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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
10 CFR 50.46 Report of Changes or Errors in
ECCS Evaluation Models

This report is provided as notification of changes or errors in the BVPS-1 and BVPS-2 Large Break LOCA evaluation models which are reportable within 30 days. Current information for other ECCS evaluation models has been provided to satisfy annual reporting requirements. The following attachments provide information as requested by 10 CFR 50.46:

Attachment 1 Provides a listing of each change or error in an acceptable evaluation model that affects the peak fuel cladding temperature (PCT) calculation for particular transients. It quantifies the effect of changes with respect to potential plant-specific impact on PCT for that transient and provides an "index" into Attachment 2 (Generic Descriptions).

Attachment 2 Provides a generic description (based on information provided by Westinghouse) for each model change or error.

Attachment 3 Provides a list of references which occur in the various descriptions. These documents have already been provided to the NRC by Westinghouse.

The PCT effects, described in Attachment 1, have been applied as penalties to the appropriate PCT calculations. This results in calculated PCTs for the large and small break LOCA transients as follows:

BVPS-1 Large Break LOCA - 2182°F
BVPS-1 Small Break LOCA - 2197°F
BVPS-2 Large Break LOCA - 2166°F
BVPS-2 Small Break LOCA - 2176°F

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The recently discovered error in the BVPS-1 LBLOCA Evaluation Model (Attachment 2, Item XIX) has been evaluated through plant-specific sensitivity studies. These studies show that 10 CFR 50.46 requirements are satisfied and the 2200°F limit is met. No further action is required.

DLC has contracted Westinghouse to analyze the BVPS-2 Large Break LOCA transient using the BASH model. This was done to provide more operating margin than the current BART model, and satisfies the 10 CFR 50.46 re-analysis option. This analysis is contracted to be complete by June 30, 1993.

Any questions pertaining to this subject may be directed to G. L. Beatty at (412) 393-5225.

Sincerely,

J. D. Sieber

J. D. Sieber

cc: Mr. L. Rossbach, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. G. E. Edison, Project Manager
Mr. M. L. Bowling (VEPCO)

Attachment 1

Summary of PCT Effects for BVPS LOCA Transients

Plant	Transient	PCT Effect (°F)	Attachment 2 Description(Page)
BVPS-1	Large Break	10	II (Page 2)
		10	V (Page 10)
		0	VIII (Page 14)
		0	IX (Page 15)
		0	XI (Page 19)
		-25	XVII (Page 25)
		0	XVIII (Page 26)
		82	XIX (Page 27)
BVPS-1	Small Break	0	III.A (Page 4)
		[illegible]7	III.B (Page 4)
		0	III.C (Page 5)
		0	III.D (Page 5)
		(Note 1)	IV (Page 8)
		37	V (Page 10)
		0	VI (Page 12)
		(Note 2)	VII (Page 13)
		81	X (Page 18)
		0	XV (Page 23)
		0	XVI (Page 24)
		6	XX (Page 28)
BVPS-2	Large Break	0	I (Page 1)
		25	V (Page 10)
		0	VIII (Page 14)
		30	IX (Page 15)
		0	XI (Page 19)
		(Note 2)	XIII (Page 21)
		-25	XVII (Page 25)
		0	XVIII (Page 26)
BVPS-2	Small Break	(Note 1)	III.A (Page 4)
		37	III.B (Page 4)
		0	III.C (Page 5)
		0	III.D (Page 5)
		0	III.E (Page 5)
		5	III.F (Page 6)
		(Note 3)	III.G (Page 6)
		(Note 1)	III.H (Page 6)
		37	V (Page 10)
		20	VI (Page 12)
		0	VII (Page 13)
		32	X (Page 18)
		-145	XII (Page 20)
		20	XIII (Page 21)
		17	XIV (Page 22)
		0	XV (Page 23)
		0	XVI (Page 24)

Note 1: Results in a small PCT reduction if corrected.

Note 2: Results in an unquantified PCT reduction if corrected.

Note 3: Under investigation by the NSSS vendor.

I. MODIFICATIONS TO THE WREFLOOD COMPUTER CODE

Background:

A modification was made to delay downcomer overfilling. The delay corresponds to backfilling of the intact cold legs. Data from tests simulating cold leg injection during the post-large break LOCA reflood phase which gave adequate safety injection flow to condense all of the available steam flow show a significant amount of subcooled liquid to be present in the cold leg pipe test section. This situation corresponds to the so-called maximum safety injection scenario of ECCS Evaluation Model analyses.

For maximum safety injection scenarios, the reflooding models in the Westinghouse 1981 ECCS Evaluation Model, the Westinghouse 1981 ECCS Evaluation Model incorporating the BART analysis technology, and the Westinghouse 1981 ECCS Evaluation Model incorporating the BASH analysis technology use WREFLOOD code versions which predict the downcomer to overfill. Flow through the vessel side of the break is computed based upon the available head of water in the downcomer in WREFLOOD using an incompressible flow in an open channel method. A modification to the WREFLOOD computer code was made to consider the cold leg inventory which would be present in conjunction with the enhanced downcomer level in the non-faulted loops.

Change Description:

WREFLOOD code logic was altered to consider the filling of the cold legs together with downcomer overfilling. Under this coding update, when the downcomer level exceeds its maximum value as input to WREFLOOD, liquid flow into the intact cold leg, as well as spillage out the break, is considered. This logic modification stabilizes the overfilling of the vessel downcomer as it approaches it equilibrium level. The appropriate WREFLOOD code versions associated with the 1981 Westinghouse ECCS Evaluation Model and the 1981 Westinghouse ECCS Evaluation Model incorporating the BART and BASH technology have been modified to incorporate the downcomer overfill logic update.

Effect of Change:

This change represents a model enhancement in terms of the consistency of the approach in the WREFLOOD code and the actual response of the downcomer level. In some cases this change could delay the overfilling process, which could result in a peak cladding temperature (PCT) penalty. The magnitude of the possible PCT penalty was assessed by reanalyzing the plant which is maximum safeguards limited (CD=0.6 DECLG case) and which is most sensitive to the changes in the WREFLOOD code. The PCT penalty of 16°F which resulted for this case represents the maximum PCT penalty which could exhibited for any plant due to the WREFLOOD logic change.

II. MODIFICATIONS TO THE BASH ECCS EVALUATION MODEL

Background:

In the BASH ECCS Evaluation Model (reference 3), the BART core model is coupled with equilibrium-NOTRUMP computer code to calculate the dynamic interaction between the core thermohydraulics and system behavior in the reactor coolant system during core reflood. The BASH code reflood model replaces the WREFLOOD calculation to produce a more dynamic flooding transient which reflects the close coupling between core thermohydraulic and loop behavior. Special treatment of the BASH computer code outputs is used to provide the core flooding rate for use in the LOCBART computer code. The LOCBART computer code results from the direct coupling of the BART computer code and the LOCTA computer code to directly calculate the peak cladding temperature.

Change Description:

Modifications to the BASH ECCS Evaluation Model include the modifications made to the 1981 ECCS Evaluation Model, discussed previously, and the following previously unreported modifications;

Several improvements were made to the BASH computer code to treat special analysis cases which are related to the tracking of fluid interfaces;

1) A modification, to prevent the code from aborting, was made to the heat transfer model for the special situation when the quench front region moves to the bottom the BASH core channel. The quench heat supplied to the fluid node below the bottom of the active fuel was set to zero.

2) A modification, to prevent the code from aborting, was made to allow negative initial movement of the liquid/two-phase and liquid-vapor interfaces. The coding these areas was generalized to prevent mass imbalance in the special case where the liquid/two-phase interface reaches the bottom of the BASH core channel.

3) Modifications, to prevent the code from aborting, were made to increase the dimensions of certain arrays for special applications.

4) A modification was made to write additional variables to the tape of information to be provided to LOCBART.

5) Typographical errors in the coding of some convective heat transfer terms were corrected, but the corrections have no effect on the BASH analysis results since the related terms are always set equal to zero.

6) A modification was made to the BASH coding to reset the cold leg conditions, in a conservative manner, when the accumulators empty. The BASH model is initialized at the bottom of core recovery with the intact cold legs, lower plenum full of liquid. Flow into the downcomer then equals the accumulator flow. The modification removed most of the intact cold leg water at the accumulator empty time by resetting the intact cold leg conditions to a high quality two phase mixture.

   In a typical BASH calculation, the downcomer is nearly full when the accumulators emptied. The delay time, prior to the intact cold leg water reaching saturation, is sufficient to allow the downcomer to fill from the addition of safety injection fluid before the water in the cold legs reaches saturation. When the intact cold leg water reached saturation it merely flowed out of the break. The cold leg water therefore, did not affect the reflood transient.

However, in a special case, a substantial time was required to fill the downcomer after the accumulators emptied. The fluid in the intact cold legs reached saturation before the downcomer filled, which artificially perturbed the transient response by incorrectly altering the downcomer fluid conditions causing the code to abort.

Effect of Change:

For typical calculations, there is no effect on the PCT calculation for the majority of the changes discussed above. A conservative estimate of the effect of the modifications on the calculations was determined to be less than 10°F, singly or in combination.

III. MODIFICATIONS TO THE NOTRUMP SMALL BREAK LOCA EVALUATION MODEL

Background:

The NOTRUMP small break LOCA ECCS Evaluation Model (Reference 4) was developed by Westinghouse in cooperation with the Westinghouse Owners Group to address technical issues expressed in NUREG-0611, "Small Break LOCA and Feedwater Transients in W PWRs," in compliance with the requirements of NUREG-0737, "Implementation of the TMI Action Plan," Section II.K.3.30. In the NOTRUMP small break LOCA ECCS Evaluation model, the NOTRUMP code is used to calculate the thermal-hydraulic response of the reactor coolant system during a small break LOCA and the SBLOCTA-IV computer program is used to calculated the performance of fuel rods in the hot assembly.

Several modifications have been made to the NOTRUMP computer (Reference 1) to correct erroneous coding or improve the coding logic to preclude err..eous calculations. The modifications indicated in A through I below have been incorporated into the production version of the code. Remaining corrections and modifications are not significant and will be incorporated during the next code update in accordance with the Westinghouse quality assurance procedures for computer code maintenance. The following modifications to the NOTRUMP small break LOCA ECCS Evaluation Model have been made:

A. Change Description:

A modification was made to preclude changing the region designation (upper, lower) for a node in a stack which does not contain the mixture-vapor interface. The purpose of the modification was to enhance tracking of the mixture-vapor interface in a stacked series of fluid nodes and to preclude a node in a stack, which does not contain the mixture-vapor interface, from changing the region designation.

The update does not affect the fluid conditions in the node, only the designation of the region of the node. The region designation does not typically affect the calculations, except for the nodes representing the core fluid volume (core nodes). In core nodes which are designated as containing vapor regions, the use of the steam cooling heat transfer correlation is forced on the calculation in compliance with the requirements of Appendix K to 10CFR50, even if the node conditions would indicate otherwise. The use of the steam cooling heat transfer regime above the mixture level is documented on page 3-1 of reference 2.

Effect of Change:

In rare instances, an incorrect heat transfer correlation could be selected if the region designation was improperly reflected. An analysis calculation was performed for a three-loop plant which resulted in a decrease in the PCT of 6.5°F when the corrections were made for a calculation which would be affected by the change.

B. Change Description:

Typographical errors in the equations which calculate the heat transfer rate derivatives for subcooled, saturated, and superheated natural convection conditions for the upper region of interior fluid nodes were corrected. The heat transfer rate derivatives for subcooled, saturated, and superheated natural convection conditions for the upper region of interior fluid nodes are given by equations 6-55, 6-56, and 6-57 of reference 2. A typographical error led to the use of the lower region heat transfer area instead of the upper region heat transfer area in the calculation of derivatives. The error affected only the upper region heat transfer derivatives which are used by the code to characterize the implicit coupling of the heat rates to changes in the independent nodal variables.

Effect of Change:

In rare instances, the amount of heat that could be transferred to the fluid could be improperly calculated. The effect of the errors was expected to be small since the error would only affect the derivatives of the heat rates for vapor regions that are in natural convection. An analysis calculation was performed for a three-loop plant which resulted in a larger than expected increase in the PCT of 36.7°F when the correction was made on a calculation which would be affected by the change.

C. Change Description:

Typographical errors in equations which calculate the derivatives of the natural convection mode of heat transfer in the subroutine HEAT were corrected. A conductivity term used in the equations which calculate the derivatives of the natural convection mode of heat transfer was incorrectly typed as CK (to be used for the Thom or McBeth correlations), instead of CKNC (to be used for the desired McAdams correlation).

Effect of Change:

A review of the code logic was performed to assess the effect of the error. In all equations that contain the typographical error, the incorrect variable is multiplied by zero. Therefore the typographical errors have no effect on the PCT results of the calculations.

D. Change Description:

A typographical error was corrected in an equation which calculates the internal energy for nodes associated with the reactor coolant pump model when the associated reactor coolant pump flow links are found to be in critical flow. An incorrect value for the mixture region internal energy in the fluid node downstream of a pump flow link would be calculated if the pump flow link were in critical flow.

Effect of Change:

This section of coding is not expected to be executed for small break LOCA Evaluation Model calculations since critical flow in the reactor coolant pump flow links does not occur. Therefore this modification has no effect on the calculations. This was confirmed in an analysis calculation for a three-loop plant which demonstrated no change to the PCT.

E. Change Description:

A modification was made to properly call some doubly dimensioned variables in subroutines INIT and TRANSNT. Some variables are doubly dimensioned (X,Y) but were being used as if they were singly dimensioned.

Effect of Change:

A detailed review of the code logic indicated that all of the doubly dimensioned variables had 1 as the second dimension in any of the erroneous calls. The computer inferred a 1 for the second dimension in the improper subroutine calls. Therefore, there is no effect of this modification on the PCT.

F. Change Description:

A modification was made to prevent code aborts resulting from implementation of a new FORTRAN compiler. Due to the different treatments of the precision of numbers between the FORTRAN compilers, the subtraction of two large, but close numbers resulted in exactly zero. The zero value was used in the denominator of a derivative equation, which resulted in the code aborts. This situation only occurred when the mass of a region in a node approached, but was not equal to zero.

Effect of Change:

An analysis calculation was performed for a four-loop plant which resulted in a larger than expected increase in the PCT of 4.8°F when the modification was implemented.

G. Change Description:

An error in the implementation of equation 5-33 of reference 2 was corrected. Equations 5-33 describes the calculation of the flow link friction parameter c^k for single phase flow in a non-critical flow link k. In the erroneous implementation, equation 5-33 was replaced by equation 5-34 which is used for all flow conditions. For the case where the flow quality is zero, equation 5-34 is similar In form to equation 5-33 since the two-phase friction multipliers are exactly unity when the flow quality is zero and the donor cell and flow link fluids are saturated, equations 5-33 and 5-34 are equivalent. However, for subcooled flow the flow link specific volume v_k in equation 5-33 is not equivalent to the saturated fluid donor cell specific volume (v_{k},donor(k)) in equation 5-34.

Effect of Change:

This modification was expected to have only a small beneficial effect on the analysis. However, an analysis calculation was performed for a three-loop plant to quantify the effect and a larger than expected decrease in tne peak cladding temperature of 217°F resulted. Larger than expected peak cladding temperature sensitivities, in some instances, have been observed when analyses to support safety evaluations of the effect of plant design changes under 10CFR50.59 were performed using the NOTRUMP computer code. The unexpected sensitivity results are under investigation at Westinghouse and may be due to the artificial restrictions on loop seal steam venting placed on the model for conservatism. Evaluations of the effect of this change will be examined as part of the investigation of the larger than expected sensitivity results.

H. Change Description:

A modification was made to correct an error in implementing equations L-28, L-52 and L-29, L-53 of reference 2. The two pairs of equations respectively describe the Partial derivatives of F^k with respect to pressure and specific enthalpy. F^k is an interpolation parameter that is defined by equations L-27, L-51 of reference 2. In each pair the lower equation number is for the subcooled condition, and the higher equation number is for the superheated condition The denominator of each equation contains the differences between h^k and h^{k-1} where h^k is defined by equations L-21, L-45 and h^{k-1} is defined by equations L-22, L-46 of reference 2. Although the expression defining h^k and h^{k-1} were correctly calculated in NOTRUMP, they were not used in equations L-28, L-52 and L-29, L-53 as they should have been.

Effect of Change:

An analysis calculation was performed for a four-loop plant which resulted in a decrease in the PCT of 12.8°F when the modification was made for a calculation which would be affected.

IV. MODIFICATIONS TO THE SMALL BREAK LOCTA-IV COMPUTER CODE

The following modifications to the LOCTA-IV computer code in the small break LOCA ECCS Evaluation Model have been made:

A. Change Description:

A test was added in the rod-to-steam radiation heat transfer coefficient calculation to preclude the use of the correlation when the wall-to-steam temperature-differential dropped below the useful range of the correlation. This limit was derived based upon the physical limitations of the radiation phenomena.

Effect of Change:

There is no effect of the modification on reported PCTs since the erroneous use of the correlation forced the calculations into aborted conditions.

B. Change Description:

An update was performed to allow the use of fuel rod performance data from the revised Westinghouse (PAD 3.3) model.

Effect of Change:

An evaluation indicated that there is an insignificant effect of the modification on reported PCTs.

C. Change Description:

Modifications supporting a general upgrade of the computer program were Implemented as follows:
1) the removal of unused or redundant coding,
2) better coding organization to increase the efficiency of calculations, and
3) improvements in user friendliness
   a) through defaulting of some input variables,
   b) simplification of input,
   c) input diagnostic checks, and
   d) clarification of the output.

Effect of Change:

Verification analyses calculations demonstrated that there was no effect on the calculated output resulting from these changes.

D. Change Description:

Two modifications improving the consistency between the Westinghouse fuel rod performance data (PAD) and the small break LOCTA-IV fuel rod models were implemented:

1) The form of the equation for the density of uranium-dioxide in the specific heat correlation, which modeled three dimensional expansion was corrected to account for only two-dimensional thermal expansion due to the way the fuel rod is modeled.

2) An error in the equation for the pellet/clad contact pressure was corrected. The contact resistance is never used in licensing calculations.

Effect of Change:

The uranium-dioxide density correction is estimated to have a maximum PCT benefit of less than 2°F, while the contact resistance modification has no PCT effect since it is not used.

V. FUEL ROD MODEL REVISIONS

Background:

During the review of the original Westinghouse ECCS Evaluation Model following the promulgation of 10CFR50.46 in 1974, Westinghouse committed to maintain consistency between future loss-of-coolant accident (LOCA) fuel rod computer models and the fuel rod design computer models used to predict fuel rod normal operation performance. These fuel rod design codes are also used to establish initial conditions for the LOCA analysis.

Change Description:

It was found that the large break and small break LOCA code versions were not consistent with fuel design codes in the following areas:

1. The LOCA codes were not consistent with the fuel rod design code relative to the flux depression factors at higher fuel enrichment.

2. The LOCA codes were not consistent with the fuel rod design code relative to the fuel rod gap gas conductivities and pellet surface roughness models.

3. The coding of the pellet/clad contact resistance model required revision.

Modifications were made to the fuel rod models used in the LOCA Evaluation Models to maintain consistency with the latest approved version of the fuel rod design code.

In addition, it was determined that integration of the cladding strain rate equation used in the large break LOCA Evaluation Model, as described in Reference 5, was being calculated twice each time step instead of once. The coding was corrected to properly integrate the strain rate equation.

Effect of Changes:

The changes made to make the LOCA fuel rod models consistent with the fuel design codes were judged to be insignificant, as defined by 10CFR50.46(a)(3)(i). To quantify the effect on the calculated peak cladding temperature (PCT), calculations were performed which incorporated the changes, including the cladding strain model correction for the large break LOCA. For the large break LOCA Evaluation Model, additional calculations, incorporating only the cladding strain corrections were performed and the results supported the conclusion that compensating effects were not present. The PCT effects reported below will bound the effects taken separately for the large break LOCA.

a) Large Break LOCA

The effect of the changes on the large break LOCA peak cladding temperature was determined using the BASH large break LOCA Evaluation Model. The effects were judged applicable to older evaluation models. Several calculations were performed to assess the effect of the changes on the calculated results as follows:

1. Blowdown Analysis -

It was determined that the changes will have a small effect on the core average rod and hot ... average rod performance during the blowdown analysis. The effect of the changes on the blowdo...

analysis was determined by performing a blowdown depressurization computer calculation for a typical three-loop plant and a typical four-loop plant using the SATAN-VI computer code.

2. Hot Assembly Rod Heatup Analysis -

The hot rod heatup calculations would typically show the largest effect of the changes. Hot rod heatup computer analysis calculations were performed using the LOCBART computer code to assess the effect of the changes on the hot assembly average rod, hot rod and adjacent rod.

3. Determination of the Effect on the Peak Cladding Temperature -

The effect of the changes on the calculated peak cladding temperature was determined by performing a calculation for typical three-loop and four-loop plants using the BASH Evaluation Model. The analysis calculations confirmed that the effect of the ECCS Evaluation Model changes were insignificant as defined by 10CFR50.46(a)(3)(i). The calculations showed that the peak cladding temperatures increased by less than by 10°F for the BASH Evaluation Model. It was judged that 25°F would bound the effect on the peak cladding temperature for the BART Evaluation Model, while calculations performed for the Westinghouse 1981 Evaluation Model showed that the peak cladding temperature could increase by approximately 41°F.

b) Small Break LOCA

The effect of the changes on the small break LOCA analysis peak cladding temperature calculations was determined using the 1985 small break LOCA Evaluation Model by performing a computer analysis calculations for a typical three-loop plant and a typical four-loop plant. The analysis calculations confirmed that the effect of the changes on the small break LOCA ECCS Evaluation Model were insignificant as defined by 10CFR50.46(a)(3)(i). The calculations showed that 37°F would bound the effect on the calculated peak cladding temperatures for the four-loop plants and the three-loop plants. It was judged that an increase of 37°F would bound the effect of the changes for the 2-loop plants.

VI. SMALL BREAK LOCA ROD INTERNAL PRESSURE INITIAL CONDITION ASSUMPTION

Change Description:

The Westinghouse small break loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) Evaluation Model analyses assume that higher fuel rod initial fill pressure leads to a higher calculated peak cladding temperature (PCT), as found in studies with the Westinghouse large break LOCA ECCS Evaluation Model. However, lower fuel rod internal pressure could result in decreased cladding creep (rod swelling) away from the fuel pellets when the fuel rod internal pressure was higher than the reactor coolant system (RCS) pressure. A lower fuel rod initial fill pressure could then result in a higher calculated peak cladding temperature.

The Westinghouse small break LOCA cladding strain model is based upon a correlation of Hardy's data, as described in Section 3.5.1 of Reference 5. Evaluation of the limiting fuel rod initial fill pressure assumption revealed that this model was used outside of the applicable range in the small break LOCA Evaluation Model calculations, allowing the cladding to expand and contract more rapidly than it should. The model was corrected to fit applicable data over the range of small break LOCA conditions. Correction of the cladding strain model affects the small break LOCA Evaluation Model calculations through the fuel rod internal pressure initial condition assumption.

Effect of Changes:

Implementation of the corrected cladding creep equation results in a small reduction in the pellet to cladding gap when the RCS pressure exceeds the rod internal pressure and increases the gap after RCS pressure falls below the rod internal pressure. Since the cladding typically demonstrates very little creep toward the fuel pellet prior to core uncovery when the RCS pressure exceeds the rod internal pressure, implementation of the correlation for the appropriate range has a negligible benefit on the peak cladding temperature calculation during this portion of the transient. However, after the RCS pressure falls below the rod internal pressure, implementation of an accurate correlation for cladding creep in small break LOCA analyses would reduce the expansion of the cladding away from the fuel compared to what was previously calculated and results in a PCT penalty because the cladding is closer to the fuel.

Calculations were performed to assess the effect of the cladding strain modifications for the limiting three-inch equivalent diameter cold leg break in typical three-loop and four-loop plants. The results indicated that the change to the calculated peak cladding temperature resulting from the cladding strain model change would be less than 20°F. The effect on the calculated peak cladding temperature depended upon when the peak cladding temperature occurs and whether the rod internal pressure was above or below the system pressure when the peak cladding temperature occurs. For the range of fuel rod internal pressure initial conditions, the combined effect of the fuel rod internal pressure and the cladding strain model revision is typically bounded by 40°F. However, in an extreme case the combined effect could be as large as 60°F.

BVPS-1 was initially assigned a PCT penalty of 20°F pertaining to the Creep Model and 33°F pertaining to the fill pressure assumption. Subsequently, a plant-specific sensitivity analysis found the 20°F penalty was not applicable. Westinghouse is currently evaluating a SBLOCA Burst/Blockage Issue which is potentially more limiting at BVPS-1 than the issue discussed above. The Rod Internal Pressure item is associated with a transient configuration where rod burst does not occur. On the other hand, the Burst/Blockage issue applies if the rod bursts at the limiting time in life. The rod burst causes a rather sharp PCT spike as both sides of the clad react with water. Since a rod cannot both burst and not burst, the higher PCT penalty from either scenario is applied to PCT. For Unit 1, the Burst/Blockage penalty of 204°F (temporary) exceeds the non-burst issues penalty (33°F) which no longer contributes to calculation of the most limiting PCT. The Burst/Blockage evaluation technique is PCT dependent, in an exponential fashion due to the dependence of the Zirc-water reaction on clad temperature, and therefore was derived after inclusion of other changes and error estimates affecting the model.

VII. NOTRUMP CODE SOLUTION CONVERGENCE

Change Description:

In the development of the NOTRUMP small break LOCA ECCS Evaluation Model, a number of noding sensitivity studies were performed to demonstrate acceptable solution convergence as required by Appendix K to 10CFR50. Temporal solution convergence sensitivity studies were performed by varying input parameters which govern the rate of change of key process variables, such as changes in the pressure, mass, and internal energy. Standard input values were specified for the input parameters which govern the time step size selection. However, since the initial studies, modifications were made to the NOTRUMP computer program to enhance code performance and implement necessary modifications (Reference 7). Subsequent to the modifications, solution convergence was not re-confirmed.

To analyze changes in plant operating conditions, sensitivity studies were performed with the NOTRUMP computer code for variations in initial RCS pressure, auxiliary feedwater flow rates, power distribution, etc., which resulted in peak cladding temperature (PCT) variations which were greater than anticipated based upon engineering judgment. In addition, the direction of the PCT variation conflicted with engineering judgment expectations in some cases. The unexpected variability of the sensitivity study results indicated that the numerical solution may not be properly converged.

Sensitivity studies were performed for the time step size selection criteria which culminated in a revision to the recommended time step size selection criteria inputs. Fixed input values originally recommended for the steady state and all break transient calculations were modified to assure converged results. The NOTRUMP code was re-verified against the SUT-08 Semiscale experiment and it was confirmed that the code adequately predicts key small break phenomena.

Effect of Changes:

Generally, the modifications result in small shifts in timing of core uncovery and recovery. However, these changes may result in a change in the calculated peak cladding temperature which exceeds 50°F for some plants. Based on representative calculations, however, this change will most likely result in a reduction in the calculated peak cladding temperature. Since the potential beneficial effect of a non-converged solution is plant specific, a generic PCT effect cannot be provided. However, it has been concluded that current licensing basis results remain valid since the results are conservative relative to the change.

VIII. LARGE BREAK LOCA BURST AND BLOCKAGE ASSUMPTION

Background:

The cladding swelling and flow blockage models were reviewed in detail during the NRC's evaluation of the Westinghouse Evaluation Model. However, the use of the average rod in the hot assembly may not have been documented in a manner detailed enough to allow the staff to adequately assess this aspect of the model.

Appendix K to 10CFR50 requires consideration of the effects of flow blockage resulting from the swelling and rupture of the fuel rods during a loss-of-coolant accident (LOCA). 10CFR50 Appendix K Paragraph I.B states:

"... To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated."

In Westinghouse ECCS Evaluation Model calculations, the average rod in the hot assembly is used as the basis for calculating the effects of flow blockage. If a significant number of fuel rods in the hot assembly are operating at power levels greater than that of the average rod, the time at which cladding swelling and rupture is calculated to occur may be predicted later in the LOCA transient, since the lower power rod will take longer to heat up to levels where swelling and rupture will occur.

A review of the Westinghouse model used to predict assembly blockage was performed. This model was developed from the Westinghouse Multi-Rod Burst Tests (MRBT) and was the model used to determine assembly wide blockage until replaced by the NUREG-0630 model starting in 1980. These models provide the means for determining assembly wide blockage once the mean burst strain has been established. Implementation of these burst models has relied upon the average rod to provide the mean burst strain. The average rod is a low power rod producing the power of the average of rods in the hot assembly and is primarily used to calculate the enthalpy rise in the hot assembly. Use of the average rod in the model assumes that the time at which blockage is calculated to occur is represented by the burst of the average rod. A review of current hot assembly power distributions indicates that in general the average rod in the hot assembly is also representative of the largest number of rods in the assembly, so that burst of this rod adequately represents when most of the rods will burst. With this representation, however, the true onset of blockage would likely begin earlier, as the highest power rods reach their burst temperature. This time is estimated to be a few seconds prior to the time when the average rod bursts.

Large break LOCA Evaluation Models which use BART or BASH simulate the hot assembly rod with the actual average power, while older Evaluation Models use an average rod power which is adjusted downward to account for thimbles (this is described in detail in Addendum 3 to reference (6)). If burst occurs after the flooding rate has fallen below one inch per second, the time at which the blockage penalty is calculated will be delayed for these older Evaluation Models.

Change Description:

Ample experimental evidence currently exists which shows that flow blockage does not result in a heat transfer penalty during a LOCA. In addition, newer Evaluation Models have been developed and licensed which demonstrate that the older Evaluation Models contain a substantial amount of conservatism. Westinghouse concluded that further artificial changes to the ECCS Evaluation Models to force the calculation of an earlier burst time were not necessary. In rare instances where burst has not occurred prior to the flooding rate falling below 1.0-inch/second, the results of the ECCS analysis calculation are supplemented by a permanent assessment of margin. Typically this will only occur in cases where the calculated PCT is low. Westinghouse concludes that no model change is required to calculate an earlier burst time.

IX. STEAM GENERATOR FLOW AREA

Background:

Licensees are normally required to provide assurance that there exists only an extremely low probability of abnormal leakage or gross rupture of any part of the reactor coolant pressure boundary (General design criteria 14 and 31). The NRC issued a regulatory guide (RG 1.121) which addressed this requirement specifically for steam generator tubes in pressurized water reactors. in that guide, the staff required analytical and experimental evidence that steam generator tube integrity will be maintained for the combinations of the loads resulting from a LOCA with the loads from a safe shutdown earthquake (SSE). These loads are combined for added conservatism in the calculation of structural integrity. This analysis provides the basis for establishing criteria for removing from service tubes which had experienced significant degradation.

Analyses performed by Westinghouse in support of the above requirement for various utilities, combined the most severe LOCA loads with the plant specific SSE, as delineated in the design criteria and the Regulatory Guide. Generally, these analyses showed that while tube integrity was maintained, the combined loads led to some tube deformation. This deformation reduces the flow area through the steam generator. The reduced flow area increases the resistance through the steam generator to the flow of steam from the core during a LOCA, which potentially could increase the calculated PCT.

Change Description:

The effect of tube deformation and flow area reduction in the steam generator was analyzed and evaluated for some plants by Westinghouse in the late 1970's and early 1980's. The combination of LOCA and SSE loads led to the following calculated phenomena:

1. LOCA and SSE loads cause the steam generator tube bundle to vibrate.

2. The tube support plates may be deformed as a result of lateral loads at the wedge supports at the periphery of the plate. The tube support plate deformation may cause tube deformation.

3. During a postulated large LOCA, the primary side depressurizes to containment pressure. Applying the resulting pressure differential to the deformed tubes causes some of these tubes to collapse, and reduces the effective flow area through the steam generator.

4. The reduced flow area increases the resistance to venting of steam generated in the core during the reflood phase of the LOCA, increasing the calculated peak cladding temperature (PCT).

The ability of the steam generator to continue to perform its safety function was established by evaluating the effect of the resulting flow area reduction on the LOCA PCT. The postulated break examined was the steam generator outlet break, because this break was judged to result in the greatest loads on the steam generator, and thus the greatest flow area reduction. It was concluded that the steam generator would continue to meet its safety function because the degree of flow area reduction was small, and the postulated break at the steam generator outlet resulted in a low PCT.

In April of 1990, in considering the effect of the combination of LOCA + SSE loadings on the steam generator component, it was determined that the potential for flow area reduction due to the contribution of SSE loadings should be included in other LOCA analyses. With SSE loadings, flow area reduction may occur in all steam generators (not just the faulted loop). Therefore, it was concluded that the effects of flow area reduction during the most limiting primary pipe break affecting LOCA PCT, i.e., the reactor vessel inlet break (cold leg break LOCA), had to be evaluated to confirm that 10CFR50.46 limits continue to be met and that the affected steam generators will continue to perform their intended safety function.

Consequently, the action was taken to address the safety significance of steam generator tube collapse during a cold leg break LOCA. The effect of flow area reduction from combined LOCA and SSE loads was estimated. The magnitude of the flow area reduction was considered equivalent to an increased level of steam generator tube plugging. Typically, the area reduction was estimated to range from 0 to 7.5%, depending on the magnitude of the seismic loads. Since detailed non-linear seismic analyses are not available for Series 51 and earlier design steam generators, some area reductions had to be estimated based on available information. For most of these plants, a 5 percent flow area reduction was assumed to occur in each steam generator as a result of the SSE. For these evaluations, the contribution of loadings at the tube support plates from the LOCA cold leg break was assumed negligible, since the additional area reduction, if it occurred, would occur only in the broken loop steam generator.

Westinghouse recognizes that, for most plants, as required by GDC 2, "Design Basis for Protection Against Natural Phenomena", that steam generators must be able to withstand the effects of combined LOCA + SSE loadings and continue to perform their intended safety function. It is judged that this requirement applies to undegraded as well as locally degraded steam generator tubes. Compliance with GDC 2 is addressed below for both conditions.

For tubes which have not experienced cracking at the tube support plate elevations, it is Westinghouse's engineering judgment that the calculation of steam generator tube deformation or collapse as a result of the combination of LOCA loads with SSE loads does not conflict with the requirements of GDC 2. During a large break LOCA, the intended safety functions of the steam generator tubes are to provide a flow path for the venting of steam generated in the core through the RCS pipe break and to provide a flow path such that the other plant systems can perform their intended safety functions in mitigating the LOCA event.

Tube deformation has the same effect on the LOCA event as the plugging of steam generator tubes. The effect of tube deformation and/or collapse can be taken into account by assigning an appropriate PCT penalty, or accounting for the area reduction directly in the analysis. Evaluations completed to date show that tube deformation results in acceptable LOCA PCT. From a steam generator structural integrity perspective, Section III of the ASME Code recognizes that inelastic deformation can occur for faulted condition loadings. There are no requirements that equate steam generator tube deformation, per se, with loss of safety function. Cross-sectional bending stresses in the tubes at the tube support plate elevations are considered secondary stresses within the definitions of the ASME Code and need not be considered in establishing the limits for allowable steam generator tube wall degradation. Therefore, for undegraded tubes, for the expected degree of flow area reduction and despite the calculation showing potential tube collapse for a limited number of tubes, the steam generators continue to perform their required safety functions after the combination of LOCA + SSE loads, meeting the requirements of GDC 2.

During a November 7, 1990 meeting with a utility and the NRC staff on this subject, a concern was raised that tubes with partial wall cracks at the tube support plate elevations could progress to through-wall cracks during tube deformation. This may result in the potential for significant secondary to primary inleakage during a LOCA event; it was noted that inleakage is not addressed in the existing ECCS analysis. Westinghouse did not consider the potential for secondary to primary inleakage during resolution of the steam generator tube collapse item. This is a relatively new item, not previously addressed, since cracking at the tube support plate elevations had been insignificant in the early 1980's when the tube collapse item was evaluated in depth. There is ample data available which demonstrates that undegraded tubes maintain their integrity under collapse loads. There is also some data which shows that cracked tubes do not behave significantly differently from uncracked tubes when collapse loads are applied. However, cracked tube data is available only for round or slightly ovalized tubes.

It is important to recognize that the core melt frequency resulting from a combined LOCA + SSE event, subsequent tube collapse, and significant steam generator tube inleakage is very low, on the order of 10^{-8}/RY or less. This estimate takes into account such factors as the possibility of a seismically induced LOCA, the expected occurrence of cracking in a tube as a function of height in the steam generator tube bundle, the localized effect of the tube support plate deformation, and the possibility that a tube which is identified to deform during LOCA + SSE loadings would also contain a partial through-wall crack which would result in significant inleakage. To further reduce the likelihood that cracked tubes would be subjected to collapse loads, eddy current inspection requirements can be established. The inspection plan would reduce the potential for the presence of cracking in the regions of the

tube support plate elevations near wedges that are most susceptible to collapse which may then lead to penetration of the primary pressure boundary and significant inleakage during a LOCA + SSE event.

Change Description:

As noted above, detailed analyses which provide an estimate of the degree of flow area reduction due to both seismic and LOCA forces are not available for all steam generators. The information that does exist indicates that the flow area reduction may range from 0 to 7.5 percent, depending on the magnitude of the postulated forces, and accounting for uncertainties. It is difficult to estimate the flow area reduction for a particular steam generator design, based on the results of a different design, due to the differences in the design and materials used for the tube support plates.

While a specific flow area reduction has not been determined for some earlier design steam generators, the risk associated with flow area reduction and tube leakage from a combined seismic and LOCA event has been shown to be exceedingly low. Based on this low risk, it is considered adequate to assume, for those plants which do not have a detailed analysis, that 5 percent of the tubes are susceptible to deformation.

The effect of potential steam generator area reduction on the cold leg break LOCA peak cladding temperature has been either analyzed or estimated for each Westinghouse plant. A value of 5 percent area reduction has been applied, unless a detailed non-linear analysis is available. The effect of tube deformation and/or collapse will be taken into account by allocating the appropriate PCT margin, or by representing the area reduction by assuming additional tube plugging in the analysis.

X. AUXILIARY FEEDWATER ENTHALPY SWITCHOVER FOR SBLOCA ANALYSIS

Change Description:

During a review of Westinghouse SBLOCA analysis methods, a question arose with respect to the computer code input used to represent the time required for the lower flow, lower enthalpy auxiliary feedwater to purge the higher enthalpy main feedwater from the feedwater piping after actuation of auxiliary feedwater. In the Westinghouse SBLOCA ECCS Evaluation models using either the WFLASH or NOTRUMP analysis technologies, this time is used to switch the enthalpy of the fluid provided to the steam generators from the main feedwater enthalpy to the auxiliary feedwater enthalpy.

Effect of Change:

A review and investigation of the concern indicated that, in some instances, the time assumed for the auxiliary feedwater enthalpy purge delay time was shorter than times calculated from the actual plant configuration. The inconsistency between the Westinghouse SBLOCA ECCS Evaluation Model input value and a value corresponding to the plant configuration results from the specific guidance provided to the analyst for determining the auxiliary feedwater enthalpy delay time. In both the WFLASH and NOTRUMP methods, a standard purge delay time was recommended. In the NOTRUMP analysis methodology, a standard input value judged to be conservative based upon phenomena observed during experiment SUT-08 in the Semiscale test facility was used. However, further investigation showed that the standard input value could result in a non-conservative calculation of the peak cladding temperature.

XI. POWER SHAPE SENSITIVITY MODEL (PSSM)

Background:

Large Break LOCA analyses have been traditionally performed using a symmetric, chopped cosine axial power shape. Recent calculations have shown that there was a potential for top-skewed power distributions to result in Peak Cladding Temperatures (PCT) greater than those calculated with a chopped cosine axial power distribution. Westinghouse has developed a process, which was applied to the reload for Beaver Valley Unit 2 Cycle 4, that ensures that the cosine remains the limiting power distribution, by defining appropriate power distribution surveillance data. This process, called the Power Shape Sensitivity Model (PSSM), is described in a topical report (WCAP-12909-P).

In May, 1991, Westinghouse transmitted to the NRC the report titled, "Westinghouse ECCS Evaluation Model: Revised Large Break LOCA Power Distribution Methodology," which describes the process that Westinghouse is now using to more accurately account for the effect of power distribution in the core reload design. In January, 1991, the implementation of this approach was discussed with the NRC. In a May, 1991 meeting with the NRC, Westinghouse again told the NRC that they planned on implementing the PSSM process shortly after the topical report was submitted. Westinghouse indicated in the transmittal letter of the topical (NS-NRC-91-3578) that it was their intent to implement the PSSM process for future reload design applications. The NRC has informally stated that it is acceptable to use revised LOCA methodology that corrects a potential deficiency, until it has been reviewed by the Staff.

Effect of Change:

Implementation of this methodology has no effect on calculated PCT.

XII. PLANT-SPECIFIC INPUTS AND MISCELLANEOUS INPUT ERRORS

Background:

To maintain conformance with 10CFR50.46 PCT limits, compensatory benefits had to be documented for the BVPS-2 SBLOCA analysis. The compensatory benefit was derived by performing plant specific sensitivity analyses. The benefit results from the following:

Plant specific peaking factors consistent with the Cycle 3 & Cycle 4 design/Technical Specification limits were used. The previous UFSAR analysis was based in part on bounding generic values.

Plant specific Axial Offset consistent with the Cycle 3 & 4 design/Technical Specification limits, using the current plant specific methodology was used. The previous UFSAR analysis was based on generic power shapes.

A miscellaneous input error(s) in the UFSAR analysis was corrected.

Effect of Change:

The combination of the above changes and error corrections results in a 145° F reduction in calculated PCT.

XIII. RCS TEMPERATURE DISTRIBUTION

Background:

While evaluating the effects of reduced thermal design flow (TDF), anomalies were discovered in the interaction between the RCS temperature distribution methodology and the actual analysis inputs for both SBLOCA & LBLOCA analyses. Action was taken to investigate and evaluate the anomalies. A portion of the apparent discrepancy is attributed to slight changes in the LOCA inputs that result from miscellaneous evolutionary changes to the plant characteristics such as the Steam Generator Fouling Factor, which was recently recalculated. Other differences are attributed to deviations that occurred in the LOCA analyses themselves, such as an extraction error resulting in incorrect LBLOCA input.

Effect of Change:

The LBLOCA analysis RCS T_{avg} inputs are greater than necessary for either current TDF or reduced TDF cases, and thus the analysis remains bounding, though the benefit is unquantified.

The SBLOCA analysis RCS T_{avg} inputs are greater than necessary for either current TDF or reduced TDF cases, and thus a PCT penalty of 20°F is assigned. This PCT penalty is considered to be reportable under 10CFR50.46 as an error in the application of the Evaluation Model.

XIV. RCCA GUIDE THIMBLE AREA

Background:

The only VANTAGE 5H Zircaloy grid feature which significantly affects the SBLOCA analysis is the increase in design rod drop time. The Westinghouse Small Break model assumes the reactor core is brought to a subcritical condition by the negative reactivity of the control rods. The increase in the design rod drop time to a maximum value of 2.7 seconds exceeds the 2.4 second value in the existing SBLOCA analysis.

Effect of Change:

An evaluation was performed which determined that a 3°F PCT penalty applied for the increase in rod drop time of 0.3 seconds. The decrease in core pressure drop associated with thimble plug removal has an inconsequential effect on the SBLOCA analysis. However, the SBLOCA analysis did not model the effect of the guide thimble interior area and volume on the transient response. Studies have shown that the fluid in this volume will interact with the remaining core fluid. An assessment of this interaction based on previous sensitivities indicated that a 17°F PCT penalty applies.

XV. Auxiliary Feedwater Flow Table Error

Change Description:

The steam generator auxiliary feedwater (AFW) flow rate is governed by the timing variable TIMESG(I). A minor logic error associated with this variable was discovered which led to a step change in the AFW flow rate once the transient time passed the value of TIMESG(7). Typically, this value is set equal to 11000 seconds and so this error would only affect very long transient calculations. In addition, the nature of the error is to allow the AFW flow rate to immediately revert to the full value of the Main Feedwater flow rate. This enormous step change has led to code aborts in the cases where it has occurred.

This logic was corrected as a "Discretionary Change" as described in Section 4.1.1 of WCAP-13451. This determination is based on the fact that SBLOCA transients are generally terminated before the logic error can have an effect coupled with the code's lack of capability to handle the step change if it does occur. Therefore, it was reasoned that the logic could not affect LOCA results.

Effect of Change:

This error correction has no effect on any current or prior applications of the Evaluation Model.

XVI. Steam Generator Secondary Side Modeling Enhancements

Change Description:

A set of related changes which make steam generator secondary side modeling more convenient for the user were implemented into NOTRUMP. This model improvement involved several facets of feedwater flow modeling. First, the common donor boundary node for the standard evaluation model nodalization has been separated into two identical boundary nodes. These donor nodes are used to set the feedwater enthalpy. The common donor node configuration did not allow for loop specific enthalpy changeover times in cases where asymmetric AFW flow rates or purge volumes were being modeled for plant specific sensitivities.

The second improvement is the additional capability to initiate main feedwater isolation on either loss of offsite power coincident with reactor trip (low pressurizer pressure) or alternatively on safety injection signal low-low pressurizer pressure). The previous model allowed this function only on loss of offsite power coincident with reactor trip. The auxiliary feedwater pumps are still assumed to start after a loss of offsite power with an appropriate delay time to model diesel generator start-up and buss loading times.

The final improvement is in the area of modeling the purging of high enthalpy main feedwater after auxiliary feedwater is calculated to start. This was previously modeled through an approximate time delay necessary to purge the lines of the high enthalpy main feedwater before credit could be taken for the much lower enthalpy auxiliary feedwater reaching the steam generator secondary. This time delay was a function of the plant specific purge volume and the auxiliary feedwater flow rate. The new modeling allows the user to input the purge volume directly. This then is used together with the code calculated integrated feedwater flow to determine the appropriate time at which the feedwater enthalpy can be assumed to change.

These improvements are considered to be a "Discretionary Change" as described in Section 4.1.1 of WCAP-13451. Since they involve only enhancements to the capabilities and useability of the evaluation model, and not changes to results calculated consistently with the previous model, these changes were implemented without prior review as discussed in Section 4.1.1 of WCAP-13451.

Effect of Change:

Because these enhancements only allow greater ease in modeling plant specific steam generator secondary side behavior over the previous model, it is estimated that no effect will be seen in evaluation model calculations.

XVII. Structural Metal Heat Modeling

Change Description:

A discrepancy was discovered during review of the finite element heat conduction model used in the WREFLOOD-INTERIM code to calculate heat transfer from structural metal in the vessel during the reflood phase. It was noted that the material properties available in the code corresponded to those of stainless steel. While this is correct for the internal structures, it is inappropriate for the vessel wall which consists of carbon steel with a thin stainless internal clad. This was defined as a "Non-discretionary Change" per Section 4.1.2 of WCAP-13451, since there was thought to be potential for increased PCT with a more sophisticated composite model. The model was revised by replacing it with a more flexible one that allows detailed specification of structures.

Effect of Change:

The estimated effect of this correction is a 25°F PCT benefit.

XVIII. Spacer Grid Heat Transfer Error in BART

Change Description:

During investigations into anomolous wetting and dryout behavior demonstrated by the BART grid model a programming logic error was discovered in the grid heat transfer model. The error caused the solution to be performed twice for each timestep. The error was traced back to the original coding used in all of the BART and LOCBART codes. This was defined as a "Non-discretionary Change" per Section 4.1.2 of WCAP-13451. The error was corrected, and a complete reverification of the grid model was conducted and transmitted to the NRC (WCAP-10484, Addendum 1).

Estimated Effects

Calculations performed with the affected code have consistently demonstrated significantly better grid wetting and lower clad temperatures. A conservative estimate of zero degrees PCT penalty has been assigned for this issue.

XIX. Large Break LOCA BASH Metal Heat Link Error

Background:

During the reflood portion of the accident, the BASH code determines the flooding rate. Due to a code input error associated with the Thermal Shield heat link, the analysis resulted in no heat transfer from the Thermal Shield to the downcomer region of the BVPS-1 vessel. This in turn resulted in a non-conservative effect on the reflood period flooding rate.

Effect of Change:

Westinghouse has performed plant-specific sensitivity studies for this input error and concluded that the effect is an 82°F PCT penalty. Since this penalty, if not offset, would cause PCT to exceed 2200°F, Westinghouse quantified PCT margin available in the cycle 9 core design. Since the current analysis assumes a P-bar-HA LOCA hot assembly relative peaking factor of 1.42, and the actual core design value is 1.40, plant-specific sensitivity studies demonstrate that 28°F of PCT benefit is available by changing model input assumptions. This allows the calculated PCT to remain below 2200°F.

XX. BVPS-1 Turbine Driven AFW Pump Actuation

Background:

Westinghouse has performed a Small Break LOCA (SBLOCA) technical evaluation for (1) the removal of credit for the Turbine Driven AFW actuation via Loss-of-Offsite Power Undervoltage Relays and (2) as a compensatory measure, installing a SI actuation circuit for the Turbine Driven (TD) AFW pump using a delay time of 60 seconds.

The BVPS-1 SBLOCA analy is of record is a NOTRUMP Evaluation Model analysis which assumes that AFW delivery actuates on the combination of Reactor Trip/coincident Loss-of-Offsite Power (LOOP), consistent with the model presented in the NOTRUMP Topical Report. In the new design, the Motor Driven (MD) pumps and Turbine Driven pump will actuate, with a 60 second delay.

Although two MD and one TD AFW pumps are available, the existing analysis credits only one MD and one TD pump due to the limiting single failure, of one Diesel Generator, which precludes operation of the second MD pump (as well as 1 train of SI Pumps).

In the SBLOCA analysis, the Low Pressurizer Pressure (LPP) SI and LPP Reactor Trip times are both modeled, with the SI signal occurring approximately 5 seconds later due to its lower setpoint. In addition, the analysis assumed a 60 second delay time for AFW pumps actuation. Therefore, crediting AFW operation based on LPP SI signal instead of LPP Reactor Trip/coincident LOOP represents an additional 5 second delay for AFW delivery.

Effect of Change:

To complete the evaluation, AFW delivery data and BVPS-1 plant specific AFW reduction sensitivities were utilized. The evaluation concludes that a PCT penalty of 6°F is incurred.

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2. WCAP-9220-P-A, Revision 1 (Proprietary), WCAP-9221-A, Revision 1 (Non-Proprietary), "Westinghouse ECCS Evaluation Model - 1981 Version," 1981, Eicheldinger, C.

3. WCAP-10266-P-A, Revision 2 (Proprietary), WCAP-10267-A, Revision 2 (Non-Proprietary), Besspiata,J.J., et.al., "1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

4. WCAP-10054-P-A (Proprietary), WCAP-10081-A (Non-Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," Lee, N., et. al., August 1985.

5. "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8305, (Non-Proprietary), June 1974.

6. "BART-A1 A Computer Code for the Best Estimate Analysis of Reflood Transients", WCAP-9695-A (Non-Proprietary), March 1984.

7. "10CFR50.46 Annual Notification for 1989 of Modifications in Westinghouse ECCS Evaluation Models," NS-NRC-89-3463, Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), Dated October 5, 1989.

8. WCAP-12909-P (Proprietary), "Westinghouse ECCS Evaluation Model Revised LBLOCA Power Distribution Methodology," Dated May 22, 1991.

9. NS-NRC-91-3578, "Westinghouse ECCS Evaluation Model Revised LBLOCA Power Distribution Methodology," Dated May 22, 1991.

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11. WCAP-10484, Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood," Shimeck, December 1992.

12. ET-NRC-91-3633, "Methodology Clarifications to WCAP-12909-P," Letter from S.R. TRITCH (Westinghouse) to R.C. Jones (NRC), Dated November 21, 1991.