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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

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 an annual notification of changes or errors in emergency core cooling system (ECCS) evaluation models for BVPS-1 and BVPS-2. Current information for both large and small break transients 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 (Descriptions).

Attachment 2 Provides a description for each model change or error.

Attachment 3 Provides a list of references, including those identified in the various descriptions. These documents have already been provided to the NRC by Westinghouse.

The PCT effects, listed in Attachment 1, have been applied as adjustments to the appropriate PCT calculations. This results in calculated PCTs for the large and small break loss-of-coolant accident (LOCA) transients as follows:

BVPS-1 Large Break LOCA - 2018°F
BVPS-1 Small Break LOCA - 1849°F
BVPS-2 Large Break LOCA - 2044°F
BVPS-2 Small Break LOCA - 2105°F

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As committed in our letter dated September 25, 2002 (Ref. L-02-101), new small break LOCA analyses results have been reflected in this report. There are no new regulatory commitments identified in this document.

Any questions pertaining to this subject may be directed to Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

Sincerely,

L. William Pearce

Attachments:
1) Summary of PCT Effects for BVPS LOCA Transients
2) Descriptions of Model Changes or Errors
3) References

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator

ATTACHMENT 1

SUMMARY OF PCT EFFECTS FOR BVPS LOCA TRANSIENTS

ATTACHMENT 1

SUMMARY OF PCT EFFECTS FOR BVPS LOCA TRANSIENTS

DESCRIPTION	PCT EFFECT (°F)	ATTACHMENT 2 PAGE
BVPS-1 Large Break LOCA		
MODIFICATIONS TO THE WREFLOOD COMPUTER CODE	(NOTE 1)	1
MODIFICATIONS TO THE BASH ECCS EVALUATION MODEL	(NOTE 1)	2
FUEL ROD MODEL REVISIONS	(NOTE 1)	6
STEAM GENERATOR FLOW AREA	(NOTE 1)	10
STRUCTURAL METAL HEAT MODELING	(NOTE 1)	14
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BVPS-1 Small Break LOCA		
MODIFICATIONS TO THE SMALL BREAK LOCTA-IV COMPUTER CODE	(NOTE 1)	4
FUEL ROD MODEL REVISIONS	(NOTE 1)	6
SMALL BREAK LOCA ROD INTERNAL PRESSURE INITIAL CONDITION ASSUMPTION	(NOTE 1)	8
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TURBINE DRIVEN AFW PUMP ACTUATION	(NOTE 1)	15
SAFETY INJECTION IN THE BROKEN LOOP	(NOTE 1)	17
IMPROVED CONDENSATION MODEL	(NOTE 1)	17
REVISED BURST STRAIN LIMIT MODEL	(NOTE 1)	18
HOT ASSEMBLY AVERAGE ROD BURST EFFECTS	(NOTE 1)	19
SMALL BREAK LOCA LIMITING TIME IN LIFE	26	20
BVPS-2 Large Break LOCA		
MODIFICATIONS TO THE WREFLOOD COMPUTER CODE	(NOTE 1)	1
MODIFICATIONS TO THE BASH ECCS EVALUATION MODEL	(NOTE 1)	2
FUEL ROD MODEL REVISIONS	(NOTE 1)	6
STEAM GENERATOR FLOW AREA	(NOTE 1)	10
STRUCTURAL METAL HEAT MODELING	(NOTE 1)	14
LARGE BREAK LOCA ROD INTERNAL PRESSURE ISSUES	(NOTE 1)	16
REVISED BURST STRAIN LIMIT MODEL	(NOTE 1)	18
CODE STREAM IMPROVEMENT	(NOTE 1)	21
POWER DISTRIBUTION ASSUMPTION	(NOTE 1)	22
BVPS-2 Small Break LOCA		
MODIFICATIONS TO THE SMALL BREAK LOCTA-IV COMPUTER CODE	(NOTE 1)	4
FUEL ROD MODEL REVISIONS	(NOTE 1)	6
SMALL BREAK LOCA ROD INTERNAL PRESSURE INITIAL CONDITION ASSUMPTION	(NOTE 1)	8
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REVISED BURST STRAIN LIMIT MODEL	(NOTE 1)	18
HOT ASSEMBLY AVERAGE ROD BURST EFFECTS	(NOTE 1)	19
SMALL BREAK LOCA LIMITING TIME IN LIFE	138	20

Note 1: The specific impact on PCT for this item is no longer being estimated because a reanalysis has incorporated the change.

ATTACHMENT 2

<u>DESCRIPTIONS OF MODEL CHANGES OR ERRORS</u>

Attachment 2
Descriptions of Model Changes or Errors

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 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.

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 be exhibited for any plant due to the WREFLOOD logic change.

Descriptions of Model Changes or Errors
(Continued)

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.

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 singularly or in combination.

Descriptions of Model Changes or Errors
(Continued)

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.

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.

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.

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.

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 assembly average rod performance during the blowdown analysis. The effect of the changes on the blowdown 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.

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.

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.

Westinghouse is currently evaluating a SBLOCA Burst/Blockage Issue which is potentially more limiting 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. 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 is derived after inclusion of other changes and error estimates affecting the model.

Descriptions of Model Changes or Errors
(Continued)

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.

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.

Descriptions of Model Changes or Errors
(Continued)

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 to show that steam generator tube integrity will be maintained for the combinations of the loads resulting from a LOCA and 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 tubes from service 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, 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 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, and thus, 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.

Descriptions of Model Changes or Errors
(Continued)

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.

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.

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.

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.

TURBINE DRIVEN AFW PUMP ACTUATION

Change Description:

The BVPS-1 and BVPS-2 Small Break LOCA (SBLOCA) analyses differ from the approved model by (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 and BVPS-2 SBLOCA analyses of record are NOTRUMP Evaluation Model analyses which assume 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.

Descriptions of Model Changes or Errors
(Continued)

LARGE BREAK LOCA ROD INTERNAL PRESSURE ISSUES

Change Description:

Westinghouse recently completed an evaluation of a potential issue concerning the impact of increased beginning of life rod internal pressure (RIP) uncertainties on LOCA analyses. Historically, beginning of life fuel pressure and temperature uncertainties, were based upon end of life considerations. These RIP uncertainties were found to be potentially nonconservative. During the evaluation of this issue, a second issue related to the applicability of generic IFBA fuel analyses to updated LOCA Evaluation Models was also identified and combined with this issue because the underlying mechanisms were the same.

The technical evaluation of this issue concluded that both the RIP uncertainty and the current IFBA fuel designs with 200 psig initial fill pressure typically will result in a maximum ±15°F PCT variation. Consequently, RIP manufacturing uncertainties and 200 psig initial fill pressure IFBA fuel do not have significant effects on the large break LOCA analyses. Also, based on these results, it was concluded that only nominal RIP (with an upper bound bias) should be used in the LOCA analyses for fuel designs with an initial cold fill pressure ≥ 200 psig. This is consistent with past LOCA analysis.

SAFETY INJECTION IN THE BROKEN LOOP/IMPROVED CONDENSATION MODEL

Change Description:

Westinghouse recently completed an evaluation of a potential issue concerning the modeling of Safety Injection (SI) flow into the broken RCS loop for small break loss of coolant accidents (SBLOCA). Previously it had been assumed that SI to the broken RCS loop would result in a lower calculated PCT. Therefore, the ECCS broken loop branch line was modeled to spill the SI to the containment sump. The basis for this assumption included consideration for the effect of back pressure on the spilling ECCS line for cold leg breaks, which would see a higher back pressure for SI connected to the broken RCS loop when compared to spilling against containment back pressure. Spilling to the higher RCS pressure would increase SI to the intact loops, which is a benefit for PCT. The effect on intact loop SI flow rates as well as the assumption that some of the SI to the broken loop would aid in RCS/Core recovery resulted in the Westinghouse ECCS model assumption that SI to the broken loop was a benefit. However, when SI is modeled to enter into the broken loop, a significant PCT penalty is calculated by the NOTRUMP small break evaluation model.

When a newer conservative model based on prototypic test is used to model the configuration of the SI piping to the RCS cold leg in a Westinghouse designed PWR, a net PCT benefit is calculated. Improved condensation of the loop steam in the intact loops results in lower RCS pressure and larger SI flow rates. The increase in SI flow rates, due to lower RCS pressure, leads to the lower calculated PCT. Thus, the negative effects of SI into the broken loop can be offset by an improved SI condensation model in the intact RCS loops.

The improved condensation model is based on data obtained from the COSI test facility. The COSI test facility is a 1/100 scale representation of the cold leg and SI injection ports in a Westinghouse designed PWR. COSI tests demonstrated that the current NOTRUMP condensation model under-predicted condensation in the intact loops during SI and thus is a conservative model. Use of the improved condensation model has demonstrated that the current NOTRUMP small break LOCA analyses without the improved condensation model and no SI into the broken loop are more conservative (higher calculated PCT) than a case which includes SI into the broken loop and the improved condensation model.

Descriptions of Model Changes or Errors
(Continued)

REVISED BURST STRAIN LIMIT MODEL

Change Description:

A revised burst strain limit model which limits strains is being implemented into the rod heat up codes used in both Large Break and Small Break LOCA. This model, which is identical to that previously approved for use in Appendix K analyses of Upper Plenum Injection plants with WCOBRA/TRAC, as described in WCAP-10924-P-A, Rev. 1, Vol. 1, Add. 4, "Westinghouse Large Break LOCA Best Estimate Methodology: Volume 1: Model Description and Validation, Addendum 4: Model Revisions," 1991.

The estimated effect on Large Break LOCA PCT's ranges from negligible, to a moderate unquantified benefit, which will be inherent in calculations once this model is implemented. In Small Break LOCA, representative plant calculations indicate that the magnitude of the benefit is conservatively estimated to be exactly offsetting to the penalty introduced by the Hot Assembly Average Rod Burst issue.

HOT ASSEMBLY AVERAGE ROD BURST EFFECTS

Change Description:

The rod heat up code used in Small Break LOCA calculations contains a model to calculate the amount of clad strain that accompanies rod burst. However, the methodology which has historically been used is to not apply this burst strain model to the hot assembly average rod. This was done so as to minimize the rod gap and therefore maximize the heat transferred to the fluid channel, which in turn would maximize the hot rod temperature. However, due to mechanisms governing the zirc-water temperature excursion (which is the subject of the SBLOCA Limiting Time-in-Life penalty for the hot rod), modeling of clad burst strain for the hot assembly average rod can result in a penalty for the hot rod by increasing the channel enthalpy at the time of PCT. Therefore, the methodology has been revised such that burst strain will also be modeled on the hot assembly average rod.

Representative plant calculations have shown that this change introduces an approximate 10 percent increase in the SBLOCA Limiting Time-in-Life penalty on the hot rod. However, this penalty is being offset, in affected plants, by the revised Burst Strain Limit Model.

SMALL BREAK LOCA LIMITING TIME IN LIFE

Change Description:

Westinghouse recently completed an evaluation of a potential issue with regard to burst/blockage modeling in the Westinghouse small break LOCA evaluation model. This potential issue involved a number of synergistic effects, all related to the manner in which the small break model accounts for the swelling and burst of fuel rods, modeling of the rod burst strain, and resulting effects on clad temperature and oxidation from the metal/water reaction models and channel blockage.

Fuel rod burst during the course of a small break LOCA analysis was found to potentially result in a significant temperature excursion above the clad temperature transient for a non-burst case. Since the methodology for SBLOCA analyses had been to perform the analyses at or near beginning of life (BOL) condition, where rod internal pressures are relatively low, most analyses did not result in the occurrence of rod burst, and therefore may not have reflected the most limiting time in life PCT. In order to evaluate the effects of this phenomenon, Westinghouse has developed an analytical model which allows the prediction of rod burst PCT effects based upon the existing analysis of record.

CODE STREAM IMPROVEMENT

Change Description:

Revisions were made to the procedures used to interface the various codes that comprise the entire execution stream for performing a large break LOCA analysis with the BASH evaluation model. Previously, the coupled WREFLOOD/COCO code for calculating containment pressure response was transferred as a boundary condition to the BASH code. This transfer has been replaced with direct coupling of the BASH and COCO codes such that the same code used to calculated the RCS conditions during reflood, also supplies the boundary conditions for the containment pressure calculation. In conjunction with this, the portion of the WREFLOOD code which calculated the refill phase of the transient has been reprogrammed into a separate, but identical code called REFILL, which is also coupled with COCO.

This methodology revision was made only as a process improvement for conducting analyses and involved no changes to the approved physical models, nor basic solution techniques governing the solutions provided by the individual computer codes. The NRC was advised of the implementation of this methodology on a forward-fit basis via reference 16.

POWER DISTRIBUTION ASSUMPTION

Change Description:

Large Break LOCA analyses have been traditionally performed using a symmetric, chopped cosine, core axial power distribution. Under certain conditions, calculations have shown that there is a potential for top-skewed power distributions to result in Peak Cladding Temperatures (PCTs) greater than those calculated with chopped cosine axial power distributions. In 1991 Westinghouse developed a statistical methodology to evaluate and assure that the cosine distribution remains the limiting distribution. This methodology, Power Shape Sensitivity Model (PSSM), was submitted to the NRC for review and approval via Reference 8 and has been applied to both BVPS-1 and BVPS-2. Although Westinghouse believed that PSSM was conservative without additional modifications, Westinghouse later decided not to continue pursuing licensing of PSSM.

An alternate methodology to replace PSSM, ESHAPE (Explicit Shape Analysis for PCT Effects), is based on an explicit analysis of the Large Break LOCA transient with a set of skewed axial power shapes to supplement the standard analysis done with the chopped cosine. Development of this methodology was completed in June 1995. A submittal notifying the NRC of this substitution was made by Reference 17.

LOCBART RADIATION TO LIQUID LOGIC ERROR

Change Description:

An error was discovered in LOCBART that allowed radiation to liquid to occur after the core inlet flooding rate dropped below 1 in/s, if the channel blockage fraction was simultaneously equal to zero. This situation does not occur for most PWR licensing calculations but, for affected cases, resulted in an overprediction of the cladding-to-fluid heat transfer coefficient for some portion of the reflood phase of the transient. For affected plants, representative plant calculations using the LOCBART code showed that correcting the error generally produced a small-to-moderate increase in peak cladding temperature, and plant-specific assessments were derived from the representative calculations in a conservative manner.

ATTACHMENT 3

REFERENCES

ATTACHMENT 3

REFERENCES

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14. WCAP-10924-P-A, Revision 1, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology," 1991.

15. NTD-NRC-94-4343, "Interim Report of an Evaluation of a Failure to Comply Pursuant to 10 CFR 21.21(a)(2)-Closeout 94-002," letter from N. J. Liparulo (Westinghouse) to NRC, Dated November 15, 1994.

16. NTD-NRC-94-4143, "Change in Methodology for Execution of Bash Evaluation Model," letter from N. J. Liparulo (Westinghouse) to W. T. Russell (NRC), Dated May 23, 1994.

17. NTD-NRC-95-4518, "Withdrawal of WCAP 12909-P on Power Shape Sensitivity Model (PSSM)," letter from N. J. Liparulo (Westinghouse) to NRC, Dated August 7, 1995.

ATTACHMENT 3

REFERENCES

(Continued)

18. NTD-NRC-95-4477, "Transmittal of Topical Reports WCAP-14404-P and WCAP-14405-P, "Methodology for Incorporating Hot Leg Nozzle Gaps into BASH,"" letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), Dated July 25, 1995.

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