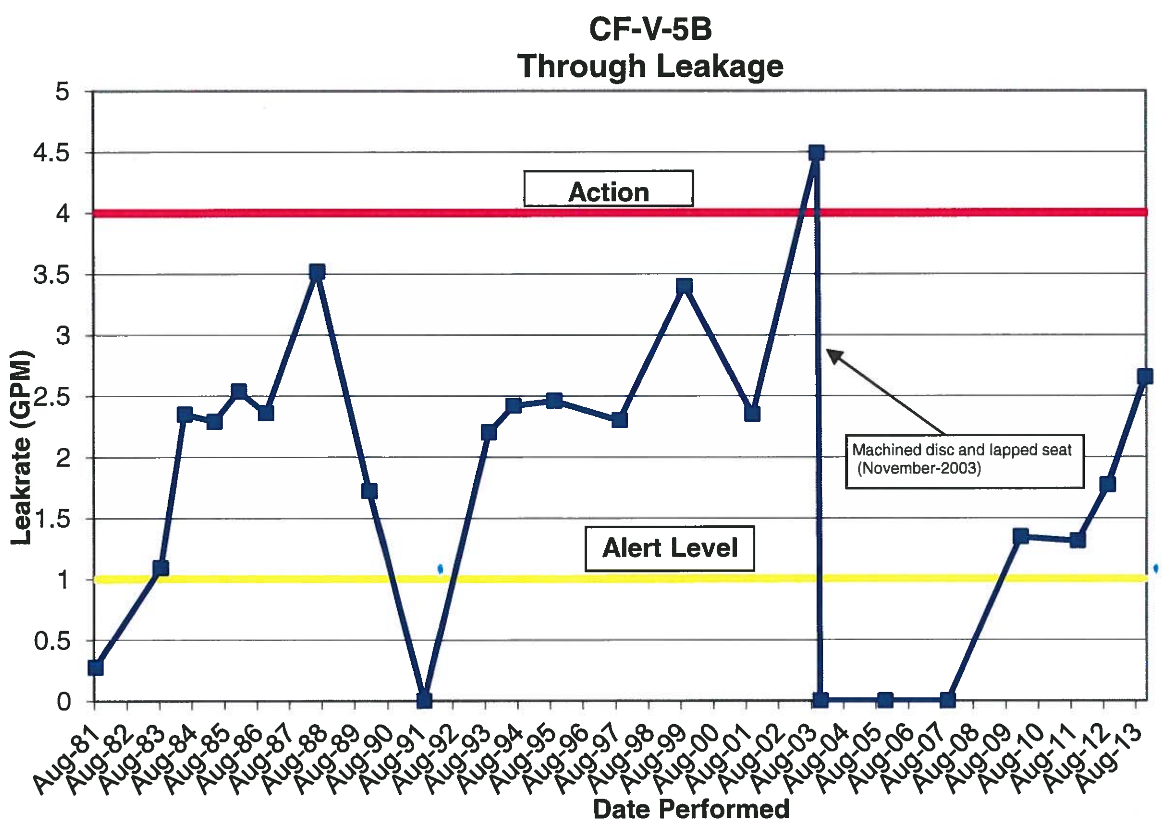
N2
Supply
650#
Fill from
M/U
Sys
CF-V19A
Aux Bldg
Rx Bldg
E.S.
Atmos
CF-V21A
Vent
CF-V3A
Sample
&
Drain
Vent
Atmos
CF-V21B
E.S.
CF-V19B
Aux Bldg
Rx Bldg
CF-V3B
Core
Flood
Tank
A
E.S.
E.S.
E.S.
E.S.
Core
Flood
Tank
B
CF-V2A
CF-V20A
Aux Bldg
CF-V20B
CF-V2B
Rx Bldg
CF-V1A
CF-V1B
CF-V4A
CF-V5A
CF-V5B
CF-V4B
DH-V4A
DH-V4B
DH-V22A
DH-V22B

ATTACHMENT 4

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE THROUGH-LEAKAGE TESTING HISTORY



CF-V-5A
Through Leakage
Action Level
Alert Level
Modified hinge pins (October-1995)
Leakrate (GPM)
4
3.5
3
2.5
2
1.5
1
0.5
0
-0.5
Aug-81 Aug-83 Aug-85 Aug-87 Aug-89 Aug-91 Aug-93 Aug-95 Aug-97 Aug-99 Aug-01 Aug-03 Aug-05 Aug-07 Aug-09 Aug-11 Aug-13
Date Performed



CF-V-5B
Through Leakage
Action
Alert Level
Machined disc and lapped seat
(November-2003)
Leakrate (GPM)
5
4.5
4
3.5
3
2.5
2
1.5
1
0.5
0
Aug-81 Aug-82 Aug-83 Aug-84 Aug-85 Aug-86 Aug-87 Aug-88 Aug-89 Aug-90 Aug-91 Aug-92 Aug-93 Aug-94 Aug-95 Aug-96 Aug-97 Aug-98 Aug-99 Aug-00 Aug-01 Aug-02 Aug-03 Aug-04 Aug-05 Aug-06 Aug-07 Aug-08 Aug-09 Aug-10 Aug-11 Aug-12 Aug-13
Date Performed

DH PIV Leakage
(T.S. limit is the lesser of 5 gpm or halfway from last test to 5 gpm)