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Docket No. 50-461 10CFR50.90

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

Subject: Application for Amendment of Facility Operating License No. NPF-62 for Clinton Power Station (LS-97-006)

Dear Madam or Sir:

Pursuant to 10CFR50.90, Illinois Power (IP) hereby applies for amendment of Facility Operating License No. NPF-62, Appendix A - Technical Specifications (TS), for Clinton Power Station (CPS). This request consists of proposed changes necessary for implementation of a feedwater leakage control system (FWLCS) mode of the residual heat removal (RHR) system. Specifically, the requested changes are to add a new TS Limiting Condition for Operation (LCO) 3.6.1.9, "Feedwater Leakage Control System (FWLCS)," complete with associated ACTIONS and SURVEILLANCE REQUIREMENTS. Additionally, a new TS Surveillance Requirement (SR) is being proposed for TS LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," to periodically verify that leakage through the primary containment feedwater penetration isolation valves due to pressurization from the FWLCS will remain within the limits assumed in the supporting analyses. Similarly, a change to TS SR 3.6.2.3.2 is requested to account for system changes resulting from the addition of the FWLCS. This amendment, if approved, will change the periodic leakage testing requirement for the primary containment feedwater penetration isolation valves such that a water leakage test would be performed in lieu of the presently required air leakage test. The motivation for pursuing this amendment is to enhance the isolation capability of the primary containment feedwater penetrations.

It is important to note that two changes are being made to the dose assessment model described in Chapter 15 of the CPS Updated Safety Analysis Report (USAR) as a result of the proposed change. The first is the use of ICRP 30 dose conversion factors for calculating the thyroid inhalation dose from airborne iodine, and the other change is that credit is taken for removal of iodine by suppression pool scrubbing. The acceptability and methodology of these changes is more fully described in Attachment 6.

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A description of the proposed changes and 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 TS is provided in Attachment 3. A marked-up copy of the affected pages from the current TS Bases is provided in Attachment 4. A detailed description of the technical bases for the FWLCS mode is provided in Attachment 5. A description of the radiological impact of the proposed change is provided in Attachment 6. Further, an affidavit supporting the facts set forth in this letter and its attachments is provided in Attachment 1. Following NRC approval of this request, IP will revise the CPS TS Bases, in accordance with the TS Bases Control Program of TS 5.5.11. Changes to the CPS TS Bases, consistent with the proposed TS changes, are provided for information in Attachment 4.

IP will be installing the FWLCS modification during the current outage, which is presently anticipated to end in the first quarter of 1999. IP desires to have the subject license amendment effective upon startup from this outage period. As such, IP respectfully requests review and approval of this amendment in a manner that would support the proposed implementation time frame.

IP has reviewed the proposed changes against the criteria of 10CFR51.22 for categorical exclusion from environmental impact considerations. The proposed changes do not involve a significant hazards consideration, or significantly increase the amounts or change the types of effluents that may be released offsite, nor do they significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, IP concludes that the proposed changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

Sincerely yours,

Walter G. MacFarland

Walter G. MacFarland, IV
Senior Vice President and
Chief Nuclear Officer

JFK/krk

Attachments

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

Walter G. MacFarland, IV, being first duly sworn, deposes and says: That he is Senior Vice President and Chief Nuclear Officer for Clinton Power Station; that this 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 23rd day of October 1998.

Signed: Walter G. MacFarland

Walter G. MacFarland, IV

STATE OF ILLINOIS
DEWITT COUNTY } SS.

"OFFICIAL SEAL"
Thomas B. Elwood
Notary Public, State of Illinois
My Commission Expires [illegible]

Subscribed and sworn to before me this 23rd day of October 1998.

Thomas B. Elwood

(Notary Public)

Background

Regulatory Basis/Requirements

General Design Criteria (GDC) 16, 50, 52, 53, 54, 55, 56, and 57 of 10CFR50, Appendix A provide the requirements for containment design, leakage testing and inspection, and containment isolation. These criteria (especially GDC 16) ensure that the reactor containment and associated systems provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. The feedwater system falls under the requirements of GDC 55 as an influent line which penetrates the primary containment and connects directly to the reactor pressure vessel.

In order to ensure offsite doses remain below those previously evaluated in the event of a design basis accident, leakage from the primary containment must be limited. To ensure that containment leakage remains within acceptable limits, periodic leakage rate tests must be performed. 10CFR50.54(o) requires primary reactor containments for water cooled power reactors to be subject to the leakage rate testing requirements set forth in Appendix J to 10CFR50.

Approval for CPS to adopt 10CFR50, Appendix J - Option B, was granted by Amendment 105 to the CPS Operating License and incorporated into the Technical Specifications as TS 5.5.13, "Primary Containment Leakage Rate Testing Program." Compliance with TS 5.5.13 includes following the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." This Regulatory Guide recognizes NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," prepared by the Nuclear Energy Institute, as providing methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10CFR50 (subject to certain limitations).

Per Section 6.0 of NEI 94-01, a local leak rate test (LLRT) is not required for boundaries that are sealed with a qualified seal system. NEI 94-01 invokes ANSI/ANS-56.8-1994, including Section 3.3.1(2), which also recognizes that primary containment boundaries not requiring an LLRT include "boundaries sealed with a qualified seal system." The definition of a "qualified seal system" is contained in ANSI/ANS-56.8-1994 and is as follows: "A system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 P_{ac} for at least 30 days following the DBA." Section 3.4, "Qualified Seal System Testing Requirements," of ANSI/ANS-56.8-1994, states "Primary containment barriers sealed with a qualified seal system are not required to be local leakage rate tested. If a seal system is used as a primary containment barrier, it shall be periodically tested to prove its functionality. This functional test shall demonstrate that the seal system is capable of sealing the primary containment barrier(s) with the sealing liquid at a differential pressure of not less than 1.1 P_{ac} for at least 30 days following a DBA. Qualified seal system testing is as specified in the plant's licensing basis."

Design Basis

At CPS, the feedwater lines are part of the reactor coolant pressure boundary as they penetrate the containment and drywell, and connect to the reactor pressure vessel. Each of the two feedwater line primary containment penetrations incorporates three isolation valves in series. (This portion of the feedwater system is shown on the piping schematic diagram included in Attachment 5.) The isolation valve inside the drywell is a simple check valve [1B21-F010A(B)], located as close as practicable to the drywell wall. Outside the primary containment is an air-assisted check valve [1B21-F032A(B)] located as close as practicable to the containment wall. Farther away from the primary containment is a motor-operated gate valve [1B21-F065A(B)]. This arrangement is designed such that, should a break occur in the feedwater line, the check valves prevent a significant loss of reactor coolant inventory and offer immediate isolation. The air-assisted check valve is "power assisted" closed and is actuated by the protection system. During the postulated loss-of-coolant accident, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the outermost valve does not automatically isolate upon a signal from the protection system. However, this valve is capable of being closed from the control room to provide long-term leakage protection when continued makeup from the feedwater source is unavailable or unnecessary.

The current CPS licensing basis requires the feedwater penetrations to be subject to a Type C leak test (air test performed at $\geq P_a$) pursuant to TS SR 3.6.1.3.8 and 10CFR50 Appendix J. In addition, since the feedwater lines traverse secondary containment without terminating, leakage through these penetrations is considered to be secondary containment bypass leakage for purposes of primary containment leakage accounting. The combined leakage rate for all secondary containment bypass leakage paths is required to be less than or equal to 0.08 L_a. Thus, the feedwater containment isolation valves are subject to stringent leak rate acceptance criteria.

Applicable Accident Analyses

There are three major design basis accident analyses described in the CPS USAR for which the feedwater penetration isolation valves are assumed to provide a significant mitigative function. These three postulated loss-of-coolant accident (LOCA) scenarios are: (1) recirculation line break (i.e., the design basis loss-of-coolant accident), (2) feedwater line break inside containment, and (3) feedwater line break outside containment.

(1) Recirculation Line Break (RLB)

As described in the CPS USAR, the postulated instantaneous guillotine rupture of a reactor recirculation line produces the highest peak containment pressure (P_a) and worst postulated offsite dose consequences. For the plant design basis accident (DBA) a complete loss of offsite power is assumed concurrent with the RLB.

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines set forth in the NRC Standard Review Plan (SRP) 15.6.5 and Regulatory Guides 1.3, Rev. 2 and 1.7, Rev. 2. Specific values of parameters used in this evaluation are presented in USAR Table 15.6.5-1.

With respect to the assumptions of the fission products released from the fuel, it is assumed that 100% of the noble gases and 50% of the iodine are released from an equilibrium core operating at a power level of 3039 MWt for 1000 days prior to the accident. While not specifically stated in Regulatory Guide 1.3 the assumed release of 100% of the core noble gas activity and 50% of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (as discussed in USAR Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100% of the noble gases and 50% of the iodine become airborne. The remaining 50% of the iodine is removed by plate-out and condensation, therefore, it is not available for airborne release to the environment. The activity airborne in the containment is presented in USAR Table 15.6.5-2, and in Attachment 6.

Regarding fission product transport to the environment, the transport pathways include leakage from the containment to the secondary containment which is discharged to the environment through the Standby Gas Treatment System (SGTS), and leakage from the containment directly to the environs (i.e., secondary containment bypass leakage). These pathways are further discussed in Attachment 6.

The calculated exposures (dose consequences) for the original design basis analysis are presented in USAR Table 15.6.5-6 and are within the guidelines of 10CFR100.11. The radiological analysis of the effects of this event on personnel in the main control room is discussed in USAR Section 15.6.5.5.3, and also in Attachment 6.

(2) Feedwater Line Break Inside Containment (FLBIC)

The CPS USAR Section 6.2.1.2 analysis of the postulated FLBIC accident scenario shows that the containment sub-compartment pressurization effects of this accident are less pronounced than the effects of the recirculation line break accident scenario. The CPS USAR postulates a guillotine type rupture of the feedwater line in the annular space between the reactor pressure vessel and the biological shield wall as the worst location for a break.

For LOCAs resulting from a postulated pipe break inside the containment, including the FLBIC, Section 15.6.5 of the CPS USAR notes that the most severe nuclear system effects and the greatest release of radioactive material to the containment result from the complete circumferential break of one of the two reactor recirculation loops, i.e., the DBA LOCA. Thus, the consequences of the FLBIC are bounded by the DBA-LOCA (recirculation line break) which was previously discussed.

(3) Feedwater Line Break Outside Containment (FLBOC)

The FLBOC described in the CPS USAR, Section 15.6.6, postulates an instantaneous circumferential break in the piping outside containment, upstream of the 1B21-F032A(B) check valves. The two check valves 1B21-F010A(B) and 1B21-F032A(B)) in the feedwater lines are assumed to terminate reactor coolant flow from the reactor vessel through the break. Initiation of the emergency core cooling systems (ECCS) maintains the reactor water level above the low-low-low level 1 trip and eventually restores it to the normal level. Thus, no fuel damage is expected to occur. For this reason, and because the SRP does not specify or address a source term for this accident, no quantitative design basis analysis of the radiological consequences for this event is presented in the CPS USAR. The qualitative design basis discussion in the USAR notes that the feedwater line break outside the containment is less limiting than either the steam line break outside the containment (analysis presented in USAR Sections 6.3 and 15.6.4) or the feedwater line break inside the containment (analysis presented in USAR Sections 6.3.3 and 15.6.5), and that it is much less limiting than the design basis accident (the recirculation line break) analysis presented in USAR Sections 6.3.3 and 15.6.5.

<u>Feedwater Leakage Control System - Overview</u>

The purpose of this license amendment request is to incorporate into the CPS licensing basis a new system, the feedwater leakage control system (FWLCS). The FWLCS will provide a means to seal the primary containment feedwater penetration isolation valves, thereby changing the leakage rate testing requirement for these valves to a periodic functional water test, in lieu of the presently required air test. The FWLCS will be a qualified seal system as defined and described in ANSI/ANS-56.8-1994. Regulatory Guide 1.96 establishes the requirements for the design and analytical evaluations of the effectiveness of the MSIV leakage control system. In the absence of a specific regulatory guide for the feedwater leakage control system, the guidance of RG 1.96 was applied to the design for the FWLCS.

The water for sealing the feedwater penetrations will be supplied from the suppression pool via the residual heat removal (RHR) system. The two subsystems of the FWLCS will be divisionally separate such that failure of one division will not impact the ability of the remaining division to completely establish a water seal on both trains of the feedwater system. The system will be able to be manually initiated approximately 20 minutes after a DBA LOCA (i.e., when the reactor vessel/coolant system pressure is reduced to a sufficiently low pressure), such that the associated feedwater piping will be completely filled within one hour of accident initiation. Attachment 5 to this letter contains a detailed description, simplified piping diagram, and design bases considerations associated with the implementation of the FWLCS.

The result of incorporating the FWLCS into the CPS licensing basis will be that periodic primary containment feedwater penetration leakage testing will be performed using the methods presented in ANSI/ANS-56.8-1994, Section 3.4, for a periodic functional water test in lieu of the currently required air leakage test.

Description of Proposed Technical Specification Changes

In accordance with 10CFR50.90, Illinois Power (IP) proposes the following change to the CPS TS:

1) Under new Technical Specification (TS) 3.6.1.9, "Feedwater Leakage Control System (FWLCS)," LCO 3.6.1.9 is proposed such that it requires two FWLCS subsystems to be OPERABLE.

2) An Applicability of MODES 1, 2, and 3 is proposed such that both FWLCS subsystems must be OPERABLE when primary containment integrity is required.

3) Required Actions for when one FWLCS subsystem is inoperable, and for when both FWLCS subsystems are inoperable, are proposed. A Completion Time of 30 days is proposed for the former (Condition "A"), and a Completion Time of 7 days (to restore one FWLCS subsystem to OPERABLE status) is proposed for the latter (Condition "B"). With either of these Required Actions not met within its required Completion Time, proposed Condition "C" is entered, wherein plant shutdown is required such that MODE 3 must be entered within 12 hours (Required Action "C.1") and MODE 4 must be entered within 36 hours (Required Action "C.2").

4) One Surveillance Requirement under the FWLCS Technical Specification is proposed. Specifically, SR 3.6.1.9.1 would require, at least once every 18 months, performance of a system functional test of each FWLCS subsystem.

5) An additional Surveillance Requirement is proposed for TS LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)." SR 3.6.1.3.12 would require, at least once every 18 months, verification that the combined leakage rate for both primary containment feedwater penetrations is less than or equal to 3 gallons per minute when pressurized to greater than or equal to 1.1 P_a. This proposed SR will be modified by a NOTE that will only require it to be met in MODES 1, 2, and 3.

6) Additionally, due to considerations associated with RHR water being diverted due to FWLCS operation during operation in the suppression pool cooling mode, a change to SR 3.6.2.3.2 is proposed to revise the required RHR pump flow rate for this mode of operation to $\geq$4550 gpm through the associated RHR heat exchanger to the suppression pool.

A copy of the proposed TS changes is provided in Attachment 3. In addition, changes to the CPS TS Bases, consistent with the proposed TS changes, are provided for information in Attachment 4.

Justification for Proposed Changes

As noted previously, the proposed FWLCS will supply a qualified seal system to the isolation valves in the primary containment feedwater penetrations. Following adoption of the proposed change, these penetrations will be water-sealed to prevent the post-accident primary containment atmosphere from leaking through this potential pathway. The FWLCS consists of two independent, manually initiated subsystems, either of which is capable of preventing airborne fission product leakage from the containment via the feedwater lines (after the lines are filled). Each subsystem uses an RHR pump (taking suction from the suppression pool) and a header to provide sealing water for pressurizing the feedwater piping either between the inboard and outboard feedwater line isolation check valves, or between the outboard feedwater line isolation check valve and an additional outboard motor-operated gate valve. The proposed changes to the CPS Technical Specifications to incorporate the LCO and Surveillance Requirements for the FWLCS subsystems ensure that the FWLCS subsystems will be maintained OPERABLE as required and that appropriate action is taken in the event one or both FWLCS subsystems are determined to be inoperable.

Proposed LCO 3.6.1.9 and its Applicability would require both FWLCS subsystems to be OPERABLE whenever the plant is in Modes 1, 2, and 3. Two FWLCS subsystems must be operable so that in the event of an accident, at least one subsystem is operable assuming a worst-case single active failure. A FWLCS subsystem is operable when all necessary components are available to pressurize each feedwater piping section with sufficient water pressure to preclude containment leakage (following the time period required to fill and pressurize the feedwater piping sections) when the containment atmosphere is at the maximum peak containment pressure. In Modes 1, 2, and 3 a DBA could cause a release of radioactive material to primary containment. In Modes 4 and 5, the probability and consequences of such an event are reduced due to the pressure and temperature limitations of these Modes. Therefore, the FWLCS is not required to be operable in Modes 4 and 5 to prevent leakage of radioactive material from primary containment.

With regard to operability requirements for the containment isolation valves associated with the feedwater penetrations, no changes are proposed. Since the feedwater penetrations are secondary containment bypass leakage paths, the valves will continue to be required to be operable in Modes 1, 2, and 3 as well as during movement of irradiated fuel assemblies in the primary or secondary containment, during core alterations, and during operations with a potential for draining the reactor vessel. The leakage limits for the feedwater penetrations are, however, only required to be met in Modes 1, 2, and 3 per the Note associated with proposed SR 3.6.1.3.12. That is consistent with SR 3.6.1.3.8 which is applicable for the remaining secondary containment bypass leakage paths.

The proposed Conditions, Required Actions and Completion Times address having one or both FWLCS subsystems inoperable. An allowed out-of-service time (i.e., Completion Time) of 30 days for one FWLCS subsystem inoperable (Condition "A") has been determined to be acceptable on the basis of a low probability of occurrence of a DBA LOCA, the amount of time available after the event for operator action, the low probability of failure of the OPERABLE FWLCS subsystem, and the availability of the primary containment isolation valves (PCIVs). This permits a reasonable time to restore OPERABILITY relative to the risk of having only one FWLCS subsystem OPERABLE. Similarly, the Completion Time of 7 days for having both FWLCS subsystems inoperable (Condition "B") is based on the low probability of the occurrence of a DBA LOCA, the availability of operator action, and the availability of the PCIVs. If neither of these Required Actions can be met within the specified Completion Times, Condition "C" applies such that the plant must be brought to a MODE in which the LCO does not apply. In that regard, Required Action C.1 requires placing the plant in MODE 3 within 12 hours and Required Action C.2 requires placing the plant into MODE 4 within 36 hours. The allowed completion times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. These allowed out-of-service times are also consistent with those specified in the TS for the MSIV-LCS.

The Surveillance Requirement proposed for the FWLCS ensures or verifies that the FWLCS will operate through its required operating sequence. This includes verifying that the automatic positioning of valves and the operation of each interlock is correct, and that the necessary check valves open. The 18-month test interval specified in the "Frequency" column for proposed SR 3.6.1.9.1 is consistent with other Technical Specification Surveillance Requirements for similar tests, and is based on the need to perform this Surveillance under the conditions that apply during a plant outage due to the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

The additional Surveillance Requirement proposed for TS LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," ensures that the combined leakage rate of the primary containment feedwater penetrations is less than the specified leakage rate. The leakage rate is based on water as the test medium since these penetrations will be sealed by the FWLCS. Verifying the combined leakage rate is within its limit (i.e., ≤ 3 gpm) when pressurized to greater than or equal to 1.1 P_a will provide assurance that the assumptions in the radiological evaluations are met. Meeting the specified leakage limit has been shown by testing and analysis to bound the condition following a design basis loss-of-coolant accident where, for a limited time, both air and water could be postulated to leak through this pathway. The leakage rate of each primary containment feedwater penetration is assumed to be the maximum pathway leakage, i.e., the leakage through the worst of the two isolation valves [either 1B21-F032A(B) or 1B21-F065A(B)] in each penetration. The 18-month test interval specified in the "Frequency" column for proposed SR 3.6.1.3.12 is consistent with other testing used to verify PCIV leakage. Proposed SR 3.6.1.3.12 will be modified by a Note that will only require this SR to be met in Modes 1, 2, and 3.

The proposed change to SR 3.6.2.3.2 is based on consideration of the heat removal requirements necessary to maintain the design basis suppression pool temperature under postulated accident conditions. The currently specified flow rate through the RHR heat exchanger of 5050 gallons per minute (gpm) is overly restrictive such that when accounting for water diverted to the FWLCS when an RHR subsystem is operating in the suppression pool cooling mode, and accounting for flow measurement inaccuracies during performance of the surveillance test, the 5050 gpm value may not be attainable. As such, an analysis was performed to determine a more appropriate flow rate through the RHR heat exchanger such that the design basis for the suppression pool cooling mode of RHR would be maintained, and yet, adequate sealing water flow to the FWLCS would be provided. This analysis concluded that an RHR flow rate of 4550 gpm through the RHR heat exchanger would be capable of maintaining the design basis suppression pool temperature within limits during postulated accident conditions.

Radiological Considerations

As stated previously, the purpose of this license amendment request is to obtain NRC approval to adopt the use of a qualified seal system for the primary containment feedwater penetrations, thereby changing the testing requirement for these penetrations to a periodic functional water test in lieu of the presently required air test. The motivation for pursuing this amendment is to enhance the isolation capability of the primary containment feedwater penetrations.

As noted previously, the introduction of the FWLCS and corresponding changes to the leak test requirements for the feedwater containment isolation valves affects or potentially affects the consequences of three important design basis accidents evaluated in plant safety analyses (the CPS USAR), i.e., the DBA LOCA (RLB), FLBIC and FLBOC. Evaluation of the impact on the consequences of these postulated events is provided in Attachment 6. Since, from a containment leakage and offsite/onsite dose point of view, the DBA LOCA is the most limiting of these events (as discussed in Attachment 6), the radiological analysis for the impact on dose consequences is primarily focused on the DBA LOCA.

For the purposes of performing a conservative DBA LOCA dose analysis, the feedwater lines were assumed to remain empty for the entire time period (one hour) immediately following the accident until the FWLCS is assumed to completely fill the feedwater piping. This analytical approach is conservative on several accounts. No credit was taken for the blowdown phase of the feedwater piping during the reactor vessel depressurization associated with the LOCA (even though a significant water inventory could be assumed to exist in the feedwater piping during and just after feedwater pump coastdown). No credit was taken for a partial water seal on the valve seating surfaces while the FWLCS is filling the piping, and full back-leakage of sealing water is assumed to begin coincident with the initiation of the accident.

As a result of the proposed change (introduction of the FWLCS and corresponding changes to the leak test requirements for the feedwater containment penetrations), and based on the subsequent re-analyses performed for the DBA LOCA, several changes to the consequences of the postulated DBA LOCA were identified. For the MCR, the thyroid dose decreased from 27 rem to 8.6 rem (30 rem limit), the γ-whole body dose increased from 2.0 rem to 3.5 rem (5 rem limit), and the β-skin dose increased from 14.3 rem to 17.1 rem (30 rem limit). For the EAB, the thyroid dose increased from 163 rem to 198 rem (300 rem limit, and the γ-whole body does increased from 4.4 rem to 9.8 rem (25 rem limit). For the LPZ, the thyroid dose decreased from 156 rem to 75 rem (300 rem limit), and the γ-whole body dose increased from 1.7 rem to 3.0 rem (25 rem limit).

It is important to note that the supporting dose analyses for this proposed amendment employed two changes to the methodology that was originally used to calculate the post-accident dose consequences for CPS. These two changes or methods are (1) crediting suppression pool scrubbing for reduction of the radioiodine fission products assumed to be released from the containment following a DBA LOCA, and (2) utilization of the new dose conversion factors specified in ICRP 30. Approval of this amendment will incorporate these two methodologies into the licensing basis for CPS, and as such, future dose consequence analyses may utilize these methodologies in whole or in part to support plant activities or changes performed pursuant to the allowances of 10CFR50.59. Attachment 6 contains a more detailed explanation of the dose analyses performed in support of this proposed license amendment.

<u>Basis For No Significant Hazards Consideration</u>

According to 10CFR50.92, a proposed change to the 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 an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed changes are evaluated against each of these criteria below.

1) The proposed change implements a method of providing a qualified sealing system for the primary containment feedwater penetration isolation valves. This water-sealing function, i.e., the FWLCS, constitutes a new operating mode of the Residual Heat Removal (RHR) system. The FWLCS introduces new piping that constitutes an extension of the reactor coolant system (RCS); however, such piping is designed to the same requirements as other RCS piping and as such introduces no significant increase in the probability of any accident previously evaluated. Notwithstanding, a postulated line break in any of the new FWLCS piping would not, by itself, introduce any new effects or consequences not already bounded by postulated line-break or LOCA events previously evaluated in the USAR. Since the proposed change does not affect any parameters or conditions

that contribute to the initiation of any accidents previously evaluated, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the primary containment designed to mitigate the consequences of a loss-of-coolant accident (LOCA). Once the FWLCS mode has been initiated and a water seal for the seating surfaces of the primary containment feedwater penetration isolation valves has been established (within one hour after the accident), post-LOCA primary containment atmosphere will be prohibited from leaking through the feedwater penetrations and thus bypassing the secondary containment.

Calculations of post-accident (DBA LOCA) doses affected by this change use accepted ICRP 30 dose conversion factors and take credit for suppression pool scrubbing. Suppression pool scrubbing is effective in reducing iodine release but has no assumed effect on the removal of noble gases. Since the methodology and assumptions for scrubbing are acceptable to the NRC per the guidance in SRP Section 6.5.5 and the values for decontamination factors are conservative, considerable margin is preserved within the analysis. However, these calculations show increases in some of the previously evaluated post-accident doses when compared with dose calculations performed as part of the initial plant licensing basis. Although some of the newly calculated post-accident doses are larger than those that were previously approved, the increases remain small enough to be within the acceptance limits given in 10CFR50, Appendix A, GDC 19 and in 10CFR100.11.

Since all of the newly calculated post-accident doses resulting from the proposed addition of a water sealing system for the feedwater primary containment penetration isolation valves are below the 10CFR50, Appendix A, GDC 19 and 10CFR100.11 acceptance limits, IP has concluded that the proposed change does not result in a significant increase in the consequences of an accident previously evaluated.

2) The proposed change institutes a new operating mode of the RHR system (the FWLCS mode). When this mode is established, it will reduce primary containment atmosphere leakage to the environment in the event of a LOCA. Flow diverted from the RHR system to the FWLCS has been evaluated, and has been determined to have no adverse impact on the capability of the RHR system to perform its intended safety functions. Further, the additional piping added for the FWLCS is designed to appropriate requirements for the RCS, thus ensuring that RCS integrity is maintained per design. Sufficient isolation between the RCS and the RHR low-pressure piping will also be maintained per the FWLCS design. Thus, no safety functions are altered or impacted as a result of this change. Installing, operating, or testing the components that support the FWLCS mode has no influence on, nor does it contribute to the possibility of a new or different kind of accident or malfunction from those previously analyzed. Because the USAR

analysis already assumes leakage through the feedwater primary containment penetrations following a design basis LOCA, and the subject change does not affect the type of accident(s) that are postulated to occur, the proposed change does not present the possibility of an accident of a different type. Additionally, the change in dose analysis methodology does not create an accident or malfunction of a different type since it only involves the analysis of the effects of accidents or malfunctions previously evaluated in the USAR.

Based on the above, IP has concluded that the proposed change will not create the possibility of a new or different kind of accident not previously evaluated.

3) The margin of safety impacted by the proposed change involves the dose consequences of postulated accidents which are directly related to the primary containment leakage rate, specifically those consequences associated with dose attributable to leakage through the feedwater lines which are secondary containment bypass leakage paths.

Although considerable conservatisms were included in the reanalysis, this reanalysis identified some dose values that increased above the previously licensed values as well as some dose values that decreased below the previously licensed values. However, all of the radiation dose consequences resulting from the proposed change will continue to be below the 10CFR50, Appendix A, GDC 19 and 10CFR100.11 acceptance criteria.

Except for providing a method of sealing the feedwater primary containment penetration isolation valves (and therefore the method of performing periodic leakage testing of these components) no other change in the method of primary containment leakage testing or secondary containment bypass leakage path testing is being proposed. All other primary and secondary containment bypass leakage testing will continue to be performed in accordance with existing Technical Specification requirements. Adequate programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment, systems and components penetrating the primary containment, and for all secondary containment bypass leakage paths.

As a result, IP has concluded that the proposed change will not result in a significant reduction in a margin of safety.

Based upon the foregoing, IP concludes that this proposed change does not involve a significant hazards consideration.