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July 25, 2017
GO2-17-028

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 LICENSE AMENDMENT REQUEST FOR CHANGE TO TECHNICAL SPECIFICATION 3.5.1, "ECCS-OPERATING"**

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Energy Northwest hereby requests a license amendment to revise the Columbia Generating Station (Columbia) Technical Specification (TS) 3.5.1, "ECCS – Operating". This amendment is requested to delete the Note associated with Surveillance Requirement 3.5.1.2, to reflect the Residual Heat Removal (RHR) system design and ensure the RHR system operation consistent with the TS 3.5.1 Limiting Condition for Operation (LCO) requirements.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in Enclosure 1 of this submittal.

The proposed TS markup pages are included as Enclosure 2 to this submittal. Markups of the proposed TS Bases are included for information only as Enclosure 3 of this submittal. Clean pages of the proposed TS changes are included as Enclosure 4 of this submittal.

This letter and its enclosures contain no regulatory commitments.

Approval of the proposed amendment is requested within one year of the date of the submittal. Once approved, the amendment shall be implemented within 60 days.

In accordance with 10 CFR 50.91, Energy Northwest is notifying the State of Washington of this amendment request by transmitting a copy of this letter and enclosures to the designated State Official.

If there are any questions or if additional information is needed, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 25th day of July, 2017.

Respectfully,

A.L. Javorik
Vice President, Engineering

Enclosures: As stated

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399 (email)
EFSECutc.wa.gov – EFSEC (email)
RR Cowley – WDOH (email)
WA Horin – Winston & Strawn

Evaluation of Proposed Technical Specification Change

1.0 SUMMARY DESCRIPTION

This evaluation supports a License Amendment Request (LAR) to Columbia Generating Station (Columbia) Technical Specification (TS) "ECCS - Operating". This TS change will delete the Note associated with Surveillance Requirement (SR) 3.5.1.2 to reflect the Residual Heat Removal (RHR) system design and ensure the RHR system operation consistent with the TS 3.5.1 Limiting Condition for Operation (LCO) requirements.

Implementation of this LAR will result in no physical modification to the plant. This proposed change has no adverse effect on the plant or plant safety.

2.0 Description of the Proposed the Change

The proposed change will delete the following Note associated with TS SR 3.5.1.2:

> Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than 48 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable.

Enclosure 2 provides the marked-up TS page with the proposed change indicated. Enclosure 3 provides the marked-up TS Bases page with the proposed change indicated for Columbia and is provided for information only. Enclosure 4 provides the proposed clean TS page.

3.0 TECHNICAL EVALUATION

3.1 System Design and Operation

The Columbia RHR system is designed to perform different and independent functions to support plant operation. As described in the Columbia Final Safety Analysis Report (FSAR), the safety function of the RHR system is to cool the reactor core by flooding after a loss-of-coolant accident (LOCA) (i.e., LPCI mode). The RHR system is also capable of cooling the suppression pool to safely terminate a post-LOCA containment temperature transient (i.e., suppression pool cooling mode). The containment spray cooling (CSC) mode of the RHR system is designed to provide sufficient cooling to maintain the containment drywell and wetwell temperatures and pressures within acceptable limits following a LOCA by discharging the water to the wetwell and drywell spray spargers. In addition, the RHR system may be used during a normal plant shutdown in the shutdown cooling (SDC) mode to remove residual heat from the nuclear system and cool the reactor coolant system.

The RHR system consists of three independent closed loops, each containing a motor driven pump, powered by an engineered safety feature (ESF) bus, and associated piping, valves, instrumentation and controls. Two of the independent RHR loops contain a heat exchanger with associated service water supply system to support the heat removal functions. The RHR pumps are sized on the basis of the flow required during the LPCI mode of operation. The heat exchangers are sized on the basis of required duty for the SDC mode.

The emergency core cooling system (ECCS) consists of the high-pressure core spray (HPCS) system, the low-pressure core spray (LPCS) system, LPCI subsystems, and the automatic depressurization system (ADS). The LPCI mode of RHR operation supports the ECCS safety objective to limit the release of radioactive materials following a LOCA. The LPCI function is capable of delivering a large flood of water into the core to refill the reactor pressure vessel (RPV) and provide core cooling at low RPV pressures. The three RHR pumps automatically start in LPCI mode upon receipt of an ECCS initiation signal. LPCI is a low-head, high-flow function that delivers flow to the RPV when the differential pressure between the RPV and drywell is less than 222 psid. The minimum rated flow is injected at 20 psid between the RPV and the drywell. LPCI is designed to reflood the RPV to at least two-thirds core height and to maintain this level. In the LPCI mode of operation, each RHR pump takes suction from the suppression pool through an independent suction line and discharges to the reactor core through separate RPV piping penetrations.

The SDC mode of the RHR system is operated during normal unit cooldown and shutdown to remove decay heat. The initial phase of nuclear system cooldown is accomplished by dumping steam from the RPV to the main condenser. When nuclear system temperature has decreased to where the steam supply pressure is not sufficient to maintain the turbine shaft gland seals, vacuum in the main condenser cannot be maintained and the RHR system is placed in the SDC mode of operation. The RHR SDC subsystem is able to remove decay heat to complete cooldown to 125°F in less than 25 hours after the control rods have been inserted and can maintain the nuclear system at or below 125°F. In the SDC mode of operation, the 'A' or 'B' RHR pump takes suction from the 'A' Reactor Recirculation (RRC) loop; and directs the flow through the RHR heat exchanger prior to returning the water back to the RPV through RRC pump discharge piping.

3.2 Current Technical Specifications Requirements

Columbia TS 3.5.1, "ECCS-Operating," requires each ECCS injection system be OPERABLE in Modes 1, 2, and 3. With regard to the LPCI mode, this means that all three LPCI subsystems are required for the LCO to be met. If one LPCI subsystem is inoperable, TS 3.5.1, Condition A requires the inoperable subsystem to be returned to OPERABLE status within seven days. The TS SR 3.5.1.2 requires periodic verification that each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This SR is modified by a Note that allows a LPCI subsystem to be considered OPERABLE for the LPCI function when the subsystem is being aligned or is operating in the SDC mode, and the unit is in Mode 3 below a reactor steam dome pressure of 48 psig. Utilization of this Note requires that the RHR system be capable of manual realignment to the LPCI mode and not be otherwise inoperable. The Note was added to TS SR 3.5.1.2 during Columbia's conversion to the Improved Technical Specifications (ITS), which was approved as Amendment 149 on March 4, 1997 (Reference 1). The allowance provided by the Note was considered acceptable because the return to operability entails only the repositioning of valves, either remote or locally, and the energy requiring dissipation in Mode 3, with a reactor steam dome pressure less than 48 psig is considerably less than that at 100% power with normal operating temperature and pressure.

Columbia TS 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," LCO requires that two RHR SDC subsystems to be OPERABLE; and when no RRC pump is in operation, one SDC subsystem must be in operation during Mode 3. An OPERABLE RHR SDC subsystem consists of one RHR pump, one heat exchanger, and the associated piping and valves. Each SDC subsystem is considered OPERABLE if it can be manually aligned (remote or local) to the SDC mode for removal of decay heat. In Mode 3, one RHR SDC subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. If one or both SDC subsystems are inoperable, TS 3.4.9, Condition A, requires immediate initiation of actions to restore one SDC subsystem to OPERABLE status.

TS 3.4.9 is only applicable in Mode 3, with reactor steam dome pressure less than 48 psig. Mode 3 means that the reactor mode switch is in the 'Shutdown' position and the average reactor coolant temperature is greater than 200°F. In this mode, all the RPV head closure bolts are fully tensioned and therefore, RPV pressure is typically above atmospheric pressure.

3.3 Reason for the Proposed the Change

Industry Operating Experience (OE) has determined that manually realigning an RHR SDC subsystem from SDC to LPCI could result in water flashing to steam in the RHR piping, water hammer, or pressure locking or thermal binding of valves unless the RHR SDC piping is first allowed to cool. A Boiling Water Reactor Owners' Group (BWROG) Ad Hoc committee recommended that the provision allowing LPCI to be considered operable when aligned for decay heat removal be removed and, if necessary, plants should enter the Action for an inoperable LPCI subsystem when it is aligned to RHR SDC in Mode 3 with reactor steam dome pressure less than 48 psig. This Action has a

7-day completion time, which is much longer than the time typically required to transition to Mode 4 from Mode 3 at less than 48 psig steam dome pressure.

This OE is applicable to Columbia in that the design of the RHR system could be susceptible to this same phenomenon. However, there are barriers in place that currently direct the Operations crews to declare the LPCI mode inoperable when susceptible. Columbia has built into the procedure that if reactor pressure is greater than 20 psig, RHR LPCI is required to be declared inoperable due to the potential for LPCI suction line flashing and creation of voids. The issue remains that the TS SR Note would allow declaring the system operable. The proposed amendment would allow Columbia to remove a TS SR Note that is not conservative.

The proposed change will not alter the physical design. Currently the TS SR Note could make Columbia susceptible to potential water hammer in the RHR system if in the SDC Mode of RHR in Mode 3 when swapping from the SDC to the LPCI mode of the RHR system. The proposed LAR will eliminate the risk for cavitation of the pump and voiding in the suction piping, thereby avoiding potential to damage the RHR system, including water hammer.

Therefore, the removal of the TS SR 3.5.1.2 Note, and operation with one RHR subsystem inoperable for LPCI mode while being aligned or operated in SDC in accordance with TS 3.4.9, is justified.

3.4 Impact on Submittals under Review by NRC

Energy Northwest has submitted a License Amendment Request to adopt TSTF-523, Revision 2, "Generic Letter 2008-01, Managing Gas Accumulation," which adds the following Note to TS 3.5.1.2:

> “Not required to be met for system vent flow paths opened under administrative control.”

The above Note is unrelated to the Note proposed to be deleted in this amendment request. Thus, the two amendment requests are not linked.

Energy Northwest has submitted a License Amendment Request to adopt TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing.” This change affects some of the same TS pages; however it does not affect the individual SR being revised in this amendment request. Thus, the two amendment requests are not linked.

4.0 REGULATORY EVALUATION

The Columbia FSAR Chapter 3 provides detailed discussion of Columbia's compliance with the applicable regulatory requirements and guidance.

The proposed TS amendment:

- Does not result in any change in the qualifications of any component; and
- Does not result in the reclassification of any component's status in the areas of shared, safety-related, independent, redundant, and physically or electrically separated.

4.1 Applicable Regulatory Requirements

4.1.1 10 CFR 50 Appendix A General Design Criteria (GDC)

The relevant GDCs are discussed below. The following NRC requirements and guidance documents are applicable to the review of the proposed changes.

10 CFR 50, Appendix A, General Design Criterion (GDC) 34, "Residual heat removal," requires that a system to remove residual heat be provided with a safety function to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

10 CFR 50, Appendix A, GDC 35, "Emergency core cooling," requires that a system to provide abundant emergency core cooling be provided with a safety function to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

10 CFR 10, Appendix A, GDC 37, "Testing of emergency core cooling system," requires that the emergency core cooling system design provide the capability for periodic pressure and functional testing. This testing shall assure (1) structural and leaktight integrity of components, (2) operability and performance of active components, (3) operability of the whole system under conditions as close to design as possible.

10 CFR 50.36, "Technical specifications," details the content and information that must be included in a station's Technical Specifications (TS). In accordance with 10 CFR 50.36, TS are required to include (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

As described in 10 CFR 50.36(c)(2), "Limiting conditions for operation," are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any other actions permitted by TS.

10 CFR 50.46(a)(1)(i) requires that each boiling or pressurized light-water nuclear power reactor be provided with an ECCS designed with a calculated cooling performance in accordance with an acceptable evaluation model following a postulated LOCA.

The proposed change does not involve any physical changes to the structures, systems, or components at Columbia. The proposed change will reflect current plant configuration of the RHR system design and assure safe operation by continuing to meet applicable regulations and requirements described above.

4.2 PRECEDENT

The NRC has approved similar license amendment requests to remove this Note from the TS for emergency core cooling system as follows:

Letter from R. B. Ennis (NRC) to M. J. Pacilio (EGC), "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Delete Non-Conservative Note from Limiting Condition for Operation for Operation 3.5.1 (TAC Nos. MF3184 and MF3185)," dated July 28, 2014 (ADAMS Accession Number ML14163A589).

Letter from B. K. Vaidya (NRC) to B. C. Hanson (EGC), “Lasalle County Station, Units 1 and 2, Issuance of Amendments Re: Revision of Technical Specifications Section 3.5.1, Emergency Core Cooling Systems Operating" (TAC Nos. MF5570 and MF5571) dated October 14, 2015 (ADAMS Accession Number ML 15244B410).

5.0 SIGNIFICANT HAZARDS CONSIDERATION

Energy Northwest requests amendment to Facility Operating License No NPF-21 for Columbia Generating Station.

The proposed amendment would delete the Surveillance Requirement (SR) Note associated with Technical Specifications Section 3.5.1, "ECCS - Operating," SR 3.5.1.2 to reflect current plant configuration and ensure the Residual Heat Removal system operation remains consistent with the Technical Specification Section 3.5.1 Limiting Condition for Operation (LCO) requirements.

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

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

   Response: No.

   No physical changes to the facility will occur as a result of this proposed amendment. The proposed change will not alter the physical design. The current Technical Specification Note could make Columbia susceptible to potential water hammer in the Residual Heat Removal system if in the Shutdown Cooling Mode of Residual Heat Removal in Mode 3 when swapping from the Shutdown Cooling to Low Pressure Core Injection mode of Residual Heat Removal system. The proposed License Amendment Request will eliminate the risk for cavitation of the pump and voiding in the suction piping, thereby avoiding potential to damage the Residual Heat Removal system, including water hammer.

   Therefore there is no significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change does not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Accordingly, the change does not introduce any new accident initiators, nor does it reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function. Deletion of the Technical Specification Note is appropriate because current Technical Specification could put the plant at risk for potential cavitation of the pump and voiding in the suction piping, resulting in potential to damage the Residual Heat Removal system, including water hammer.

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

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

Response: No.

The proposed change does not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Accordingly, the change does not introduce any new accident initiators, nor does it reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function. Deletion of the Technical Specification Note is appropriate because current Technical Specification could put the plant at risk for potential cavitation of the pump and voiding in the suction piping, resulting in potential to damage the Residual Heat Removal system, including water hammer.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

6.0 CONCLUSIONS

Based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the applicable regulations as identified herein, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 ENVIRONMENTAL CONSIDERATION

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

8.0 REFERENCES

1. Letter from T.G. Colburn (NRC) to J.V. Parrish (Energy Northwest formerly WPPSS), Issuance of Amendment for the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (TAC NO. M94226), dated March 4, 1997. Replacement of current technical specifications (TS with a set of TS based on NUREG-1434, “Standard Technical Specifications, General Electric BWR/6 Plants,” Revision1, April 1995.

2. Energy Northwest formerly Washington Public Power Supply System (WPPSS) Safety Evaluation Report (SER), NUREG-0892, dated April 13, 1984.

Enclosure 2

Proposed Columbia Technical Specification Changes (Mark-Up)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	~~-----NOTE-----~~ ~~Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than 48 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable.~~ ~~----------~~ Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify ADS accumulator backup compressed gas system average pressure in the required bottles is ≥ 2200 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source. SYSTEM / FLOW RATE / DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE LPCS / ≥ 6200 gpm / ≥ 128 psid LPCI / ≥ 7200 gpm / ≥ 26 psid HPCS / ≥ 6350 gpm / ≥ 200 psid	In accordance with the Inservice Testing Program

Enclosure 3

Proposed Technical Specification Bases Markup Pages
For Information Only

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 11. For a large break LOCA, failure of ECCS subsystems in Division 1 (LPCS and LPCI A) or Division 2 (LPCI B and LPCI C) due to failure of its associated diesel generator is, in general, the most severe failure. For a small break LOCA, HPCS System failure is the most severe failure. The small break analysis also assumes two ADS valves are inoperable at the time of the accident. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of Reference 12.

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The low pressure ECCS injection/spray subsystems are defined as the LPCS System and the three LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in 10 CFR 50.46 (Ref. 10) could potentially be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by 10 CFR 50.46 (Ref. 10).

~~LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below 48 psig reactor steam dome pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCS System, LPCS System, and LPCI subsystems full of water ensures that the systems will perform properly, injecting their full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring the lines are full is to vent at the high points. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves potentially capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~In MODE 3 with the reactor steam dome pressure less than 48 psig, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than 48 psig, a reduced complement of low pressure ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling, when necessary.~~

Enclosure 4

Proposed Columbia Technical Specification Changes (Re-Typed)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify ADS accumulator backup compressed gas system average pressure in the required bottles is ≥ 2200 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source. SYSTEM / FLOW RATE / DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE LPCS / ≥ 6200 gpm / ≥ 128 psid LPCI / ≥ 7200 gpm / ≥ 26 psid HPCS / ≥ 6350 gpm / ≥ 200 psid	In accordance with the Inservice Testing Program