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8E.100a
JGC-224-94
August 12, 1994

Docket No. 50-461

10CFR50.90

Document Control Desk
Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station Proposed Amendment of
Facility Operating License No. NPF-62 (LS-94-006)

Dear Sir:

Pursuant to 10CFR50.90, Illinois Power (IP) hereby applies for amendment of Facility Operating License No. NPF-62, Appendix A - Technical Specifications, for Clinton Power Station (CPS). This request consists of a proposed change to Technical Specification 3/4.6.2.2, "Drywell Bypass Leakage," to allow drywell bypass leakage rate tests (DBLRTs) to be performed at intervals as long as ten years based on the demonstrated performance of the drywell structure. In the event of a DBLRT failure, the frequency would be required to be increased to at least once per 36 months. If the subsequent DBLRT meets the limit, the ten-year schedule may be resumed. In the event of two consecutive DBLRT failures, the frequency would be required to be increased to at least once per 18 months until two consecutive DBLRTs meet the limit, at which time the ten-year schedule may be resumed.

This request is being submitted as part of the cost beneficial licensing action (CBLA) program established within NRR where increased priority is granted to licensee requests for changes requiring NRC staff review that involve high cost without a commensurate safety benefit. Although the proposed change does have safety benefit (e.g., occupational dose reduction due to reduced testing), its major benefit is economic. Approximately 20 man-days of effort is required to set-up, perform, and evaluate the results of each DBLRT. During plant outages involving primary containment integrated leak rate tests (ILRTs), performance of a DBLRT requires approximately 15 hours of critical path time. During non-ILRT outages, the required critical path time increases to approximately 24 hours. Rental of the necessary air compressors and equipment necessary to conduct DBLRTs costs approximately $20,000. These costs exceed the threshold of $100,000 established under the CBLA program.

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A description of the proposed change and the associated justification (including a Basis For No Significant Hazards Consideration) are provided in Attachment 2. A marked-up copy of the affected pages from the current Technical Specifications are provided in Attachment 3. In addition, changes to IP's previous request to adopt the Improved Standard Technical Specifications (reference IP letter U-602196 dated October 26, 1993) are provided in Attachment 4. Further, an affidavit supporting the facts set forth in this letter and its attachments is provided in Attachment 1.

IP has reviewed the proposed change against the criteria of 10CFR51.22 for categorical exclusion from environmental impact considerations. The proposed change does not involve a significant hazards consideration, or significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, IP concludes that the proposed change meets the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

Due to the refueling outage safety improvement and significant resource savings that can be realized by implementation of this proposed change, IP is requesting that this application be reviewed on a schedule sufficient to support the fifth refueling outage currently scheduled to begin March 12, 1995.

Sincerely yours,

J. G. Cook
Vice President

DAS/csm

Attachments

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-690
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety

Attachment 1
to U-602320

J. G. Cook, being first duly sworn, deposes and says: That he is Vice President of Illinois Power; that the application for amendment of Facility Operating License NPF-62 has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said letter and the facts contained therein are true and correct.

Date: This 12 day of August 1994.

Signed: [signature]
J. G. Cook

STATE OF ILLINOIS } SS.
DeWitt COUNTY }

Subscribed and sworn to before me this 12th day of August 1994.

" OFFICIAL SEAL "
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Expires 11/24/97

Jacqueline S. Matthias
(Notary Public)

Background

As described in Updated Safety Analysis Report (USAR) Section 6.2.1, the Mark III containment design at Clinton Power Station (CPS) incorporates the drywell/pressure-suppression feature of previous BWR containment designs (Mark I and II) into a dry-containment type structure. The Mark III containment has three main features: (1) a drywell surrounding the reactor pressure vessel and a large part of the reactor coolant pressure boundary, (2) a suppression pool that serves as a heat sink during normal operational transients and accident conditions, and (3) a containment structure to prevent the uncontrolled release of radioactivity to the environment.

The drywell encloses the reactor pressure vessel (RPV), the reactor coolant recirculating loops, and branch connections of the Reactor Coolant System (RCS) which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a loss of coolant accident (LOCA) through the 102 horizontal vents in the drywell wall into the suppression pool. The steam is condensed in the suppression pool, and the air forced from the drywell is released into the primary containment (or wetwell). The pressure-suppression capability of the suppression pool assures that the peak LOCA temperature and pressure in the primary containment are kept below the design limits of 185 °F and 15 psig, respectively. The drywell also shields accessible areas of the containment from radiation originating in the reactor core and RCS.

The structural integrity of the primary containment is largely dependent on the drywell's ability to perform its safety function. Steam from a LOCA that bypasses the suppression pool would compress the air in the wetwell (i.e., the space between the exterior wall of the drywell and the interior wall of primary containment) and could result in excessive primary containment pressures. There are several potential sources of steam bypass leakage paths. These include potential cracks in the drywell concrete structure, the drywell vacuum breakers, and various penetrations through the drywell structure. Ventilation and piping penetrations are designed to ASME Code Class 2 and Seismic Category I requirements. These penetrations are also designed with two isolation valves in series with one valve in the drywell and another either outside containment or in the wetwell. High energy lines that extend into the wetwell, such as the main steam lines and feedwater lines, are encapsulated by guard pipes to direct energy to the drywell in case of a piping rupture. Electrical penetrations are sealed with a high strength/density material that will prevent leakage as well as provide radiation shielding.

The effect of steam bypass of the suppression pool on primary containment integrity has been evaluated, and this evaluation is described in USAR Section 6.2.1.1.5. The drywell leakage capacity was evaluated for a spectrum of primary system rupture sizes (areas), with and without containment spray and heat sinks. The limiting fault case was determined

to be a very small reactor system break which would not result in automatic reactor depressurization in conjunction with a loss of containment spray capability. The maximum allowable leakage path area (A/√k) for this case was calculated to be 0.02 ft^2. [Drywell bypass leakage area is expressed in terms of the parameter A/√k, where A is the flow area of leakage (ft^2) and k is the geometric and friction loss coefficient.] When containment spray and heat sinks are considered as a means of mitigating the effects of bypass leakage the allowable A/√k increases to 1.18 ft^2. This latter case is assumed in the design bases for the drywell since containment spray is designed to be available post-LOCA. An A/√k of 1.18 ft^2 is equivalent to a bypass leakage rate of 136,400 scfm at a drywell design pressure of 30 psig.

The preoperational test of the drywell at CPS confirmed the adequacy and conservatism of the drywell design. With the drywell pressurized to its design pressure of 30 psig, drywell structural strain and displacements were extensively monitored. Evaluation of the test results concluded that the drywell remained essentially elastic throughout the test and actual displacements were considerably smaller than predicted by design. Additionally, post-test examination of the drywell surfaces revealed only slight cracking of the concrete (which was determined to be insignificant) with no signs of distress or damage to either the concrete structure or the steel liner. The measured leakage rate was 1358 scfm (equivalent to an A/√k of 0.0067 ft^2), which is significantly less than the acceptance criterion of 13,640 scfm.

CPS Technical Specification (TS) 3/4.6.2.2, "Drywell Bypass Leakage," currently requires that a drywell bypass leakage rate test (DBLRT) be performed at least once every 18 months to verify that the steam bypass leakage area is less than or equal to 10% of the minimum acceptable A/√k design value of 1.18 ft^2. The corresponding allowable leakage rate limit is 4312 scfm with the drywell pressure at 3.0 psig. Testing is conducted at this reduced pressure so that the water level in the suppression pool remains slightly above the first row of horizontal vents. At CPS, the drywell bypass leakage rate is determined using the pressure decay test method.

Based on the excellent performance of the drywell structure to date, IP believes that a reduction in these testing requirements is warranted. The results for the DBLRTs conducted since initial plant startup, including four periodic tests (see Table 1), have revealed an A/√k that is two orders of magnitude less than the allowable limit. This testing involves approximately 20 man-days of effort to set up, perform, and evaluate the DBLRT test results. During plant outages involving primary containment integrated leak rate tests (ILRTs), performance of a DBLRT requires approximately 15 hours of critical path time. During non-ILRT outages, the required critical path time increases to approximately 24 hours. Rental of the necessary air compressors and equipment necessary to conduct DBLRTs costs approximately $20,000.

Description of Proposed Change

As described above, the CPS Technical Specifications currently require performance of a DBLRT at least once every 18 months. In accordance with 10CFR50.90, Illinois Power (IP) proposes to change TS 4.6.2.2 to allow the time interval between performances of the DBLRTs to be increased from 18 months to 10 years with requirements for more frequent testing if drywell performance degrades. Specifically, IP proposes that TS 4.6.2.2 be revised to read as follows:

> "4.6.2.2 The drywell bypass leakage rate test shall be conducted at least once per 10 years at an initial differential pressure of 3.0 psi and the $A/\sqrt{k}$ shall be calculated from the measured leakage. One drywell air lock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other leakage rate test. If any drywell bypass leakage test fails to meet the specified limit, the frequency shall be increased to at least once per 36 months; if the subsequent consecutive test meets the limit, the above 10-year schedule may be resumed. If two consecutive tests fail to meet the limit, a test shall be performed at least once every 18 months until two consecutive tests meet the limit, at which time the above test schedule may be resumed."

The proposed changes are reflected in the marked up copies of pages from the current TS contained in Attachment 3. In addition, as the NRC is currently reviewing IP's request to adopt the Improved Standard Technical Specifications (ITS) (reference IP letter U-602196 dated October 26, 1993), changes to that request are contained in Attachment 4 to reflect the changes proposed in this request.

Justification for Proposed Change

As described in USAR Section 6.2.6.5.1, the purpose of the DBLRT is to verify over the life of the plant that steam leakage bypassing the suppression pool for the full range of postulated primary system breaks is less than the maximum allowable design leakage. These tests are performed with the drywell isolated from containment and the air space exterior to the drywell near atmospheric pressure. The drywell is maintained at the test pressure (approximately 3 psig) for a minimum of 1 hour to allow the drywell atmosphere to stabilize. Afterwards, the drywell leakage rate is determined using the air flow or pressure decay methods. The drywell leakage rate is deemed acceptable if the measured leakage rate over 4 hours is less than or equal to 4312 scfm, which is 10% of the maximum allowable leakage at 3 psig.

The current testing requirements established for the DBLRT were originally based primarily on engineering judgment since no previous Mark III performance data were available. As such, Appendix A of Standard Review Plan Section 6.2.1.1.c (NUREG-

0800) conservatively established the test frequency as once every 18 months (or each refueling outage). Eight years of performance test data is now available to support a change to a ten-year DBLRT schedule and a performance-based approach for scheduling future performances of DBLRTs.

Historical Surveillance Results

Multiple bypass leakage tests have been performed since the initial startup of CPS. These tests have consistently proven the reliability of the drywell. The measured leakage rates over the six separate tests have been significantly less than both the allowable and design limits. Except for the initial low pressure drywell leakage test, bypass leakage has been less than 1% of the allowable limit and 0.1% of the design limit. Results of the previous CPS drywell bypass leakage rate surveillances are shown in Table 1.

Based on the historical test results, the calculated drywell bypass leakage area is of such a small magnitude that containment design pressure could not be exceeded even if containment spray and heat sinks were not available. The actual $A/\sqrt{k}$ is still less than 4% of the design $A/\sqrt{k}$ of 0.02 ft^2 for this limiting fault case.

The proposed change employs a performance-based approach in establishing the DBLRT frequency. The change to a 10-year interval for conducting this test is based on excellent performance of the drywell as demonstrated over the past eight years of plant operation. Future test results will then be used to provide the basis for continuing on the same schedule or increasing the frequency if a test failure occurs. The test frequency will be increased to at least once per 36 months in the event of one DBLRT failure and at least once per 18 months in the event of two consecutive failures. Resumption of the 10-year schedule would be allowed only after two subsequent consecutive tests have proven acceptable drywell performance.

As previously identified, drywell bypass leakage can occur through potential cracks in the drywell structure, the drywell vacuum breakers, and the various drywell penetrations. Additional cracking of the drywell is not expected during the remaining life of the plant based on the results of the structural integrity test conducted as part of the preoperational test program. The results of that testing showed that the stresses on the drywell structure while pressurized to 30 psig (design pressure) were essentially elastic with only slight cracking of the concrete surface. The drywell is typically exposed to less than 0.9 psig during normal plant operation and 3.0 psig (nominally) during DBLRTs. These pressures are considerably lower than the structural integrity test pressure and are less likely to initiate a crack or cause an existing crack to grow. Visual inspections of the drywell surfaces that have been performed each refueling outage since the structural integrity test have not revealed the presence of any additional cracks.

Leakage through the post-LOCA vacuum relief valves is minimized by the use of two valves in series that are normally sealed shut by the slightly higher pressure in the drywell. These valves are verified to be in the closed position at least once per day per CPS TS 4.6.5.a.

Leakage through other drywell penetrations is also minimized by the use of two valves in series that are either normally closed or can be closed automatically. These penetrations have one isolation valve located inside the drywell and one isolation valve located outside the drywell. The outer isolation valve is located either in the wetwell or outside primary containment. Drywell penetrations with isolation valves outside primary containment are also designated as primary containment penetrations and are local leak rate tested (LLRT) in accordance with 10CFR50 Appendix J. This testing ensures that the total leakage through these penetrations is kept below the allowable limits of the LLRT program, which are much lower than allowed for drywell leakage.

The remaining drywell penetrations, although not subject to LLRTs, are typically isolated by use of globe valves or gate valves that have excellent low pressure sealing capabilities. Notable exceptions to this are the Reactor Recirculation (RR) pumps seal supply check valves and the drywell vent and purge (VQ) system butterfly valves. The RR seal supply lines are designed to ASME Code Class 2 requirements up to and including the outboard drywell isolation valve. Drywell bypass leakage through this line is not likely since it would require two failures to occur - a passive failure of the ASME Code Class 2 piping between the RR pump and the first isolation valve and an active failure of both check valves to seat properly. The internal diameter of these lines is 3/4 inch and would also serve to limit the amount of drywell bypass leakage. The drywell vent and purge system drywell isolation valves are large Posi-seal butterfly valves that are normally closed when the plant is in Mode 1, making them essentially passive isolation devices. Although the CPS TS permit opening the 10-inch (1VQ005) and 24-inch (1VQ002) drywell vent and purge exhaust isolation valves to be open for up to five hours per 365 days, this allowance is typically utilized only during plant shutdown in preparation for refueling outages. The drywell vent and purge supply isolation valves are periodically verified to be sealed closed as required by the CPS TS (reference TS 4.6.2.7.1).

Risk Evaluation

Reduction of the DBLRT frequency as proposed slightly increases the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. An analysis was conducted to determine the potential risk to the public from not quantifying the magnitude of potential drywell bypass leakage as often as currently required. Based on this analysis, under several different accident scenarios, the risk of radioactivity release from the containment was found to be negligible, about 10^{-9} per year.

This risk is further reduced since an increase in bypass leakage would be detected early by reduction or cessation of normal drywell pressurization from leaks from the instrument air system or normal operation of pneumatic controls and operators as further described below.

On-Line Monitoring Capability

Due to the demonstrated leak tight performance of the drywell, CPS is able to continuously monitor the integrity of the drywell during normal plant operation. This is possible due to the existence of small instrument air system leaks and from normal operation of pneumatic controls and operators that pressurize the drywell at rate of approximately 0.04 psi/hr creating a differential pressure between the drywell and primary containment. The drywell is currently vented approximately once per day when pressure approaches the upper TS limit of 1.0 psid. Based on application the ideal gas law and known data, such as the drywell pressurization rate and the drywell leakage measured during the fourth refueling outage (RF-04), the total amount of instrument air inleakage was calculated to be between 21.5 and 22.5 scfm. The rate of drywell pressurization has remained essentially constant since drywell closeout in RF-04. This steady drywell pressurization rate allows qualitative monitoring of the drywell leakage rate. A rise in this rate, would be indication of an increase in the instrument air system inleakage, since it is improbable that the drywell became more leak tight. Conversely, a fall in this rate would be evidence of a larger drywell leakage area, but still much smaller than the allowable $A/\sqrt{k}$. This is because the maximum drywell leakage that would still maintain a differential pressure between the drywell and wetwell must be less than the instrument air inleakage rate of 23 scfm. The $A/\sqrt{k}$ for a 23 scfm leak at 0.2 psid is 0.0025 ft^2 or 0.2 % of the allowable leakage area. Therefore it has been concluded that as long as the drywell continues to pressurize, regardless of the rate, drywell integrity is always assured. The ability to qualitatively assess the integrity of the drywell during normal plant operation provides further support to extending the test interval.

Basis for No Significant Hazards Considerations

Illinois Power (IP) proposes to change the current Clinton Power Station (CPS) Technical Specifications to allow the time interval for performance of the drywell bypass leakage rate testing to be increased from at least once per 18 months to at least once per 10 years. The testing would be performed more frequently if drywell integrity degrades. In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazards consideration if operation of the facility in accordance with the proposed change would not (1) involve a significant increase in the probability or consequences of any accident previously evaluated, (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. This request is evaluated against each of these criteria below.

(1) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak tight integrity of the drywell, a structure used to mitigate the consequences of a loss of coolant accident (LOCA). The function of the drywell is to channel the steam released from a LOCA through the suppression pool, limiting the amount of steam released to the primary containment atmosphere. This limits the containment pressurization due to the LOCA. The leakage of the drywell is limited to ensure that the primary containment does not exceed its design limits of 185°F and 15 psig. Because the proposed change does not alter the plant design, only the frequency of measuring the drywell leakage, the proposed change does not directly result in an increase in the drywell leakage. However, decreasing the test frequency can increase the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. There are several potential sources of steam bypass leakage paths. These include potential cracks in drywell concrete structure, the drywell vacuum breakers, and various penetrations through the drywell structure. Based on the results of the structural integrity test conducted as part of the preoperational test program, additional cracking of the drywell is not expected during the remaining life of the plant. Ventilation and piping penetrations (including the drywell vacuum breaker penetrations) are designed to ASME Code Class 2 and Seismic Category I requirements. These penetrations are designed with two isolation valves in series with one valve in the drywell and another either outside primary containment or in the wetwell. High energy lines that extend into the wetwell, such as the main steam lines and feedwater lines, are encapsulated by guard pipes to direct energy to the drywell in case of a piping rupture. Electrical penetrations are sealed with a high strength/density material that will prevent leakage as well as provide radiation shielding. Operational experience has shown that the leak tightness of the drywell has been maintained well below the allowable leakage limits. In fact, the calculated drywell bypass leakage area is of such a small magnitude that containment design pressure could not be exceeded even if containment spray and heat sinks were not available. The TS limit of 10% of the design $A/\sqrt{k}$ provides margin for degradation. Drywell performance data to date suggest that drywell degradation, even during a ten-year interval between tests, will not exceed this margin.

Further, an analysis was conducted to determine the potential risk to the public from unacceptable drywell bypass leakage going undetected as a result of the proposed change. Based on this analysis, under several different accident scenarios, the risk of radioactivity release from containment was found to be negligible, about 10^{-9} per year.

Based on the above, IP has concluded that the proposed change will not result in a significant increase in the consequences of any accident previously evaluated.

(2) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. Thus, the proposed change cannot create the possibility of an accident not previously evaluated.

(3) The proposed change only affects the frequency of measuring the drywell leakage and does not change the bypass leakage limit for the drywell. However, the proposed change can increase the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. Operational experience has shown that the leak tightness of the drywell has been maintained well below the allowable leakage limits. In fact, the calculated drywell bypass leakage area is of such a small magnitude that containment design pressure could not be exceeded even if containment spray and heat sinks were not available. Further, an analysis was conducted to determine the potential risk to the public from the proposed change. Based on this analysis, under several different accident scenarios, the risk of radioactivity release from containment was found to be negligible, about 10^{-9} per year. As a result, IP has concluded that the proposed change will not result in a significant reduction in the margin of safety.

Based on the foregoing, IP concludes that this request does no involve a significant hazards consideration.

Table 1

PREVIOUS RESULTS OF CPS DRYWELL BYPASS LEAKAGE RATE TEST

TEST DATE	LEAK RATE (at 3.0 psig)	RATIO TO DESIGN LIMIT	CALCULATED $A/\sqrt{k}$
01/86	273.0 scfm[1]	0.63%	0.0075 ft^2
11/86	20.8 scfm	0.05%	0.0006 ft^2
04/89 (RF01)	19.0 scfm	0.04%	0.0005 ft^2
02/91 (RF02)	21.9 scfm	0.05%	0.0006 ft^2
05/92 (RF03)	18.0 scfm	0.04%	0.0005 ft^2
11/93 (RF04)	30.2 scfm	0.07%	0.0008 ft^2

[1] The leakage rate from the initial test was primarily attributed to a defective electrical penetration seal that was later repaired. Subsequent tests have found the drywell leakage to consistently be between 18 and 30 scfm.