APPENDIX 15A

SUPPLEMENTAL INFORMATION FOR CHAPTER 15 IN

RESPONSE TO WATERFORD 3 (NUREG-0787)

OPEN ITEMS 8 THROUGH 11

This Appendix is provided to document the information presented to the ACRS Subcommittee and the NRC Staff on August 5, 1981.

15A.1 SER OPEN ITEM NO. 8

Statement - "The current CESEC model does not properly account for steam formation in the reactor vessel. Therefore, for all events in which (a) the pressurizer is calculated to drain into the hot leg, or (b) the system pressure drops to the saturation pressure of the hottest fluid in the system during normal operation, we require the applicant to reanalyze these events with an acceptable model or otherwise justify the acceptability of Waterford 3 Chapter 15 analysis conclusions performed with CESEC."

Response

For all Chapter 15 events for which the pressurizer fluid is calculated to drain into the hot leg, or the system pressure drops below the saturation pressure of the hottest fluid in the system, the hottest fluid will be located in the relatively stagnant upper head region of the reactor vessel*. The CESEC-I code, used in the FSAR Chapter 15 analyses did not explicitly model the steam formation in the reactor vessel upper head region. The latest version of CESEC, namely CESEC-III, explicitly models steam void formation and collapse in the upper head region of the reactor vessel. Heat transfer from metal structures to the reactor coolant system (RCS) fluid is modeled in addition to flashing of the reactor coolant into steam during depressurization of the RCS. Following the reactor coolant pump (RCP) coastdown due to loss of offsite power or manual shutoff following SIAS, thermalhydraulic decoupling of the upper head region is characterized in CESEC-III by progressively decreasing flow to the upper head from the upper plenum region.

The Steam Generator Tube Rupture (SGTR) with Loss of Offsite Power (LOP) at the time of reactor trip is the most limiting event with respect to the duration of voids in the RCS. The Steam Line Break (SLB) inside containment with LOP and one HPSI pump failure is the most limiting event with respect to maximum steam volume in the RCS. Both of these events were reanalyzed using the CESEC-III code. The results are shown in Tables 15A.1-2 and 15A.1-3 and Figures 15A.1-1 through 15A.1-4 for the comparative analysis for the SGTR and SLB events.

The conclusions from the comparative analyses for the SGTR and SLB events bound the other depressurization events for which void formation is less limiting and/or non-existent. This is due to slower cooldown rates and higher minimum RCS pressures for the other depressurization event. Therefore, the qualification of CESEC-I against CESEC-III for the SGTR and SLB events provides the necessary justification for the acceptability of the Waterford 3 Chapter 15 analyses conclusions for depressurization events as presented in the FSAR.

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*Significant differences impacting the analyses between the two CESEC versions are summarized in Table 15A.1-1.

Steam Generator Tube Rupture Event

The reanalysis of the SGTR event indicated the following:

The modelling of the stagnant upper head region with metal structure heat transfer results in the formation of voids in this region. The void volume in the upper head region peaks at about 535 cu. ft. during the transient and gradually decreases under the combined action of the HPSI flow and the controlled cooldown at the steam generators. The duration of the voids depends on the rate of cooldown of the primary side and the HPSI flow rate.

The amount of voids predicted is not large enough to expand the steam bubble beyond the upper head region and to the elevation of the hot legs. Therefore, natural circulation cooldown of the RCS is not impaired.

The prediction of the upper head voids in the reanalysis does not alter the conclusions of the previous analysis. The results of the reanalysis not only show insignificant impact on the offsite doses, but also demonstrate that the plant can be maintained in a stable condition by the collapse of the upper head voids in a timely manner through manual control of the cooldown rate. Table 15A.1-2 lists the results of the original FSAR analysis using CESEC I and the results of the reanalysis using CESEC-III.

Subsequent to collapse of the upper head voids, the plant is maintained in a stable condition, and the operator can bring the plant to the shutdown cooling entry conditions, by cooling down the RCS at a prescribed cooldown rate using the intact steam generator and the condenser.

Steam Line Break Event

The reanalysis of the limiting* steam line break (SLB) event indicated the following:

→(DRN 00-592)

The large SLB event with concurrent loss of offsite power (LOP) results in the maximum void formation in the reactor vessel upper head. The event is initiated from 102 percent power by a double ended guillotine break of the main steam line inside containment. Reactor trip and main steam isolation signals are generated on low steam generator pressure at 2.1 seconds into the transient. Initial void formation occurs in the upper head at seven seconds into the transient as a result of the rapid cooldown of the reactor coolant system (RCS), and is further enhanced by the coastdown of the reactor coolant pumps due to the LOP. Emergency feedwater (EFW) is initiated on low steam generator water level in the intact steam generator due to level shrink that occurs after the closure of the main steam isolation valves.

←(DRN 00-592)

*The limiting SLB event with potential for post-trip degradation in fuel performance is a full power SLB inside containment, with loss of offsite power coincident with the SLB (see Appendix 15A.4), failure of one HPSI pump, a stuck CEA, and an actuation of emergency feedwater.

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Full flow from the turbine driven EFW pump plus one motor driven EFW pump is conservatively assumed to begin filling the intact steam generator coincident with closure of the main feedwater isolation valves at 22.1 seconds into the transient. (The Emergency Feedwater actuation system logic prevents feeding the ruptured steam generator.) The pressurizer empties at 30 seconds. At approximately 110 seconds the affected steam generator dries out. The void in the upper head reaches a maximum of 930 ft^3 at approximately 135 seconds into the transient. At about 150 seconds into the transient the pressurizer liquid level is reestablished. The liquid level in the intact steam generator reaches the point at which the automatic EFW system shuts off flow at approximately 370 seconds into the transient. Thereafter the void volume in the upper head continues to reduce due to the primary coolant swell from decay and residual heat in the RCS components and from safety injection and charging. At a maximum of 30 minutes after the initiation of the transient the operator initiates plant cooldown using the intact steam generator. At no time during the event does the upper head void region enter the hot leg. Therefore, natural circulation cooldown of the RCS is not inhibited.

←(DRN 00-592)

Table 15A.1-3 lists the results of the original FSAR analysis using CESEC-I and the results of the reanalysis using CESEC-III. Figure 15A.1-3 compares the RCS pressures calculated by CESEC-I and CESEC-III for this event. Figure 15A.1-4 presents the reactor vessel liquid level as a function of time. Reanalysis of this event using a model which properly accounts for void formation in the upper head region during a steam line break event does not alter any conclusions reached in the FSAR.

15A.2 SER OPEN ITEM NO. 9

Statement - "We require the applicant to clarify the differences in methodology utilized for analyzing feedwater line breaks between that for Waterford 3 and that documented in CESSAR System 80."

Response:

The comparison of methodology utilized for analyzing feedwater line break (FWLB) for Waterford 3 and that documented in Appendix 15B of CESSAR FSAR follows. The major evaluation areas unique to FWLB which the Waterford 3 and CESSAR methodologies address include the selection/treatment of:

a) affected steam generator heat transfer

b) fluid conditions at the break

c) affected steam generator low level trip

d) break discharge

e) reverse steam flow in main steam lines

f) plant initial conditions

g) break size

Both methodologies utilized simplified models rather than justifying detailed best estimate model to determine an upper limit for the reactor coolant system (RCS) pressurization transient. The comparison shows that, although differing from the CESSAR methodology, the Waterford 3 method is valid and conservative.

<u>Affected Steam Generator Heat Transfer</u>

RCS pressurization is largely a function of the rate at which the affected steam generator heat transfer decreases as its inventory is depleted. The overall heat transfer coefficient will decrease as the steam generator tubes are exposed to increasing void fractions which force the tubes from the normal nucleate boiling heat transfer regime into transition boiling and eventually into liquid deficient heat transfer. Transition boiling is anticipated when the local void fraction exceeds 0.9. A gradual heat transfer reduction is expected, starting when the affected generator liquid inventory decreases to approximately 70,000 lbm forcing portions of the tubes into transition boiling, and continuing as transition boiling and then liquid deficient heat transfer propagate throughout the tubes. Figure 15A.2-1 shows the expected behavior of the overall heat transfer coefficient, along with the behavior assumed for Waterford 3 and CESSAR evaluations.

Waterford 3 conservatively ignored the transition boiling regime, thereby delaying the heat transfer decrease until fluid conditions are predicted which correspond to liquid deficient heat transfer. CESSAR took a more simplistic approach assuming heat transfer degradation was instantaneous upon steam generator dryout.

Appendix 15B of CESSAR FSAR documents the sensitivity of RCS pressurization to steam generator heat transfer behavior. The study verified that RCS pressurization is maximized by under-estimating the affected steam generator liquid mass corresponding to the initiation of heat transfer degradation (i.e., over-estimating the rate of heat transfer decrease). Both CESSAR and Waterford 3 conservatively assumed heat transfer characteristics which were biased to under-estimate the liquid inventory at which degradation was initiated.

<u>Fluid Conditions at the Break</u>

The enthalpy of the fluid discharged from the feedline break partially determines the heat removal capability of the affected steam generator. Minimizing the discharge enthalpy reduces the heat removal and, thereby maximizes the RCS pressurization. Models for both Waterford 3 and CESSAR were biased to conservatively under-estimate the discharge enthalpy. Figure 15A.2-2 shows the behavior of the discharge enthalpy during steam generator dryout predicted by Waterford 3 and CESSAR methods, along with the expected behavior.

The expected enthalpy response for Waterford 3 can be understood by referring to the simplified drawing of the steam generator provided on Figure 15A.2-3. Fluid discharge from a feedline break is drawn from the downcomer section through the feedwater distribution ring. Saturated liquid in the downcomer normally covers the feedwater ring. During FWLB the downcomer liquid will be depleted lowering the water level and uncovering the ring. A two phase fluid (high enthalpy) will be discharged thereafter. Feedwater ring uncovery and the associated high enthalpy will occur before the steam generator liquid inventory decreased below 100,000 lbm.

Waterford 3 FWLB methods under-estimate the discharge enthalpy by assigning an enthalpy corresponding to the average downcomer fluid conditions and assuming 1) all liquid within the steam generator is located in the downcomer, and 2) the downcomer fluid level remains at the feedwater ring as the liquid is depleted.

For CESSAR FWLB evaluation a more simplistic model was used. Due to the relatively low elevation for the System 80 feedwater nozzles (Figure 15A.2-3) the model assumes that saturated liquid is discharged from the break until no liquid remains in the steam generator.

Although different, both Waterford 3 and CESSAR conservatively under-estimate the discharge enthalpy as indicated on Figure 15A.2-2.

Affected Steam Generator Low Level Trip

Reactor trip on a steam generator low water level can mitigate the RCS heatup and pressurization during FWLB. Waterford 3 and CESSAR methods for calculating affected steam generator low level trip were biased to conservatively delay the trip.

Steam generator level is inferred from the measured elevation head (ΔP_{level}) associated with the downcomer fluid between two instrument tap locations (see Figure 15A.2-3). When the measured head decreases below a predetermined setpoint a steam generator low level trip signal is generated. As the downcomer level decreases during FWLB low level trip is expected to occur with greater than 70,000 lbm of liquid in the affected steam generator. The Waterford 3 method conservatively over-estimates the measured elevation head to delay low level trip. The elevation head is based on the same assumptions used to predict discharge fluid conditions (i.e., all liquid is located in the downcomer and the downcomer level does not drop below the feedwater ring). This method calculates 9,000 lbm of liquid in the affected generator at low level trip.

The CESSAR method simply assumed affected steam generator low water level does not occur until all liquid is depleted.

Although Waterford 3 differs from CESSAR, both methods are conservative in their prediction of a low water level trip condition in the affected steam generator. Table 15A.2-1 summarizes these results.

Break Discharge Rate

Maximizing the break discharge rate, when combined with underpredicting the discharge fluid enthalpy, reduces the heat removal capability of the affected steam generator and thereby aggravates RCS heatup and pressurization. Both Waterford 3 and CESSAR evaluations conservatively estimate the flow rate assuming instantaneous establishment of frictionless critical flow through the break as predicted by the Henry/Fauske correlation.*

Reverse Steam Flow in Main Steam Lines

Reverse steam flow from the intact steam generator, through the main steam lines to the affected generator, and through the break contributes to the heat removal capability of the intact steam generator, and thereby the mitigation of RCS heatup and pressurization. The impact is greatest for large breaks which can establish the largest reverse steam flow rates.

Plant Initial Conditions

Initial conditions (e.g.RCS pressure, steam generator, liquid inventory and core burnup) can be selected to maximize the RCS heatup and pressurization. Both Waterford 3 and CESSAR FWLB evaluations selected the most adverse set of initial conditions within the allowable plant operating space, based on engineering judgement supported by sensitivity studies like those documented in Appendix 15B of CESSAR FSAR.

Break Size

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The most adverse break size was identified for both Waterford 3 and CESSAR based on sensitivity studies (0.2 ft^2 and 1.075 ft^2, respectively) consistent with their respective modeling assumptions previously described. Results of the sensitivity studies are shown on Figure 15A.2-4. The slight difference in sensitivity between CESSAR and Waterford 3 is mainly due to treatment of reverse steam flow. Figure 15A.2-4 shows the approximate change in the CESSAR results when, like the Waterford 3 method, reverse steam flow is not credited. Like Waterford 3, the maximum RCS pressure is then nearly insensitive to break size.

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In summary, the development of the FWLB evaluation method for Waterford 3 incorporated many conservative biases with respect to the prediction of maximum RCS pressure. The CESSAR method development simplified portions of the Waterford 3 method where conservatism could be maintained (e.g., treatment of the affected steam generator heat transfer, fluid conditions at the break, and affected steam generator low level trip). The maximum RCS pressure predicted for limiting break size by either evaluation model is a conservative estimate which will not be exceeded by any feedwater line break event.

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*R.E. Henry, H.K. Fauske, "The Two Phase Critical Flow of One-Component Mixtures in Nozzles, Orfices, and Short Tubes," Journal of Heat Transfer, Transactions of the ASME, May, 1971.

CESSAR methods accounted for this phenomenon, and Waterford 3 methods conservatively assumed no reverse steam flow.

Although different than the method documented in Appendix 15B of CESSAR FSAR, the Waterford 3 evaluation method for FWLB is valid and conservative.

15A.3 SER OPEN ITEM NO. 10

Statement- "We require the applicant to provide evaluation of the effect of loss of AC power during the steam line break."

Response:

The major concern for main steam line break (MSLB) events is that the CEA design provides adequate shutdown reactivity worth. Under the assumption of the most reactive CEA stuck in the fully withdrawn position, the CEA worth must be sufficient to preclude unacceptably large amounts of fuel damage due to any post-trip return to power that may be calculated to occur as a result of the severe primary system cooldown. MSLB events initiated at full power with loss of offsite power (LOP) at time zero (coincident with the MSLB) and without loss of offsite power are presented in the FSAR.

Figures 15A.3-1 and 15A.3-2 demonstrate that the consequences of a MSLB event coincident with a LOP bound the consequences of a MSLB event with a LOP occurring at times greater than zero. Figure 15A.3-1 presents the maximum post-trip reactivity versus LOP time. Figure 15A.3-2 presents the minimum post-trip DNBR versus LOP Time"*. The post-trip return to power peaks occurs at about 60 seconds into the transient if there is no LOP prior to this time. Therefore, as seen on Figures 15A.3-1 and 15A.3-2, for any cases with LOP after 60 seconds the same maximum post-trip reactivity and minimum post-trip DNBR are calculated.

The MSLB event initiated at zero power which is presented in the FSAR maximizes the potential offsite doses by maximizing the initial steam generator liquid inventory instead of maximizing the potential for post-trip degradation in fuel performance. The effect of LOP upon dose is to make the condenser unavailable for cooldowns using the intact steam generator. The technical specification on the power dependent CEA insertion limit ensures that there is no approach to DNB for MSLB events initiated at zero power.

Therefore, the full power MSLB with LOP coincident with the MSLB case presented in the FSAR is also bounding for the zero power MSLB event in terms of fuel performance. The zero power MSLB with LOP coincident with the MSLB is bounding in terms of offsite dose.

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*The results presented on Figures 15A.3-1 and 2 were obtained using the CESEC III code (see Section 15A.1). The effect of automatic actuation of emergency feedwater was included in the analyses.

15A.4 SER OPEN ITEM NO.11

Statement - "We require the applicant to provide information which explains why the stuck-open atmospheric dump valve event for Waterford 3 results in fuel damage whereas the steamline break event does not result in exceeding DNBR limit."

Response

The stuck-open atmospheric dump valve (ADV) event in combination with loss of offsite power (LOP) upon turbine-generator trip potentially results in a limited number of fuel pins experiencing DNB. The main steam line break (MSLB) events do not result in a minimum DNBR of less than 1.19. The focus of concern for analysis of degradation in fuel performance is upon two different transient regimes for the stuck-open ADV and MSLB events. The analyses for increased main steam flow events, such as the stuck-open ADV, maximize the potential for pre-reactor-trip degradation in fuel performance. The analyses for MSLB events maximize the potential for post-trip degradation in fuel performance.

The core protection calculators (CPCs) will assure that the minimum DNBR will not be less than 1.19 prior to reactor trip for any overcooling transient, unless a single failure of loss of offsite power (LOP) is assumed. However, when LOP is assumed, the increased main steam flow event can result in a calculated minimum DNBR of less than 1.19, as shown on Figure 15A.4-1. The increased main steam flow causes a reduction in DNBR and the CPC's will initiate a reactor trip signal on the projected minimum DNBR of 1.19. The reactor trip signal generates a turbine trip which is assumed to result in a LOP. The LOP then causes the reactor coolant pumps to begin to coast down. The resultant reactor coolant flow reduction causes the DNBR to decrease below 1.19 before the insertion of the control element assemblies reduces the reactor power.

The above sequence of events and consequences are the same for any increased main steam flow event, whether the initiating event is a stuck-open ADV, inadvertent opening of a turbine bypass valve, or a small steam line break.

A large MSLB will have less potential for degradation in fuel performance prior to reactor trip than the above events since reactor trip will occur earlier in the transient due to low steam generator pressure. The major concern for the MSLB events is that the CEA design provides adequate shutdown reactivity worth. Under the assumption of the most reactive CEA stuck in the fully withdrawn position, the CEA worth must be sufficient to preclude unacceptably large amounts of fuel damage. This would be due to any post-trip return to power that might be calculated to occur as a result of the severe primary system cooldown caused by the MSLB. The analyses for the FSAR demonstrate that the CEA worth for the Waterford 3 NSSS is sufficient to preclude any fuel damage during the post-trip portion of a MSLB event.

Therefore, the increased main steam flow event with LOP is calculated to result in a minimum DNBR less than 1.19 during the pretrip time period, whereas the MSLB events result in post-trip values of minimum DNBR greater than 1.19.

TABLE 15A.1-1

SUMMARY SIGNIFICANT DIFFERENCES
BETWEEN CESEC-III AND CESEC-I

MODEL	CESEC-III	CESEC-I
Thermalhydraulic	26 Nodes, Upper Head Explicitly Modeled	16 Nodes
RCS Flow	Explicitly Modeled	Input Table
RCP's	Four, Explicitly Modeled	Two
Wall Heat	Explicitly Modeled	None
SGTR Option	Critical Flow	Darcy Equation
Mixing in RV	Asymmetric Response Explicitly Modeled	Asymmetric Response Included in Reactivity Calculation for SLB
3D Feedback	Yes	No

TABLE 15A.1-2

COMPARISON OF RESULTS
STEAM GENERATOR TUBE RUPTURE WITH LOP

	CESEC-1	CESEC-III
Primary-Secondary Integrated Leak (LBM) at 1800 Secs.	60,722	66,420
Integrated MSSV Steam Release (LBM) at 1800 Secs.	78,300	84,600
Total Steam Release (LBM)	761,810	803,800
Minimum DNBR	1.28	1.21
Offsite Dose-Thyroid (REM)	66.5*	73.0**

* Limiting SER Value at EAB

** Estimated

TABLE 15A.1-3

COMPARISON OF RESULTS
FULL POWER STEAM LINE BREAK, INSIDE CONTAINMENT,
WITH LOSS OF OFFSITE POWER, HPSI PUMP FAILURE,
AUTOMATIC ACTUATION OF AUXILIARY FEEDWATER

	CESEC-I	CESEC-III
Maximum Post-Trip Reactivity (% $\Delta\rho$)	+0.0002	-0.05
Maximum Post-Trip Core Power (% of 3410 M_W)	10.4	7.10
Core Flow at Time of Max. Power (Fraction)	0.043	0.055
Minimum Post-Trip DNBR	>1.2	>1.2

TABLE 15A.2-1

AFFECTED STEAM GENERATOR LIQUID
INVENTORY AT LOW LEVEL TRIP

Assumption	Affected SG Liquid (LBM)
Best Estimate	70,000
Waterford Method	9,000
CESSAR Method	0