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March 4, 2005
GO2-05-045

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397; ENERGY NORTHWEST RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST: CONTROL ROD DROP ACCIDENT ANALYSIS**

Reference: Letter dated October 12, 2004, DK Atkinson (Energy Northwest), to NRC, "License Amendment Request: Control Rod Drop Accident Analysis"

Dear Sir or Madam:

In the referenced letter, Energy Northwest submitted a request for amendment to the Columbia Generating Station Operating License NPF-21 for the Columbia Generating Station (Columbia). The proposed amendment requested NRC approval to update the Final Safety Analysis Report (FSAR) to reflect that the reactor core isolation cooling (RCIC) system is not required to mitigate the consequences of the control rod drop accident (CRDA). An NRC staff request for additional information regarding this submittal was provided to Energy Northwest by the NRC Licensing Project Manager for Columbia. The questions and the Energy Northwest response to the questions are attached.

If you have any questions or require additional information regarding this matter, please contact Mr. GV Cullen, Licensing Supervisor, at (509) 377-6105.

Respectfully,

WS Oxenford
Vice President, Technical Services
Mail Drop PE04

Attachments: 1. Response to the first issue in the Request for Additional Information
2. Response to second and third issues in the Request for Additional Information

cc: BS Mallett - NRC – RIV
BJ Benney - NRC – NRR
NRC Sr. Resident Inspector - 988C
WA Horin - Winston & Strawn
RN Sherman - BPA/1399

A053

NRC Issue:
(1) On Page 3 of 9, FSAR Discussion: The control rod drop accident

It is stated: "-------no significant RPV water level transient is anticipated." In the event of a RPV level transient during the accident, a safety grade water supply source is required to mitigate the accident. HPCS or RCIC is required for this. The reactor is required to be in a stable condition with adequate RPV level. Scram function will shutdown the reactor and as a result of the scram, MSIVs may close and the reactor pressure will increase, SRVs may open and the RPV level may decrease. Discuss the scenario and explain in detail why RPV level transient is not possible.

Response:
Two issues are important in the evaluation of the CRDA analysis: (1) which system, structures, or components (SSCs) are credited in the CRDA analysis to mitigate the event; and (2) following the accident, which SSCs are required for the plant to achieve and maintain safe shutdown.

Table 1 provides a brief historical summary of the CRDA analysis. As stated by General Electric and demonstrated by AREVA (also known as Framatome), the scram terminates the accident. The following information is provided to show that the emergency core cooling systems (ECCS) are available and adequate to reach safe shutdown. The discussions also show that the approach of using safety relief valves (SRVs) in conjunction with an operating low pressure ECCS as a back-up to the high pressure ECCS has been accepted by the NRC.

<u>CRDA Analysis</u>
In the CRDA analyses performed by the Columbia fuel vendors, the accident is terminated by the scram initiated by the average power range monitors (APRMs) in the reactor protection system (RPS). The subsequent release is minimized by the core design parameters that reduce core damage and the fission product confinement provided by the reactor coolant pressure boundary and the main condenser. The dropping control rod does not trigger the parameters for automatic initiation of the RCIC system, the high pressure core spray (HPCS) system, or any other ECCS. Accordingly, neither the analyses of core response nor the dose consequences address a change in reactor pressure vessel (RPV) water level.

<u>Safe Shutdown</u>
Following the termination of the accident (i.e., the automatic scram), RPV level may decrease. The HPCS system is designed to initiate on low RPV water level. However, in response to the decreasing level, the operators may manually initiate the HPCS system to preclude the automatic initiation. The automatic ECCS back-up to the HPCS system is provided by automatic depressurization system (ADS) initiation in conjunction with an operating low pressure ECCS.

The HPCS system is adequate to provide makeup inventory. If the HPCS system were not available, the ADS is also adequate to reduce the RPV pressure to a point that would allow injection by either the low pressure core spray (LPCS) system or the low pressure coolant injection (LPCI) system. These systems would remain in operation, as needed, to reach and maintain safe shutdown. If available, the shutdown cooling mode of the RHR system would be used to maintain cold shutdown. If not, the RHR heat exchanger(s) would be placed in operation to remove the decay heat added to the suppression pool through the opened SRVs.

<u>License Bases</u>
The use of ADS and an operating low pressure system as a back-up for the high pressure ECCS has been reviewed and accepted by the NRC for Columbia and other boiling water reactors. As discussed in FSAR Chapter 6, the loss of coolant accident (LOCA) analyses cover the spectrum of pipe break sizes. In the small pipe break LOCA, the vessel inventory make-up is provided by the HPCS system or, if necessary, by ADS in conjunction with an operating low pressure ECCS. The LOCA analyses do not model or credit RCIC system injection.

There are also several other examples of this acceptance documented in NRC reviews of fire protection programs. General Electric (GE) Nuclear Energy (NE) discussed the issue of depending on ADS and a low pressure ECCS in GE-NE-T43-00002-00-030R01, "BWROG Position on the Use of Safety Relief Valves and Low Pressure Systems as Redundant Safe Shutdown Paths." Although the document emphasis is on achieving a safe shutdown following a fire to comply with 10 CFR 50 Appendix R, the systems are capable of establishing the safe shutdown conditions regardless of the event (normal operation, anticipated operational occurrence, or design basis accident) that initiated the need to shut down the plant.

The NRC also addressed this issue for Columbia in a memorandum from Robert M. Bernero, Director - Division of BWR Licensing (NRC) to Dennis F. Kirsch, Director - Division of Reactor Safety and Projects (NRC), dated December 4, 1986. In the Safety Evaluation Report attached to that memorandum, the NRC states:

> "Nevertheless, the use of ADS and LPCI is an approved and accepted means of achieving and maintaining a safe shutdown condition. This methodology is used by other licensees and the use of ADS and LPCI is considered to be an acceptable alternative shutdown capability."

Plant Response to Loss of Normal Feed Systems

Although the RCIC system is not credited in the mitigation of the design bases accidents, Energy Northwest recognizes the importance, the usefulness, and the flexibility of the system in providing makeup inventory when the RPV is isolated from normal feed systems.

As a normal vessel inventory makeup system, the RCIC system is designed to respond to decreasing RPV level, automatically initiating at the same RPV level as the HPCS system. The need for the RCIC system to restore RPV water level is dependant on the availability of the normal feed systems. The ability of the RCIC system to respond, following the termination of the accident, is dependant on adequate steam flow from the RPV. The manual initiation of the RCIC system can preclude the need to restore RPV level with the HPCS system. Should the RCIC system be unavailable, adequate core cooling would be provided by the ECCS to mitigate the consequences of a loss of normal inventory. As discussed in FSAR Section 15.2.7:

> "Either the RCIC or the HPCS system is capable of maintaining adequate core coverage and will provide long-term inventory control. For the complete loss of feedwater flow event, operation of RCIC or HPCS is sufficient to avoid initiation of ADS on low vessel level (L1)."

Conclusion

The accident analyses performed by the fuel vendors demonstrate that a dropped control rod does not result in a significant RPV level reduction. The analysis of the dispersion and the resultant radiological consequences of the CRDA are not mitigated by and do not depend on the operation of any core cooling system to mitigate the release from the postulated damaged fuel. If needed, the HPCS system is adequate to provide reactor coolant inventory in response to a decrease in reactor water level following the termination of the accident. The RCIC system, if adequate steam flow is available, can also provide additional inventory for a period of time. If the HPCS or RCIC systems were not available, operators could manually reduce reactor pressure using the SRVs. When reactor pressure is low enough, the low pressure ECCS can be used to supplement the reactor coolant inventory. If the operators fail to take the manual actions, the automatic initiation of ADS and LPCS or LPCI system would provide the required core cooling. These systems would remain in use, providing core cooling and reducing RPV pressure, until systems are placed in service to maintain cold shutdown. The required operator actions are detailed in the Emergency Operating Procedures (EOPs) for reactor pressure and reactor level control.

In summary: (1) there is nothing specific about the CRDA that requires the initiation of the RCIC system; and (2) Columbia is fully capable of achieving and maintaining safe shutdown conditions following a CRDA or other events that isolate the reactor vessel from normal inventory feed systems without the operation of the RCIC system.

TABLE 1 General Electric / Columbia Historical Timeline

DOCUMENT	DATE	DISCUSSION
NEDO-10527	March 1972	This topical report discusses the CRDA analyses models. The report contains five major sections: Description of the accident, Discussion of the parametric results, Description of the excursion model, Verification of the adiabatic model, and Development of the experimental and analytical data used in the analysis. The discussion of the accident clarifies that the CRDA is analyzed in terms of peak fuel enthalpy, associated fuel failure, and radiological consequences considering operation of the plant protective and safety features as shown in Figure 2-2. The paragraph goes on to explain that the consequences are not discussed in this report but will be discussed in the individual plant applications. Figure 2-1 shows the various paths to safe shutdown. The Topical Report does not contain an analysis or a description of a model of the RCIC system or the HPCS system response to the dropped control rod.
NEDO-20360 *	April 1974	This is the generic reload application for 8x8 GE fuel. In the summary of the CRDA, the topical report reiterates the design bases considerations listed in NEDO-10527, page 2-2. The topical also states that the "scram terminates the accident."
GO2-83-660	July 26, 1983	In response to an outstanding issue regarding modifications to the Automatic Depressurization System (ADS), Energy Northwest stated: "As a result of this change, the ADS will become fully automatic and, in conjunction with the existing low pressure ECCS systems, will provide backup redundancy for the HPCS under all high pressure, level decreasing events. Thus, the reliance on the RCIC System as a HPCS backup for certain selected accidents is eliminated."

TABLE 1 General Electric / Columbia Historical Timeline

DOCUMENT	DATE	DISCUSSION
23A1862 *	September 1983	GE Design Spec for RCIC states that RCIC is designed to provide reactor water inventory under the following conditions: vessel isolated and in hot standby, vessel isolated and loss of reactor feedwater; and during shutdown operations and the normal feedwater system is lost before shutdown cooling can be placed in operation.
NUREG-0892, Supplement 4	December 1983	In Section 6.3.6, TMI Actions, II.K.3.18, the staff states that the design proposed in GO2-83-660 is acceptable with 3 conditions (i.e., the inhibit switch is addressed in EOPs, the inhibit switch is in TS, and the modification is complete before startup after the first refueling outage).
NRC internal IOM: VS Noonan to A Schwencer *	February 2, 1984	Provides the opinion that because RCIC system operation is not credited in any design basis analysis, the system can be removed from the EQ program.
GE-NE-208-17-0993 *	December 1994	In the power uprate analysis, GE states that the scram terminates the CRDA. (Accordingly, no analysis of the cooling system operation as a result of the dropped rod is necessary.) The RCIC system description does not address RCIC system response to a specific design basis accident, such as the CRDA.
SECY-93-067 *	March 1993	Discusses the fourth criterion for inclusion in Tech Spec for systems not recognized in the safety analyses, such as RCIC or SLC.
BWROG-00087 *	October 2000	In this letter to the NRC, GE states "In no case is RCIC relied upon to mitigate a design basis accident nor is it an engineered safety feature. Some FSARs describe RCIC in connection with the Rod Drop Accident (RDA) as potential water make up source. Following a postulated RDA, the RCIC (assuming HPCI/HPCS is inoperable) would provide inventory control/decay heat removal function in response to a reactor vessel isolation caused by the RDA. RCIC does not mitigate the RDA."

TABLE 1 General Electric / Columbia Historical Timeline

DOCUMENT	DATE	DISCUSSION
EMF-2863	September 2003	This is the Cycle 17 reload report for Columbia. The CRDA analysis in this report does not include the use of the RCIC system or the HPCS system to mitigate the accident. The methodology approved by the NRC that is used by AREVA for CRDA analysis is documented in XN-NF-80-19(P)(A). In summary, the analysis is based on CASMO-4/MICROBURN-B2 methodology. This code does not reference or model RCIC system initiation.

* Copy provided to staff in earlier fax

NRC Issue:
(2) Specify the analysis of record. Is it GE analysis or Framatome analysis? I understand that Columbia is now loaded with Framatome fuel.

Response:
The current core comprises a mixed core of ABB/CE/Westinghouse SVEA-96 fuel and Framatome ANP Atrium-10 reload fuel. The reload analyses of record, including the control rod drop accident (CRDA) were performed by Framatome. The dose consequences analysis was performed by General Electric in accordance with "Radiological Accident Evaluation - The CONAC03 Code," (NEDO-21143-1).

NRC Issue:
(3) What are the assumptions used in the analysis of record for CRDA?

Response:
The assumptions used in the analyses for the CRDA can be found in the Columbia FSAR Section 15.4.9.

Other assumptions, specific to core performance are described in the Framatome topical report, "Exxon Nuclear Mythology for Boiling Water Reactors Neutronic Methods for Design and Analysis," XN-NF-80-19(P)(A), Volume 1.

Assumptions related to the radiological dose assessment can be reviewed in the GE NEDO-21143-1, "Radiological Accident Evaluation – The CONAC03 Code."