ENERGY NORTHWEST RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST: CONTROL ROD DROP ACCIDENT ANALYSIS
Attachment 2

GE-NE-0000-0040-1432-00-01, REVISION 0

Historical Licensing Evaluation of Post-CRDA RCIC Operation
for Columbia Generating Station

GE Nuclear Energy

General Electric Company
1989 Little Orchard St., San Jose, CA 95125-1030

GE-NE-0000-0040-1432-00-01
Revision 0
Class I
May 2005

Historical Licensing Evaluation of Post-CRDA RCIC Operation For Columbia Generating Station

Prepared by:

Kurt Schaefer

Verified by:

Joel Friday

Approved by:

George Stramback

IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

Please Read Carefully

A. Disclaimer

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between the company receiving this document and GE. Nothing contained in this document shall be construed as changing the applicable contract. The use of this information by anyone other than a customer authorized by GE to have this document, or for any purpose other than that for which it is intended, is not authorized. With respect to any unauthorized use, GE makes no representation or warranty, and assumes no liability as to the completeness, accuracy or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

Historical Licensing Evaluation of Post-CRDA RCIC Operation For Columbia Generating Station

1. Background

The Reactor Core Isolation Cooling (RCIC) system is the replacement for the Isolation Condenser (IC) systems, used in the BWR/2s and early BWR/3s (e.g., Dresden and Millstone). It was not designed to act as or supplement an engineered safety feature (ESF), e.g., emergency core cooling system (ECCS), establish containment isolation or control the release of radioactive material, and thus, was not designed to prevent or mitigate an accident. A RCIC system is not capable of shutting down a plant. With respect to meeting 10 CFR 50 Appendix A single failure criteria, the RCIC is not needed to maintain a plant in a safe shutdown condition. Therefore, the RCIC is not required to perform any safety-related function.

The RCIC was designed to operate following anticipated operational occurrences (AOOs), which result in loss of feedwater flow such as an AOO that results in a containment isolation. As defined in 10 CFR 50 Appendix A, AOOs are classified as part of normal operations. Therefore, by regulatory definition, an AOO is not an accident.

Nonsafety-related and safety-related structures, systems and components (SSCs) may be credited for in the safety analyses of non-accident type events, such as AOOs and a safe shutdown fire (Appendix R) event. However, only safety-related SSCs may be credited in the accident safety analyses. In order to credit the use of the SSC, the performance must be part of a model or method of evaluation that has been approved by the NRC. The post-Control Rod Drop Accident (CRDA) core cooling scenario for all late BWR/3s & BWR/4-5s (before the Three Mile Island (TMI) accident related modifications) and current IC BWR/2-3s, by only ESF-ECCS equipment, is shown in Figure 1. The figure demonstrates that, regardless of the BWR model, all BWR/2-5s as originally designed would perform the post-CRDA cooling function without the use of RCIC and still meet the 10 CFR 50, Appendix A single failure criterion (SFC). The GE methodology for the analysis of a CRDA does not include, and has never included explicitly modeling any cooling system response, because maintaining water level is a post-CRDA normal shutdown function (see Section 1.1).

As defined in Regulatory Guide (RG) 1.70, Chapter 6 (Engineered Safety Features) and NUREG-0800 Standard Review Plan (SRP) 6.1.1, an ESF directly mitigates the (radiological) consequences of an accident. However, no cooling system is capable of mitigating the radiological consequence of any accident that has terminated. Like for all BWRs, the Columbia RCIC system is not classified as an ESF system.

For a CRDA scenario, no cooling system is used or required to mitigate a radiological consequence. In the core performance analysis, the CRDA is terminated by the scram. Following the termination of the CRDA, maintaining adequate water level, which is only assumed and not modeled, prevents the event scenario from escalating. The fission product release from the damaged fuel is assumed to be transferred to the condenser by steam flow from the reactor. The fission product release tc the environment is from the condenser. Neither the RCIC system nor any cooling system has the capability to mitigate the consequences of this release.

A CRDA only has the potential to damage a significant amount of fuel at power levels < 10% rated thermal power (RTP). At that power level range, RPV water level is maintained by the feedwater pumps or the condensate/condensate booster pumps, which would be available regardless if a CRDA occurred or not. Feedwater-condensate flow would prevent the water level from decreasing to the point where the High Pressure Core Spray (HPCS) and RCIC systems would actuate, and no RPV depressurization and low pressure ECCS would be needed. Therefore, unless a CRDA occurred simultaneously with a loss of the feedwater/condensate systems, HPCS, RCIC and/or low pressure ECCS would not actually be used to maintain water level following a CRDA.

Prior to the Three Mile Island (TMI) accident (pre-TMI) related plant modifications, Automatic Depressurization System (ADS) initiation required both low RPV water level along with high drywell pressure. The CRDA is an example of an event that does not result in high drywell pressure. The pre-TMI core cooling paths are shown in Figure 1, and are independent of the normal or abnormal event scenario (AOO, DBA, Appendix R fire, safe shutdown earthquake, etc.) that results in a RPV water level decrease.

The original licensing safety analysis basis for all BWR/2-5s allows credit for operator action 10 minutes after any abnormal event has initiated. Therefore, after assuming no operator action for 10 minutes, taking credit for operator action(s) to depressurize the RPV (allowing LPCS and LPCI initiation), following an accident, is appropriate. Because the minimum operating shift staffing requirement is two operators and a shift supervisor, under the above circumstances, it is not realistic to assume that all three would fail to manually depressurize the RVP. Following the termination of the CRDA, RPV level would decrease, if the Feedwater-Condensate systems did not maintain water level. The RCIC system and the HPCS system would initiate if the water level dropped to the low low water level (Level 2). Although neither system is included in the CRDA analysis, Figure 1 shows, after the CRDA is terminated, the original BWR/2-5 designs meet the 10 CFR 50, Appendix A SFC for ECCS, without the use of any non-ESF system (i.e., without reliance on the RCIC system).

1.1 CRDA Safety Analyses

There are two safety analyses with respect to the CRDA. The first is a core reactivity excursion analysis to determine the energy release resulting from a worst case dropped control rod, which forms a basis for the amount of fuel rods estimated to be damaged. The most recent NRC (generically) accepted licensing topical report that describes this accident scenario is provided in NEDE-33091-A (Reference 1), and provides the following CRDA scenario.

"(a) Reactor is at a control rod pattern corresponding to maximum incremental rod worth.

(b) Rod pattern control systems (Rod Worth Minimizer, Rod Sequence Control System or Rod Pattern Controller) or operators are functioning within the constraints of the Banked Position Withdrawal Sequence (BPWS). The control rod that results in the maximum incremental reactivity worth addition at any time in core life under any operating condition while employing the BPWS becomes decoupled from the control rod drive.

(c) Operator selects and withdraws the drive of the decoupled rod along with the other required control rods assigned to the Banked-position group such that the proper core geometry for the maximum incremental rod worth exists.

(d) Decoupled control rod sticks in the fully inserted position.

(e) Control rod becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 feet per second).

(f) Reactor goes on a positive period and initial power burst is terminated by the Doppler reactivity feedback.

(g) APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).

(h) Scram terminates accident."

The core cooling function is not part of the core reactivity excursion analysis, and thus, it is not modeled in that analysis.

The second analysis is a radiological evaluation; based on assumptions such as the number of fuel rods damaged, plant isolation response, and release paths and rates.

After the scram, the CRDA is considered terminated. It is and, in the past, has been assumed that RPV water level remains above the top of the core. The radioactive material sources released to the reactor coolant in the radiological evaluation are not based on any additional fuel damage due to a loss of coolant. Thus, maintaining water level is not modeled in the radiological analysis.

Therefore, regardless of which system maintains the RPV water above the top of the core, both the core reactivity excursion and radiological analyses are not affected, and no specific coolant system is actually modeled in relation to mitigating the consequences of a CRDA.

2. Review of NEDO-10527

Because

(a) only safety-related equipment may be credited or assumed to function in evaluating any DBA and the CRDA is classified as a DBA,

(b) no BWR RCIC system is needed to perform any safety-related function,

(c) no operating BWR/3-5 RCIC system is classified as an ECCS-ESF system, and

(d) maintaining water level is a post-accident normal shutdown function, and is not a mitigation function,

NEDO-10527 (Reference 2), as a licensing topical report, should not be interpreted as crediting RCIC for mitigating the consequences of a CRDA.

3. Post-TMI Accident

One of the post-TMI accident modifications provides an additional level of ECCS redundancy. This modification initiates ADS (without the need for a high drywell pressure signal) on low low low water level (Level 1) and low pressure ECCS operating, after an ADS initiation timer, as described in FSAR Subsection 7.3.1.1.1.2 and shown in TS Table 3.3.5.1-1. This change ensures that RPV depressurization, LPCI and LPCS are available for all normal or abnormal event scenarios (AOOs, accidents, Appendix R fire, safe shutdown earthquake, etc.) that result in a RPV water level decrease. With this change, the Columbia post-CRDA initiation core cooling capability sequence is as shown in Figure 2. This figure demonstrates that Columbia can maintain post-CRDA core cooling, assuming at least one additional failure beyond the SFC, without the use of the RCIC system. Therefore, (although the RCIC shall continue to be maintained as safety grade) the Columbia RCIC is not needed to perform any safety-related function, including the effects of applying SFC to ECCS.

4. RCIC Functional Description

Regardless of the cause, when the reactor pressure vessel (RPV) is isolated from the main condenser, RPV water will boil-off due to decay heat, which post-shutdown following full power operation is < 6% RTP. The function of the RCIC has always been to maintain RPV water level, when the RPV pressure is above the Residual Heat Removal (RHR) system operating pressure and the RPV is isolated from the main condenser. The limiting design basis event, with respect to the RCIC, is a total loss of feedwater flow.

To meet its functional requirement, RCIC automatically initiates on Level 2 alone, as described in FSAR Subsections 5.4.6.2.1.1 & 7.4.1.1.2 and shown in Technical Specifications (TS) Table 3.3.5.2-1. Therefore, the RCIC initiates any time the RPV is pressurized and Level 2 is reached, regardless of what causes (e.g., operator error, equipment malfunction, AOO or accident) the RPV water level to decrease to Level 2.

The RCIC may also be manually operated to maintain water level.

Although the RCIC does not perform any safety-related function, all the BWR/5s classified (upgraded) their RCICs as safety-related. However, no BWR/5 RCIC has ever been classified as an ECCS or ESF.

Figure 2 shows the current safety-related and nonsafety-related Columbia post-CRDA core cooling paths, without the use of RCIC. The figure demonstrates that Columbia can perform all core cooling functions without the RCIC system, while still meeting SFC. Therefore, although the RCIC is currently maintained as safety-related, it is not needed or specifically required by regulation to perform any post-accident safety function.

5. Operational Considerations

Addressing the RCIC system with respect to a CRDA may have originally resulted from operational considerations, and the fact that most plants had a Main Steamline Isolation Valve (MSIV) closure from a main steamline (MSL) high radiation trip, which Columbia does not have. It was assumed that the radiation from a CRDA would result in a MSIV closure, and thus, the turbine driven feedwater pumps would not be available to maintain RPV water level. Without feedwater flow, from an operational standpoint, the RCIC system is the preferred system for maintaining reactor water level, because the suction source is the relatively clean condensate storage tank and the RCIC's lower flow rate is easier to control than the much higher flow from an ECCS loop, which pulls less clean water from the suppression pool.

Note: Because Columbia would not isolate on high MSL radiation, the Feedwater-Condensate systems would maintain RPV water level after a CRDA has terminated, and thus, neither the RCIC or HPCS would automatically initiate, as shown in Figure 2.

6. Columbia License Amendment Request

EN submitted the Reference 3 license amendment request (LAR). This LAR provides (a) the proposed FSAR changes such that the FSAR would not explicitly credit the RCIC system for mitigating the consequences of a CRDA, (b) an explanation of how post-CRDA cooling would be provided without using the RCIC and still comply with SFC, and (c) a no significant hazards consideration analysis.

GE has never included RCIC system response in its analysis of the CRDA, and thus, GE agrees that RCIC is not analyzed or credited in the CRDA analysis. Therefore, EN's technical justification for the subject LAR is correct.

7. Conclusion

The proposed FSAR changes (i.e., deleting statements that RCIC mitigates the consequences of a CRDA) do not affect any TS requirement, any safety-related SSC functional requirement, RCIC reliability, RCIC safety classification or any safety analysis result, and thus, the proposed FSAR changes cannot affect plant safety. Therefore, this evaluation concludes that there is no technical, operational, regulation or safety reason that should prevent the proposed FSAR changes from being approved and implemented.

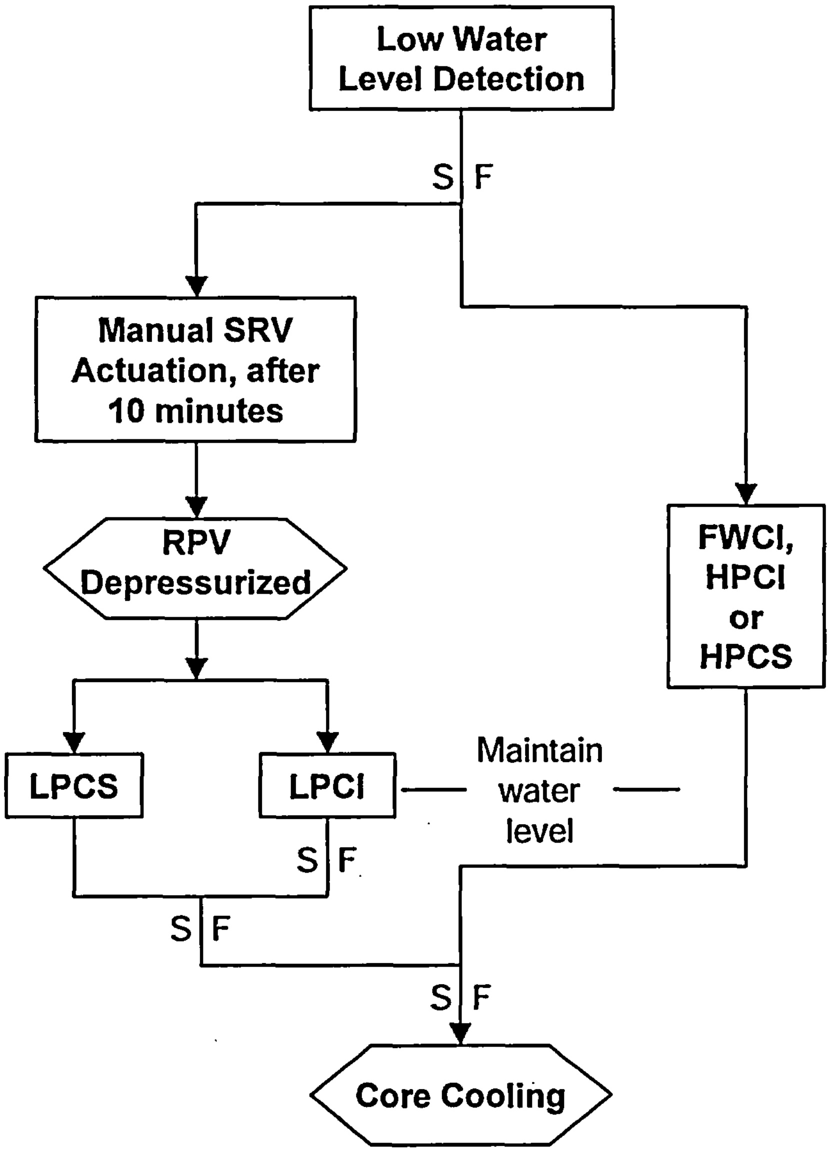
8. References

1. GE Nuclear Energy, BWR Owners' Group Licensing Topical Report, "Improved BPWS Control Rod Insertion Process," NEDO-33091-A (NRC accepted), Revision 2, July 2004.

2. General Electric Co., Licensing Topical Report, "Rod Drop Accident Analysis For Large Boiling Water Reactors," NEDO-10527, March 1972.

3. D. K. Atkinson (Energy Northeast) letter to U.S. Nuclear Regulatory Commission, "Columbia Generating Station, Docket No. 50-397 License Amendment Request: Control Rod Drop Accident Analysis," No. GO2-04-177, October 12, 2004.

4. GE Nuclear Energy, BWR Owners' Group Report, "Original Safe Shutdown Paths For The BWR," GE-NE-T43-00002-00-01-R01, Revision 1, August 1999.

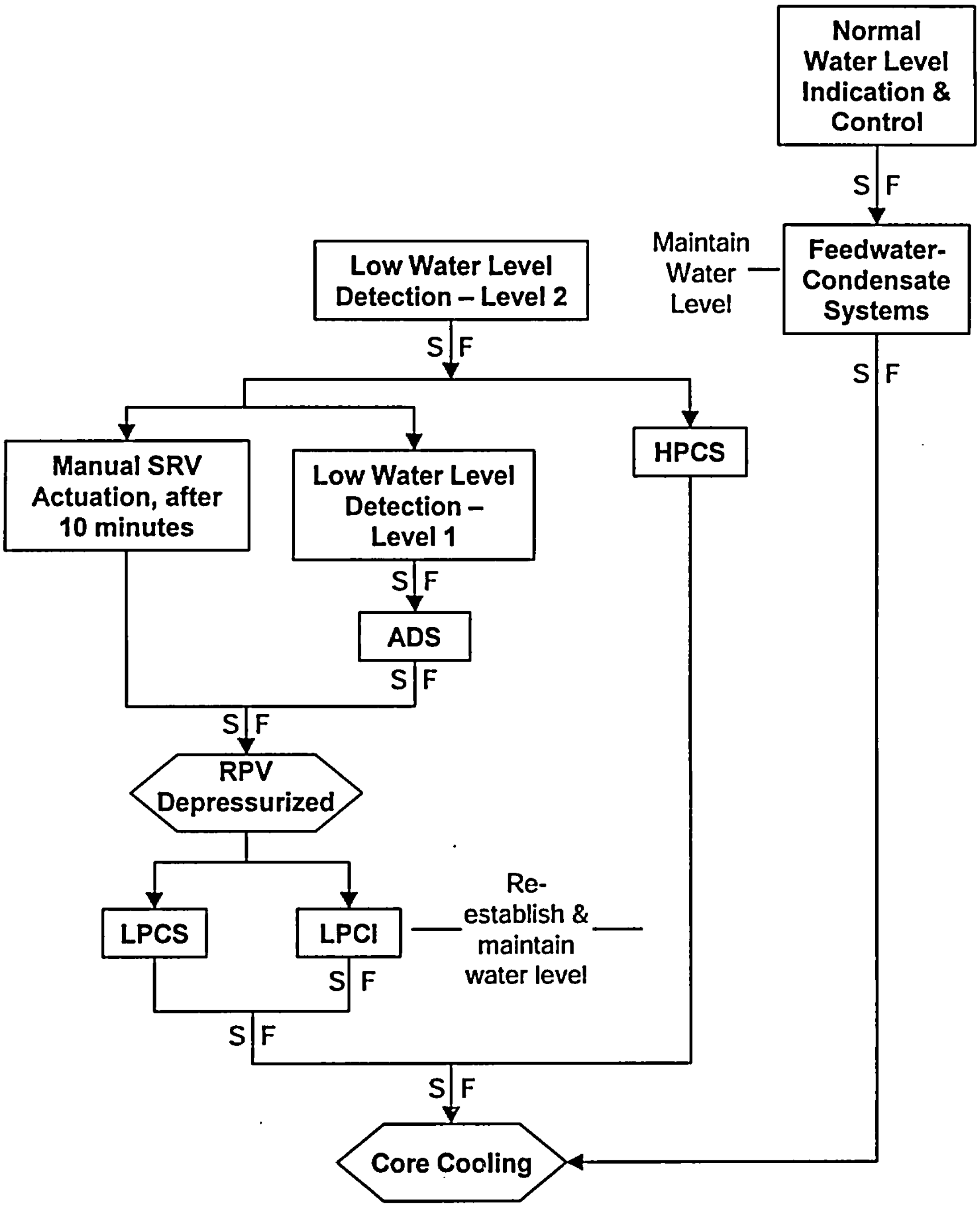


SF – Meets Single Failure Criterion

(Containment isolation is assumed.)

FWCI - Feedwater Coolant Injection
HFCI - High Pressure Coolant Injection

Figure 1. **Pre-TMI Accident BWR/3-5 & Current IC BWR/2-3 Post-CRDA Cooling Paths** (Ref. 4)



SF – Meets Single Failure Criterion

Figure 2. **Post-TMI Accident, No RCIC Post-CRDA BWR/5 Cooling Paths** (Ref. 4)