SURVEILLANCE REQUIREMENTS (Continued)

2. At least once per 18 months.

Boron Injection Throttle Valves		Safety Injection Throttle Valves	
	Valve Number		Valve Number
1.	1-SI-141 L1	1.	1-SI-121 N
2.	1-SI-141 L2	2.	1-SI-121 S
3.	1-SI-141 L3		
4.	1-SI-141 L4		

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump*	Safety Injection System Single Pump**
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow ≥ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow ≥ 300 gpm
Loop 3 Boron Injection Flow 117.5 gpm	**Combined Loop 1,2,3 and 4 Cold Leg Flow (single pump) less than or equal to 640 gpm. Total SIS (single pump) flow, including miniflow, shall not exceed 700 gpm.
Loop 4 Boron Injection Flow 117.5 gpm	

*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow in each loop. Under these conditions there is zero miniflow and 80 gpm plus or minus 5 gpm simulated RCP seal injection line flow.

The actual flow in each BI line may deviate from the nominal so long as:

a) the difference between the highest and lowest flow is 25 gpm or less.

b) the total flow to the four branch lines does not exceed 470 gpm.

c) the minimum flow (total flow) through the three most conservative (lowest flow) branch lines must not be less than 300 gpm.

d) The charging pump discharge resistance ($2.31xPd/Qd^2$) must not be less than 4.73E-3 ft/gpm^2 and must not be greater than 9.27E-3 ft/gpm^2. (Pd is the pump discharge pressure at runout; Qd is the total pump flow rate.)

ATTACHMENT 2 TO AEP:NRC:1067D

WESTINGHOUSE ELECTRIC CORP. SAFETY EVALUATION

FOR REVISED CHARGING PUMP TECHNICAL SPECIFICATION PARAMETERS

SAFETY EVALUATION FOR
REVISED CHARGING PUMP TECHNICAL SPECIFICATION PARAMETERS

Introduction

Westinghouse has performed a safety evaluation for operation of the Donald C. Cook Nuclear Plant Unit 1 with revised charging pump Technical Specification parameters. This evaluation also incorporates closure of either the HHSI or RHR system cross-tie valves. The impact of these operational changes on the safety analyses and evaluations is to reduce the minimum flow rate for Safety Injection (SI) that can be assumed. Therefore, only those safety analyses and evaluations which assume minimum SI flow are addressed. The other safety analyses and evaluations are unaffected by these changes. The safety analyses and evaluations which assume minimum SI flow are listed below.

1. Large break LOCA
2. Small break LOCA
3. LOCA hydraulic forces analysis
4. Post-LOCA long term core cooling subcriticality calculation
5. Steamline Break - Core Response
6. Steamline break mass/energy releases
7. Main steamline break containment integrity
8. LOCA containment integrity
9. Hot leg switchover to prevent boron precipitation

Maximum SI flow is conservatively assumed in the Steam Generator Tube Rupture analysis so there is no impact on this accident analysis.

Technical Specification 4.5.2 describes allowable operation of the Emergency Core Cooling System (ECCS) which includes the SI system. The charging pump, which is part of the SI system, Technical Specification parameters assumed in this evaluation are as follows:

Pump runout flow must not exceed 550 GPM.

The minimum flow rate through any three branch lines must not be less than 300 GPM.

The maximum difference in flow between any two SI branch lines must not exceed 25 GPM.

The simulated seal injection flow rate during SI flow balancing should be between 75 and 85 GPM.

The pump discharge resistance must be within,

$4.73\ E\text{-}3 < K_D < 9.27\ E\text{-}3$

where $K_D = 2.31\ P_D/Q^{**}2$ (ft/GPM**2)

P_D = Pump discharge pressure

Q = Pump flow rate (GPM)

The pump developed head must not be less than the 10% degraded curve used in the ECCS analyses.

The evaluations listed below are based on the analyses that support the Reduced Temperature and Pressure Program which are documented in Section 3 of WCAP-11902 (Reference 1).

Large Break LOCA Evaluation

The large break LOCA analyses for the Reduced Temperature and Pressure program (Section 3.1.1 of Reference 1) were performed using the 1981 Evaluation Model with BASH. A review of the charging pump flow data used in the analyses verified that the calculated flow rates for the new charging pump flow Technical Specifications are bounded by those used for the large break LOCA analyses. Therefore, the large break LOCA analyses performed for the Reduced Temperature and Pressure program support the charging pump parameters listed previously.

The large break LOCA analyses assumed a core power level of 3413 MWt and that both the HHSI and the RHR cross-ties were open. One run was performed at the limiting conditions which supports operation at a power level of 3250 MWt with the RHR cross-tie closed.

Therefore, the large break LOCA analyses support operation of Donald C. Cook Nuclear Plant Unit 1 at a power level of 3413 MWt with both cross-ties open and a power level of 3250 MWt if the RHR cross-tie is closed and the HHSI cross-tie is open. No runs were made with both cross-ties closed or with the RHR cross-tie open and the HHSI cross-tie closed.

An evaluation was performed for the case with the RHR cross-ties open, the HHSI cross-ties closed, and the new charging pump flow rates. Large break LOCA safety injection flow calculations were not performed for this case. However, the SI flow rates with the RHR cross-tie open and the HHSI cross-tie closed are known to be greater than those with the RHR cross-tie closed and the HHSI cross-tie open for the pressures of interest in the large break LOCA analysis. Therefore, it is concluded that the results of a large break analysis at 3250 MWt with the RHR cross-tie open and the HHSI cross-tie closed would be lower than the analyzed case with the RHR cross-tie closed and would not exceed the limits of 10CFR50.46 Appendix K.

Operation with both the RHR and HHSI cross-ties closed has not been supported by the large break LOCA analysis.

Small Break LOCA Evaluation

Small break LOCA analyses were recently performed for Unit 1 as part of the Reduced Temperature and Pressure program (Section 3.1.2 of Reference 1). These analyses support operation at a core power level of 3588 MWt with either the RHR or the HHSI cross-ties closed, but not simultaneously, so long as the charging flow imbalance is less than or equal to 10 gpm, per the current Technical Specification 4.5.2. The small break LOCA analyses also support operation at a core power level of 3588 MWt with the revised charging pump Technical Specification, however the HHSI cross-ties must be open. The analyses assumed a total core peaking factor of 2.32 and the safety injection to core power ratio was approximately 8.9E-03 lbm/s-MWt

The combination of the HHSI cross-ties closed with the revised charging pump Technical Specification results in safety injection flows lower than those assumed for the small break analyses. Therefore, the new charging system parameters can not be supported at a power level of 3588 MWt. In order to implement the new charging system parameters, a power level lower than 3588 MWt is required. A safety evaluation was performed for the new flow rates that supports operation at a power level of 3250 MWt with either the RHR or HHSI cross-ties closed.

A significant parameter in small break LOCA analyses is the safety injection flow rate to power ratio. This ratio reflects the ability of the ECCS to remove decay heat during a small break LOCA. Consequently, a change in this ratio will affect the depth and duration of core uncovery and as a result, the peak clad temperature (PCT). Lowering this ratio is equivalent to a reduction in the SI flow rate at a fixed power level and will result in an increase in the PCT. At a power level of 3250 MWt, the safety injection flow to power ratio, with the new flows, is slightly higher than the analyzed condition at the pressures of interest in the small break transient. Therefore, the PCT for the case assuming the new charging flow rates and a power level of 3250 MWt is expected to be lower than the same break size analyzed at 3588 MWt with the old charging pump flow rates.

In addition to the lower SI flow to power ratio, the margin between the analyzed total core peaking factor of 2.32 and the Technical Specification limit of 2.15 results in substantial PCT margin.

Therefore, the new charging system parameters are acceptable with the HHSI cross-ties closed for small break LOCA up to operation at a power level of 3250 MWt.

LOCA Hydraulic Forces Analysis

The peak loads generated on the reactor vessel and internals as a result of a postulated LOCA occur well before the initiation of safety injection. Therefore, the new charging flow system parameters and the position of the cross-ties will have no effect on the LOCA hydraulic forcing functions used in the structural analysis presented in FSAR Section 14.3.3.

Post-LOCA Long Term Core Cooling Subcriticality Calculation

The post-LOCA boron subcriticality calculation is dependent on the masses and boron concentrations of the RCS, the RWST, the accumulators, and miscellaneous other piping and equipment. The calculation is not dependent on the safety injection flow rate. Therefore, there is no effect on this calculation due to the charging flow parameter changes or the cross-tie closures.

Steamline Break - Core Response (presented in FSAR Section 14C.3.11)

A review of the charging pump flow data used in the analysis for the Reduced Temperature and Pressure program verified that the calculated safety injection flow rates for the new charging pump flow Technical Specifications are bounded by those used for the steamline break core response analysis. Therefore, the steamline break core response analysis performed for the Reduced Temperature and Pressure program supports the degraded safety injection flow.

The analysis assumed that safety injection flow is provided by only the charging system. No credit was taken for safety injection flow provided by the HHSI and/or RHR systems. Thus, cross-tie closure for either of the HHSI or RHR systems does not affect the steamline break core response analysis.

<u>Steamline Break Mass/Energy Releases (Inside Containment)</u>

For the Steamline Break Mass & Energy Releases Inside Containment, sensitivity analyses have shown that small changes in the safety injection boron concentration, safety injection flow, and/or actuation delays have little effect on the mass and energy releases. The degradation in the safety injection flow delivered by the charging pumps identified above is a small perturbation compared to changes (e.g., no SI flow, no boron, etc.) assumed in previously documented analyses. Based on these analyses, it is concluded that the degraded SI flow capability would not adversely affect the Steamline Break Mass and Energy Releases for breaks inside containment.

The analyses assumed that safety injection flow is provided by only the charging system. No credit was taken for safety injection flow provided by the HHSI and/or RHR systems. Thus, the issue of cross-tie closure for either of the HHSI or RHR systems does not affect the steamline break mass/energy release analyses.

<u>Steamline Break Mass/Energy Releases (Outside Containment)</u>

Sensitivities performed for the Steamline Break Superheated Mass & Energy Releases Outside Containment, contained in Reference 2, show that the results are not sensitive to large changes in safety injection flow. The degradation in safety injection flow delivered by the charging pumps at Cook Unit 1 is a small perturbation compared to the large change in total safety injection flow assumed in the Reference 2 sensitivity. Therefore, it is concluded that the impact on the Cook Superheated Mass & Energy Releases Outside Containment is insignificant.

The analyses assumed that safety injection flow is provided by only the charging system. No credit was taken for safety injection flow provided by the

HHSI and/or RHR systems. Thus, the issue of cross-tie closure for either of the HHSI or RHR systems does not affect the steamline break mass/energy release analyses.

Main Steamline Break Containment Integrity

The main steamline break containment integrity evaluation uses the mass/energy releases calculated in Section 3.3.4.1 of Reference 1. As stated in the discussion of Steamline Break Mass/Energy Releases (Inside Containment) the mass and energy releases are not adversely affected by the revised charging flow Technical Specification. Therefore, the main steamline break containment integrity analysis is not adversely affected by the revised charging flow Technical Specification.

Also, the steamline break mass/energy release calculations assumed that safety injection flow is provided by only the charging system. No credit was taken for safety injection flow provided by the HHSI and/or RHR systems. Thus, cross-tie closure for either of the HHSI or RHR systems does not affect the steamline break mass/energy release analyses. As such, the main steamline break containment integrity evaluation is also not affected by cross-tie closure.

LOCA Containment Integrity

The LOCA containment integrity analysis documented in WCAP-11908 (Reference 3) uses long term LOCA mass and energy releases that were generated assuming the RHR crosstie valve was closed. Closure of the RHR crosstie results in a more conservative calculation than having the SI crosstie closed. The charging pump flow rates associated with the revised Technical Specification are bounded by the flow rates assumed in the analysis. Thus the analysis performed bounds either the RHR or SI crosstie valves closed, and is not affected by the charging flow imbalance.

The LOCA subcompartment containment integrity analysis is not affected by either the crosstie valve closure or the charging flow imbalance, because the safety injection system is not actuated during the time period of the analysis.

Hot Leg Switchover to Prevent Boron Precipitation

Post-LOCA hot leg switchover time is dependent on the power level and the volumes and boron concentrations of the RCS, the RWST, and the accumulator. Changes to the safety injection flow rates will not influence these parameters so there is no impact on the calculation of switchover time. The calculation also verifies that at the time of switchover the hot leg safety injection flow rate is conservatively greater than the core boiloff. Since the charging pumps provide the hot leg injection flows at high pressures, Westinghouse verified that the flow rate with the new charging flow parameters is adequate. The new charging system parameters are acceptable for the hot leg switchover calculation.

Conclusions

The potential safety impact of operation of the Donald C. Cook Nuclear Plant Unit 1 with the revised charging flow Technical Specification parameters, and either the HHSI or RHR system cross-tie valves closed, but not simultaneously, on the safety analyses and evaluations performed for the Reduced Temperature and Pressure program has been evaluated. It is determined that operation of the Donald C. Cook Nuclear Plant Unit 1 with the revised conditions described above does not affect the conclusions of the safety analyses and evaluations as long as core power is $\leq$ 3250 MWt.

Based on the results of the evaluation described above, it has been determined that:

- the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased;

- the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created; and

- the margin of safety as defined in the basis for any technical specification is not reduced.

Thus, operation of the Donald C. Cook Nuclear Plant Unit 1 with the revised charging flow Technical Specification parameters, and <u>either</u> the HHSI or RHR system cross-tie valves closed does not affect the conclusions of the safety analyses and evaluations as long as core power is $\leq$ 3250 MWt and, therefore, does not produce an unreviewed safety question.

<u>References</u>

1. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988.

2. WCAP-10961, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," October 1985.

3. WCAP-11908, "Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 and 2," July 1988.

SECL NO. 89-790
Customer Reference No(s).

Westinghouse Reference No(s).

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST

1.) NUCLEAR PLANT(S): Donald C. Cook Nuclear Plant Unit 1

2.) SUBJECT (TITLE): Revised Charging Pump Technical Specification

3.) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59(b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10CFR50.59(a)(1)

3.1) Yes___ No_X_ A change to the plant as described in the FSAR?
3.2) Yes___ No_X_ A change to procedures as described in the FSAR?
3.3) Yes___ No_X_ A test or experiment not described in the FSAR?
3.4) Yes_X_ No___ A change to the plant technical specifications? (See Note on Page 2.)

4.) CHECK LIST - PART B - 10CFR50.59(a)(2) (Justification for Part B answers must be included on page 2.)

4.1) Yes___ No_X_ Will the probability of an accident previously evaluated in the FSAR be increased?
4.2) Yes___ No_X_ Will the consequences of an accident previously evaluated in the FSAR be increased?
4.3) Yes___ No_X_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
4.4) Yes___ No_X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.5) Yes___ No_X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.6) Yes___ No_X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
4.7) Yes___ No_X_ Will the margin of safety as described in the bases to any technical specification be reduced?

NOTES:

If the answer to any of the above questions is unknown, indicate under 5.) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 10CFR50.90.

5.) REMARKS:

The following summarizes the justification based upon the written safety evaluation (1) for answers given in Part A (3.4) and Part B of this SECL.

See preceding safety evaluation.

(1) Reference to document(s) containing written safety evaluation:

FOR FSAR UPDATE

Section:__________ Pages:________ Tables:__________ Figures:_________

Reason for / Description of Change:

No FSAR changes

SAFETY EVALUATION APPROVAL LADDER:

Prepared by (Nuclear Safety): D. H. Behnke DH Behnke Date: 6/1/89

Coordinated with Engineer: Signature on File Date: ________

Coordinating Group Manager: Signature on File Date: ________

Nuclear Safety Group Manager: E. M. Burns E. M. Burns Date: 6/1/89