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March 24, 1998

LTR: BYRON 98-0099
FILE: 2.01.0301

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Application for Amendment to Appendix A, Technical Specifications, for Facility Operating Licenses:

Byron Nuclear Power Station, Units 1 and 2
Facility Operating Licenses NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Containment Leak Rate Testing

Reference: Letter from J.B. Hosmer (Commonwealth Edison) to NRC Document Control Desk dated November 7, 1997

Pursuant to Title 10, Code of Federal Regulations, Part 50, Section 90 (10 CFR 50.90), Commonwealth Edison Company (ComEd) proposes to revise Technical Specifications (TS) Surveillance Sections 4.6.1.1.c , 4.6.1.2.a, 4.6.1.2.b and the bases to allow ComEd to defer the 10 CFR 50, Appendix J, Type A testing of Byron Unit 2 containment until the next refuel outage in 1999.

This submittal is in addition to the request that was provided in the referenced letter to amend Appendix A, Technical Specifications (TS), for Byron Nuclear Power Station, Units 1 and 2, and Braidwood Nuclear Power Station, Units 1 and 2. The referenced request proposes to adopt a 10 CFR 50, Appendix J, Type A test interval of 10 years for Byron Unit 2 and Braidwood Unit 2 based on a single successful Type A test rather than two successful Type A tests.

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Currently, the Type A test surveillance period for Byron 2 expires in December of 1998 [1]. Therefore, this proposed schedule exemption to Fall 1999 requests a surveillance extension of approximately 10 months for Byron Unit 2. The last Byron Unit 2 Type A test was successfully performed in September 1993; therefore, this extension would result in a Type A test interval of 73 months.

In support of the revision of the Type A test interval, this submittal provides information on: the design and functionality of the steam generator secondary side manways, the operating experience in identifying secondary side leakage, and physical behavior of the steam generator secondary in a post-loss-of-coolant accident (LOCA) situation.

This amendment request applies to Byron Unit 2 only. It is being docketed for Byron Units 1 and 2 because the Technical Specifications for Units 1 and 2 are shared.

The proposed changes in this license amendment request have been reviewed and approved by both On-site and Off-site Review in accordance with ComEd procedures. A detailed description and a safety analysis of the proposed changes are presented in Attachment A. The proposed changes to the TS are presented in Attachment B. ComEd has reviewed this proposed license amendment in accordance with 10 CFR 50.92(c) and has determined that no significant hazards consideration exists. This evaluation is documented in Attachment C. An Environmental Assessment has been completed and is contained in Attachment D.

ComEd is notifying the State of Illinois of its application for this license amendment by copy of this letter and its attachments to the designated State Official.

ComEd respectfully requests the NRC review and approve this license amendment request by April 27, 1998, to support the B2R07 outage scheduling activities at Byron.

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[1] The surveillance expiration date of December 1998 is the result of extending the test interval by 15 months to accommodate the changes in Byron Unit 2's refueling outage schedule as allowed by NEI 94-01, (Nuclear Energy Institute "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J"), Section 9.1.

To the best of my knowledge, the statements contained in this document are true and correct.

Please address any comments or questions regarding this matter to our Nuclear Regulatory Services Department.

Sincerely,

K. L. Graesser
Byron Site Vice President
Byron Nuclear Power Station

Subscribed and sworn to before me, this 24th day of

March, 1998.

"OFFICIAL SEAL"
Tracey Fluck
Notary Public State of Illinois
My Commission Expires 1-13-00

Attachments A - Description and Safety Analysis of the Proposed Changes
B - The proposed changes to Byron Technical Specifications
C - Evaluation of Significant Hazards Consideration
D - Environmental Assessment

cc: Regional Administrator-RIII
Byron Project Manager-NRR
Senior Resident Inspector-Byron
Office of Nuclear Safety-IDNS

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGE

A. SUMMARY OF PROPOSED CHANGE

Commonwealth Edison Company (ComEd) proposes to revise Technical Specifications (TS) Surveillance Requirements 4.6.1.1.c, 4.6.1.2.a, 4.6.1.2.b, and the Bases to permit a deferral that allows the 10 CFR 50, Appendix J Type A testing of the Byron Unit 2 containment to be deferred 10 months. The test will be extended beyond the interval allowed in the Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J" which is endorsed by Regulatory Guide 1.163, "Performance Based Containment Leak Test Program" to Fall 1999 (B2RO8).

B. DESCRIPTION OF THE CURRENT REQUIREMENT

Technical Specification Surveillance Requirements 4.6.1.1.c , 4.6.1.2.a, 4.6.1.2.b, and the Bases require containment leakage rate testing in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B. The requirements for Type A testing are stated in NEI 94-01, which is endorsed by Regulatory Guide 1.163. Based on the ILRT Performance history, the current Type A test interval at Byron Unit 2 is 48 months plus 15 months which ends the current ILRT surveillance period in December 1998.

C. BASES FOR THE CURRENT REQUIREMENT

Maintaining containment integrity ensures that release of radioactive materials from the containment atmosphere will be restricted to leakage paths and associated leak rates assumed in the accident analysis. Containment leakage testing ensures that the leakage rate will not exceed that identified in the accident analysis at the peak accident pressure. As an added conservatism, the individual Type B and C tests are performed on a periodic basis to detect possible degradation of the containment leakage barriers between Type A tests.

D. NEED FOR REVISION OF THE REQUIREMENT

The current Type A test requirement requires an ILRT at Byron Unit 2 by December 1998. Since ILRTs are performed during Unit outages, this would require Byron 2 to either perform an ILRT in the April, 1998 outage (B2RO7) or force a Unit 2 shutdown before December of 1998 to perform the test. Currently a Type A Leak Test is not in the Byron Unit 2 April 1998 outage schedule.

During the time that Byron 2 had opportunity to schedule an ILRT, ComEd was actively engaged with NRC Staff in pursuit of a request to adopt a 10 year test interval for Type A testing on Byron 2 and Braidwood 2. Starting in July of 1997 (See July 25, 1997, J. Hosmer letter to USNRC Document Control Desk) ComEd requested an exemption under 10 CFR 50.12 to allow ComEd to perform Type A testing of Byron Unit 2 and Braidwood Unit 2 containments at least once per 10 years based on a single successful

Type A test, rather than two successful Type A tests. This request was converted, at the request of NRC Staff, to a License Amendment request (Referenced Letter) to again adopt the 10 year test interval for Unit 2 at Byron and Braidwood. Approval of this request would have allowed the next Type A test to be performed in September, 2003 for Byron 2.

Based on discussions with the Staff, it was apparent that the Referenced amendment may not have been granted prior to the Byron Unit 2 surveillance expiration date of December 1998. As a result, Byron would be forced to perform the test during the B2R07 outage or conduct a forced outage to perform the surveillance. Therefore, because this test was never included in the B2R07 outage and because it is resource and schedule intensive, ComEd is providing this additional amendment request to defer the 10 CFR 50, Appendix J Type A testing of the Byron Unit 2 containment approximately 10 months.

Past containment Type A testing and surveillance practices demonstrate that the underlying purpose of 10 CFR 50, Appendix J, Option B can be met with an extended Type A test interval. Postponing the performance of a Type A test to the next refueling outage for Byron 2 will not impact the health and safety of the public.

E. DESCRIPTION OF THE PROPOSED CHANGE

ComEd proposes to revise TS Surveillance Requirement 4.6.1.1.c , 4.6.1.2.a, 4.6.1.2.b, and the Bases to note that requirements are modified by an approved schedular exemption in the ILRT testing interval. The Bases for 3/4.6.1.2 will describe the approved extension. Specifically, Byron Unit 2 would extend the ILRT test interval by 10 months to the Fall of 1999. The changes are shown in Attachment B.

F. BASES FOR THE PROPOSED CHANGE

ComEd believes that extending the Type A test interval by 10 months for Byron Unit 2 will not impact containment leakrate and will not pose any undue risk to the public. The integrity of the Byron 2 containment is demonstrated through prior testing history, the design of the secondary side steam generator manways, operating practices related to secondary side leakage, and the behavior of the secondary side of the steam generators during accident conditions. In particular:

Type A Test Performance History

The first Byron Unit 2 Type A test following the successful pre-operational test was conducted in September 1990. During the operating cycle preceding the test, a steam generator secondary manway leak developed. The test was performed in the as-found condition, so no repairs were affected prior to the test. During this test, a steam generator manway leaked. The leakage was isolated by closing the Main Steam Isolation Valve and pressurizing the secondary side of the steam generator to approximately one psig below the containment test pressure. Initially the test was considered successful since the steam generator pressure is greater than containment pressure expected during post-accident, peak pressure conditions. The manway leak was not considered a post-accident leakage path. However, following consultation with Nuclear Reactor Regulation, Region III informed

ComEd that the manway leakage should be treated as valve leakage, and the as-found test was considered a failure. The next Type A test, conducted in September 1993, was completed successfully on both an as-found and a performance basis.

Secondary Side Design Issues

The secondary manways of the steam generators are periodically removed during refueling outages to support such maintenance activities as inspections and sludge lancing. At the end of each refueling outage, a walkdown of components inside containment is performed as the unit approaches the normal operating parameters of 557 °F and 2235 psig in the Reactor Coolant System. Under these conditions, the secondary side of the steam generators is pressurized to greater than 1000 psig. This walkdown identifies any manway leakage prior to commencement of power operation. Leakage would be assessed and repaired as needed prior to returning the unit to service.

The ability to detect secondary side manway leakage during normal plant operations, which have the potential to result in failure of the Type A test, was evaluated. The evaluation was performed on Byron Unit 2 secondary side manway closures, using original Westinghouse design conditions, to assess the relative susceptibility of the closures to leakage during normal plant operation and during accident conditions that are simulated during Type A testing.

The first case evaluates the manway closure during normal plant operating conditions. The secondary side original design conditions are 990 psia acting to unload the closure gasket at 543°F. The second case used to represent accident conditions and conditions during Type A testing is cold bolt-up. In the second case the only load on the gasket is from the bolt-up torque. The external pressure normally acting on the manway during an accident and Type A testing is neglected because it is small compared to normal operating conditions and neglecting the load, acting to load the gasket, is conservative in this evaluation.

The results of the evaluation show that the gasket load is 8.6% higher during accident conditions and Type A testing than during normal plant operation. Therefore, it can be concluded that the secondary manways would be more susceptible to leak during normal plant operation than during an accident and Type A testing.

Operating Practices and Secondary Side Leakage

During operation, containment parameters are routinely monitored. Any leakage from secondary manways would condense and be transported to the Containment floor drain sump via the floor drain system. This sump is equipped with a weir plate which is capable of detecting a one gallon per minute increase in sump input within one hour. This ensures that any unexpected input is promptly identified and assessed. This assessment would identify if leakage from the secondary side of the steam generators has developed.

Currently at Byron the monitoring of flow rate into the containment floor drain is accomplished through routine daily surveillances. In addition, a control room alarm (Containment Floor Drain Leak Detect Flow High) activates on floor drain flow of 1 gpm. Upon receipt of this alarm, operating surveillance RF-1, "Containment Floor Drain Monitoring System - Non Routine Surveillance," is initiated which identifies and trends leakage. The assessment of secondary side leakage is typically accomplished through the ComEd "Problem Identification Form" (PIF) process. This process will identify issues of operability, corrective action, and disposition of degraded equipment.

Behavior of the Secondary Side Post Accident

The potential for a leaking steam generator (SG) manway to become a significant containment bypass path is extremely small for the following reasons:

For leakage to occur by the SG manway several things must happen. First, a hole in the manway gasket must develop during operation to establish a leakage path for air to flow through it once the plant is shutdown. Secondly, for air to flow through the leak path a negative pressure gradient across the manway must occur, i.e., the pressure in the containment must be larger than the pressure in the SG. Thirdly, the path from the secondary side of the SG to the outside must be large and unconstricted enough to allow air and/or steam to exit

For leakage to occur by the SG manway, a hole in the manway gasket must develop during operation to establish a leakage path for air to flow through it once the plant is shutdown. For a hole to develop in the manway gasket, a weakness in the gasket material must occur subsequent to start-up and pressurization of SG secondary side. As previously discussed, if a hole in the gasket occurred prior to start-up, walk downs during start-up would identify potential leaks in the primary and secondary systems.

For leakage to occur by the SG manway, a negative pressure gradient across the manway must occur, i.e., the pressure in the containment must be larger than the pressure in the SG. To assess the transient pressure gradient across the SG manway, the transient pressure for the SG secondary side and the containment must be evaluated for 30 days.

The containment pressure and temperature responses for 30 days were evaluated as part of the design basis accident calculations for the plants. For this evaluation, the double-ended pump suction (DEPS) pipe break using minimum safety injection (SI) was used to provide the long term pressure and temperature responses. The DEPS pipe break produces the highest long term pressure response in the containment. Figure 1 provides the long-term containment pressure response and Figure 2 provides the short-term sump temperature response. The model used to represent the accident is shown on Figure 3. In the event of a pipe break of the reactor coolant loop, both safety injection and containment spray will be initiated. The DEPS pipe break was assumed to occur at the beginning of the accident. At the beginning of the accident, the residual heat removal (RHR) and containment spray systems' water supply is from the refueling water storage tank (RWST). The RHR pumps will automatically transfer to the recirculation mode when approximately 260,000 gallons of borated water have been pumped into the containment from the RWST. The charging and safety injection pumps are then manually aligned for the recirculation mode. The containment spray pumps will continue to operate until the RWST empty alarm is sounded when a total of approximately 415,000 gallons of borated water have been pumped. The operator will then manually align the containment spray pumps with the sumps for service in the recirculation mode. Final cool down of the containment, primary system and secondary system will primarily depend on reactor decay heat, containment structures heat transfer, fan cooler and RHR efficiency. As shown in Figure 1, the containment pressure is at approximately 7.8 psig at 1 day, 5.1 psig at 5 days and 3.33 psig at 30 days. This represents an average pressure of 5.2 psig for 30 days. This provides a very low pressure to drive the radioactive air/steam mixture through a manway leak hole into the SG secondary side. Depending on the cool down of the primary system, heat transfer from secondary to primary and the leakage through the main steam isolation valve, main steam safety valves, and the power operated relief valve, the SG secondary side will remain pressurized for an extended period after the accident.

The physical behavior of the steam generator secondary side in a post-loss of coolant accident (LOCA) situation will mitigate against initial inleakage from the containment. In a design basis accident (DBA) LOCA, the main steam isolation valves (MSIVs) rapidly isolate the SGs due to the containment high-pressure signal. The SGs will slowly depressurize as a result of the cool down of the primary side by the RHR heat exchangers. In the DBA LOCA for containment pressure, the DEPS break was postulated. After the break occurs, the primary system depressurizes and the primary water cools. Initially water injection to the primary side is from the RWST, at RWST low-level actuation, the safety injection path switches into recirculation. In the recirculation mode, the water is taken from the sump, pumped through a RHR heat exchanger, and back to the primary system where it picks up decay heat (See Figure 3). The containment pressure and sump temperature decreases very rapidly over the first 24 hours after a DBA LOCA and continues to decrease as shown in Figures 1 and 2. The sump temperature which is dictated by the decay heat and heat removal from the reactor, RHR heat exchanger performance and heat removal from the containment, can be used as a preliminary guide to the SG secondary side cool down. Based on the sump temperature given in Figure 2, the cool down of the SG secondary side will occur at about 6 to 12 hours after the LOCA. Assuming that the SG secondary side temperature leads the sump temperature by 90° F, at 12 hours the SG secondary side pressure is at 12 psig. So it can be conservatively assumed that the SG secondary pressure will be above the containment pressure for the first 6-12 hours after the LOCA and slowly decrease until the pressure is at 14.7 psia. This is a very conservative assumption since the primary water will not be held up in the SG U-Tubes because of the SG gravity drain down subsequent to the DEPS break. The lack of water in the SG U-Tubes will significantly reduce the heat transfer from the secondary to the primary side, which will help maintain secondary pressure. At 12 hours, the containment pressure is at 10.8 psig, which decreases to 3.3 psig at the end of 30 days. This low-pressure differential across the manway over a 30-day period substantially reduces the risk of a significant leak path to the SG secondary side.

For leakage to occur by the SG manway and to the outside, the path from the secondary side of the SG to the outside must be unconstricted enough to allow air and/or steam to exit. The isolation valves associated with the main steam system minimize leakage from the SG to the outside. The steam piping from the SGs to the turbine consist, among other things, of relief and safety valves, main steam isolation valves (MSIVs), and the turbine stop valves which are just upstream of the turbine. The MSIVs are located just outside the containment and provide isolation to the system in the event of a LOCA. The MSIVs, turbine stop valves, safety valves, and relief valves provide a very tortuous restricted path to the outside and would essentially eliminate flow to the outside especially at the low flows that one could expect from a depressurized containment and SG secondary side. Once the SG secondary side depressurized to 14.7 psia, the SG secondary side pressure source term would be provided by the manway leak path and the containment pressure. For example, assume the containment is at 5 psig; hence for a given manway leak path the pressure drop to the SG secondary side could be as much as 2-3 psid. Therefore, the pressure in the SG secondary side could be as low as 2 psig. The SG secondary side pressure at 2 psig would provide a very low flow to the outside (i.e., estimated to be approximately 67 SCFH), given the restrictive and tortuous path of the MSIVs and other associated valves. Based on this flow assessment, ComEd has concluded that the risk associated with the potential of a significant SG manway leak (i.e., manway leak large enough to impact offsite radiological dose) developing during operation at Byron is minimal.

Therefore, ComEd believes the prior testing history, the design of the secondary side steam generator manways, the operating practices related to secondary side leakage, and the behavior of the secondary side of the steam generators during accident conditions demonstrate that extending the test interval on the affected units will not represent an undue risk to health and safety of the public.

FIGURE 1 LONG TERM CONTAINMENT RESPONSE FOR DEPS-MINIMUM SI

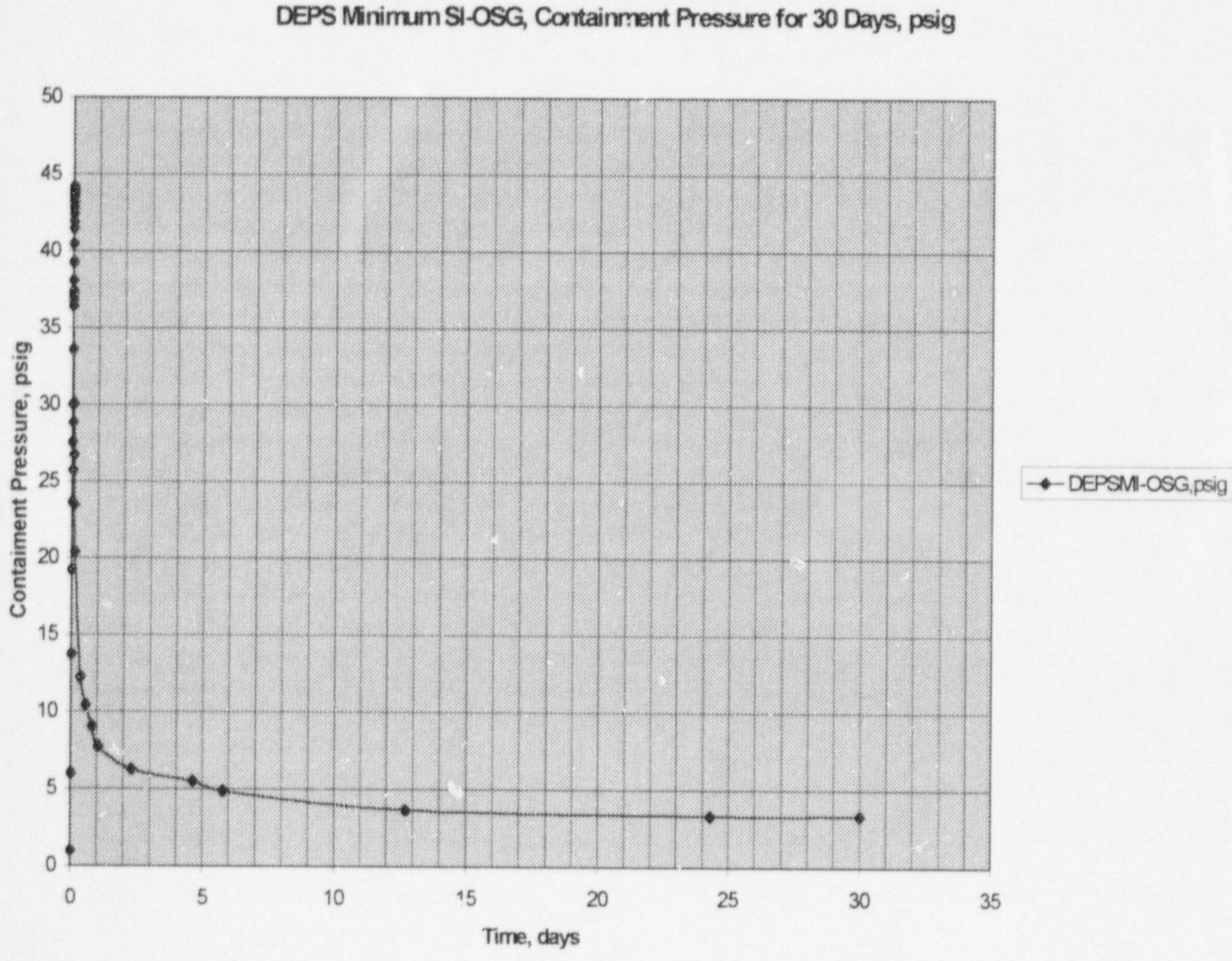

FIGURE 2 SHORT TERM CONTAINMENT SUMP TEMPERATURE FOR DEPS-MIN SI

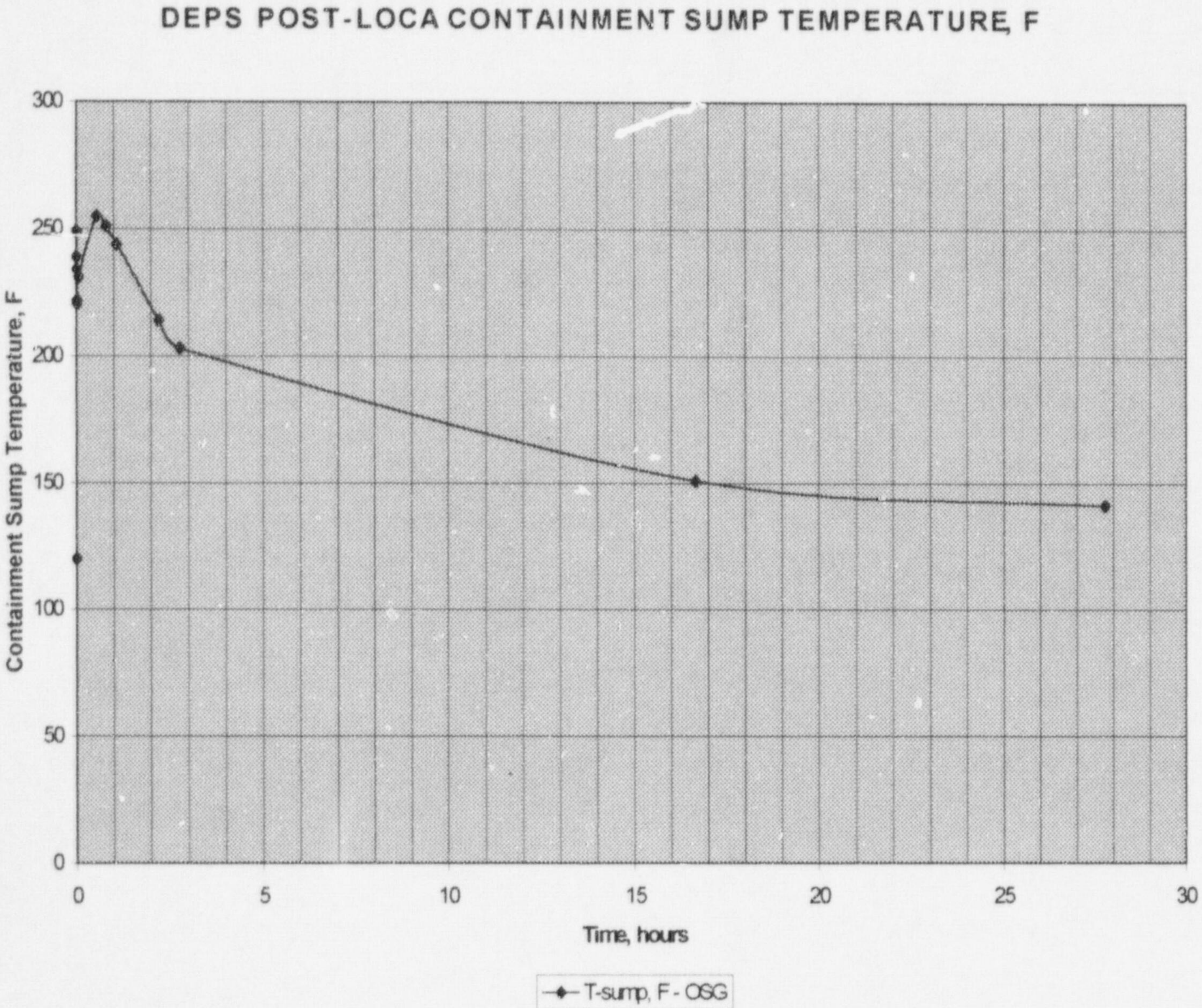

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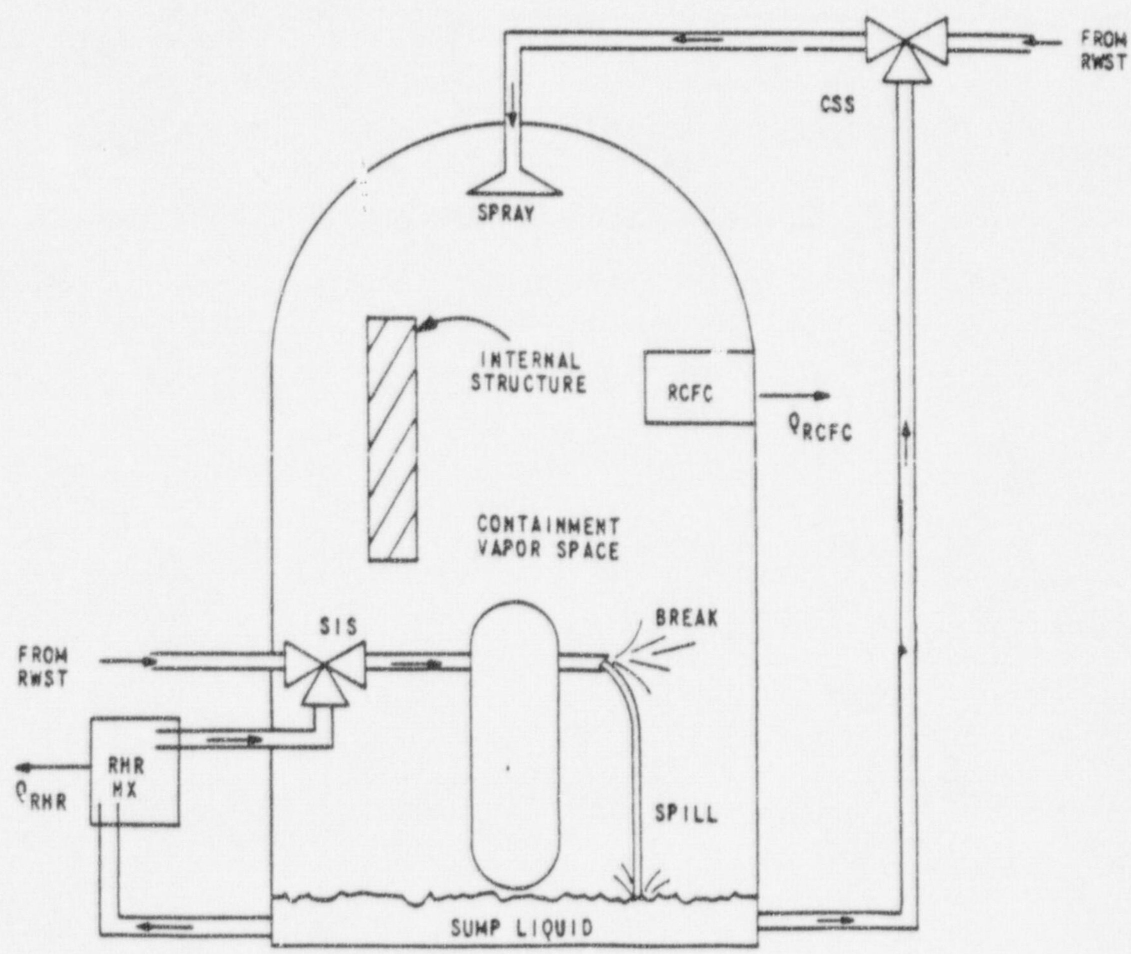
FIGURE 3 TYPICAL CONTAINMENT MODEL FOR POST-LOCA HEAT REMOVAL EVALUATION



RWST - REFUELING WATER STORAGE TANK

RHR HX - RESIDUAL HEAT REMOVAL HEAT EXCHANGER

SIS - SAFETY INJECTION SYSTEM

RCFC - REACTOR CONTAINMENT FAN COOLER

CSS - CONTAINMENT SPRAY SYSTEM

Q_{RHR} - RHR HEAT REMOVAL

Q_{RCFC} - RCFC HEAT REMOVAL

G. PROPOSED SAFETY ANALYSIS OF THE PROPOSED CHANGE

The function of the containment is to maintain structural integrity and leaktightness following a LOCA to ensure that the release of radioactive materials from the containment atmosphere will be restricted to those leak paths and associated leak rates assumed in the accident analysis. The limitations on containment leakage rates ensure that the values assumed in the accident analyses are not exceeded at peak accident pressures. Since the proposed change does not alter the plant design, but allows for an extension to the Type A test interval, there is no physical impact on containment structural integrity and therefore no direct increase in containment leakage. Although extending the test interval can increase the probability that a small increase in containment leakage could go undetected for a short period of time, the risk resulting from this proposed change is inconsequential as documented in NUREG-1493, "Performance-Based Containment Leak Test Program."

The purpose of the testing requirements of 10 CFR 50, Appendix J is to assure that (a) leakage through the primary reactor containment and systems and components penetrating containment does not exceed allowable leakage rate values as specified in the TS or associated Bases and (b) proper maintenance and repairs of the reactor containment penetrations and isolation valves are made during the service life of the containment.

Each of these requirements has been demonstrated both through prior Type A testing and maintenance practices. Even though the initial Type A test at Byron Unit 2 was conservatively classified as a failure, it was demonstrated that the as-left leakage through the primary reactor containment and systems and components penetrating containment did not exceed allowable leakage rate values as specified in the Technical Specifications or associated Bases. A successful Type A Test performed later further demonstrated the integrity of the containment and underlying requirements continued to be met.

Existing activities such as steam generator manway installation procedures, walkdown of components inside containment prior to commencement of power operation, and sump monitoring for leakage also support the requirements of 10 CFR 50, Appendix J. Additionally, the leakage mechanism responsible for the only failure at Byron, i.e., secondary manway leakage, is addressed on an 18 month frequency. Therefore, the probability for undetected leakage continues to be small.

The proposed license amendment will not reduce the availability of systems and components associated with containment integrity required to mitigate accident conditions. Containment leakage rates, parameters and accident assumptions would not be affected by the proposed license amendment.

The preceding discussion demonstrates that the underlying purpose of the rule can be met without the performance of an Type A test during the next refueling outage for each unit, and that the health and safety of the public will continue to be protected.

H. IMPACT ON PREVIOUS SUBMITTALS

This submittal is in addition to the request provided in a November 7, 1997 letter from J.B. Hosmer (Commonwealth Edison) to NRC Document Control Desk.

I. SCHEDULE REQUIREMENTS

ComEd respectfully requests that the NRC review and approve this license amendment request by April 27, 1998, to support outage scheduling activities at Byron Unit 2.