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WESTINGHOUSE EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL - SUMMARY

By

F. M. Bordelon H. W. Massie Jr.

T. A. Zordan

PWR Systems Division

Nuclear Safety

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APPROVED: Romano Salvatori

R. Salvatori, Manager

Nuclear Safety

WESTINGHOUSE ELECTRIC CORPORATION

PWR Systems Division

P. O. Box 355

Pittsburgh, Pennsylvania 15230

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ABSTRACT

This report presents an overview of the Westinghouse Emergency Core Cooling Systems (ECCS) Evaluation Model which has been developed in accordance with the Atomic Energy Commission's (AEC) regulations 10CFR50.46. Compliance of the Westinghouse Evaluation Model with the requirements of Appendix K of 10CFR50 is demonstrated. Computer code interfaces and the method of analysis for the evaluation of ECCS performance are discussed.

1.0 INTRODUCTION

This report presents an overview (summary) of the Westinghouse (W) ECCS Evaluation Model. Section 2 of this report compares the Acceptance Criteria of 10CFR50.46 to the Interim Acceptance Criteria (IAC) in terms of key requirements. Section 3 compares the features of the Westinghouse Evaluation Model with the requirements of Appendix K of 10CFR50 and demonstrates compliance with Appendix K. Detailed discussion of individual models in the Westinghouse Evaluation Model is presented in referenced Westinghouse reports. Section 4 presents the method of analysis for the large break and small break ECCS analyses. The large break analysis is performed with the following codes: SATAN VI[1], WREFLOOD[5], LOCTA IV[2], COCO[6] or LOTIC[9]. The small break analysis is performed with WFLASH[3] and LOCTA IV. The role of the various computer codes in the ECCS analysis and code interface is discussed in this section. Also key assumptions made in the ECCS analysis are discussed. Appendix A presents the Westinghouse containment back pressure model for ECCS analysis including assumptions employed.

The W Evaluation Model presented in the report is applicable to all W Pressurized Water Reactors (PWR) with Zircaloy cladding with present type ECCS.

2.0 THE ACCEPTANCE CRITERIA

On December 28, 1973, the U.S. Atomic Energy Commision (AEC) issued its opinion concerning "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors and an attached Appendix in which acceptance criteria and required and acceptable features of evaluation models were set forth. These requirements (10CFR50.46 and Appendix K of 10CFR50) which were published in the Federal Register on January 4, 1974 supersede the Interim Acceptance Criteria set forth in the Interim Policy Statement.

The basic intent of the Acceptance Criteria of 10CFR50.46 as compared to the Interim Acceptance Criteria remain unchanged. Certain specific requirements of the criteria have changed, however. Generally these changes have been in the direction of making the analysis more mechanistic, and in some areas more conservative. The research performed on ECCS Systems and the LOCA phenomena have contributed to removing many of the arbitrary requirements available in the IAC.

The basic requirements of the IAC and those of 10CFR50.46 are summarized in Table 1. The changes largely reflect changes in understanding of LOCA phenomena gained by the AEC and the nuclear industry during the time period between the two sets of criteria.

The reduction in the allowable peak clad temperature from 2300°F to 2200°F, in our opinion, reflects additional conservatism rather than additional understanding of LOCA events. The criteria maintain the requirements of a core geometry amenable to cooling, long term core cooling and for less than 1% total Zircaloy cladding inventory to react with steam. These latter criteria are further delineated in Appendix K of 10CFR50 in terms of H_2 production and acceptable chemical kinetics models. The IAC imposed no specific requirement on clad embrittlement. The requirements of 10CFR50.46 place specific limits on local clad oxidation, hydrogen generation, and incorporate features which require calculation of local clad swelling and rupture.

In summary, the 10CFR50.46 requirements place the required ECCS analysis on a more mechanistic basis. This is discussed further in Section 3.0 where features of the Westinghouse Evaluation Model are presented and conformance with the requirements of 10CFR50, Appendix K is demonstrated.

TABLE 1

ACCEPTANCE CRITERIA

IAC	10CFR50.46
PEAK CLAD TEMP. - 2300°F	2200°F
LESS THAN 1% TOTAL CLADDING - WATER/STEAM REACTION IN THE CORE	SAME (ESSENTIALLY)
NO SPECIFIC REQUIREMENT FOR CLAD EMBRITTLEMENT	LESS THAN 17% LOCAL OXIDATION
CORE GEOMETRY AMENABLE TO COOLING	CALCULATED CHANGES IN CORE GEOMETRY SHALL BE SUCH THAT THE CORE REMAINS AMENABLE TO COOLING
LONG TERM COOLING CAPABILITY MAINTENANCE	SAME

Table 1 Comparison of 10CFR50.46 and Interim Acceptance Criteria Requirements

3.0 FEATURES OF THE WESTINGHOUSE EVALUATION MODEL

Appendix K of 10CFR50 presents various required and acceptable features of ECCS Evaluation Models. Westinghouse has developed a model that meets these requirements which is used to demonstrate conformance with the five criteria presented in paragraph (b) of 10CFR50.46 (summarized in Table 1) for ECC systems.

Table 2 compares 10CFR50.46 to IAC in terms of acceptable models. Of the features listed, two types of changes are noted. The first type is exemplified by changes in the break size requirements and in the treatment of decay heat. These changes are typical of added conservatisms in 10CFR50.46. The second type is exemplified by the required calculation of DNB time, hot channel flow and steam-water mixing. Such changes place the LOCA analysis on a more mechanistic but still conservative basis.

Table 3 summarizes the evaluation model features presented in 10CFR50 Appendix K and the provisions in the Westinghouse evaluation model in compliance with those required features. Comparison of the various features in the Westinghouse evaluation model to all 10CFR50 Appendix K requirements is made on a point by point basis and compliance is demonstrated. This comparison is presented in the remainder of this section.

REGULATORY REQUIREMENT:

I. REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODELS

A. SOURCES OF HEAT DURING THE LOCA

For the heat sources listed in paragraphs 1 to 4 below it shall be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for such uncertainties as instrumentation error), with the maximum peaking factor allowed by the technical specifications. A range of power distribution shapes and peaking factors representing power distribution that may occur over the core lifetime shall be studied and the one selected should be that which results in the

TABLE 2

EVALUATION MODELS

IAC	10CFR50.46
$C_D = 1.0$	VARIOUS VALUES OF C_D
GUILLOTINE AND SPLIT BREAKS	DOUBLE-ENDED GUILLOTINE AND VARIOUS SPLITS
DECAY HEAT = ANS + 20%	ANS + 20% + INFINITE LIFE
DNB @ 0.1 SEC.	DNB TIME CALCULATED
100% ACC. BYPASS	ACCUMULATOR BYPASS CALCULATED
90/80 CONTAINMENT PRESSURE	CONTAINMENT PRESSURE CALCULATED
PUMP HEAD DEGRADED	PUMP CHARACTERISTICS DERIVED FROM A DYNAMIC PUMP MODEL
ACCUMULATOR PLUGGING	STEAM WATER MIXING
LOCKED ROTOR PUMP RESISTANCE DURING REFLOOD	SAME
BURST & BLOCKAGE NOT EXPLICITLY DEFINED	BURST & BLOCKAGE CALCULATED
HOT CHANNEL FLOW = 0.8 *AVG.	HOT CHANNEL FLOW CALCULATED
NO MOMENTUM FLUX	MOMENTUM FLUX CALCULATED
COSINE SHAPES	VARIOUS POWER SHAPES

Table 2: Comparison between IAC and 10CFR50.46 of Acceptable Evaluation Model Features

TABLE 3

REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODEL

Appendix K Paragraph	Requirement	Westinghouse Evaluation Model
I.A	Core Power Rating and shapes	Power Level $\geq$1.02 Licensed Power
		Worst Shape and Peaking Factor
I.A.1	Initial Stored Energy	Worst Time in Life. Fuel Densification
I.A.2, .3, .4	Fission and Decay Energy	ANS+20% + Infinite Life
I.A.5	Metal/Water Reaction	Inside/Outside Reaction. Baker-Just Kinetics
I.A.6	Metal Heat Transfer	Lumped Parameter Model
I.A.7	Primary-To-Secondary Heat Transfer	Secondary Side Modeled
I.B	Swelling and Rupture	Clad Deformation Modeled
I.C.1	Break Characteristics	Spectrum of Sizes. Zaloudek & Moody Models
		End of Blowdown Calculated
I.C.2	2-Phase Friction Factors	Harwell
I.C.3	Momentum Flux	Pressure Change Included
I.C.4	Critical Heat Flux	Time to DNB Calculated
I.C.5	Post DNB Heat Transfer	No Rewetting. Westinghouse Transition Boiling Correlation
I.C.6	Pump	2-Phase Homologous Model
I.C.7	Core Flow Distribution	Hot Assembly Flow and Crossflow Calculated
I.D.1	Single Failure	Effects included
I.D.2	Containment Pressure	Conservatively Low Value Calculated
I.D.3	Reflood Rate	Locked Pump Rotor resistance. FLECHT heat transfer
I.D.4	Steam/Water Mixing	Pressure Drop and Condensation Modeled
I.D.5	Refill/Reflood Heat Transfer	FLECHT, Steam and Radiation Heat Transfer used.

most severe calculated consequences, for the spectrum of postulated breaks and single failures analyzed.

FEATURES OF WESTINGHOUSE (W) EVALUATION MODEL:

The analyses reported here are conducted at a power level at least 1.02 times the licensed power level. In addition, an instrument deadband error of $\pm 4°F$ is included on reactor coolant system temperatures.

The power distribution shape resulting in worst calculated consequences is used in ECCS analysis. Normally, the worst calculated consequences will refer to highest calculated clad temperatures. The maximum allowable peaking factors as presented in plant Technical Specifications is used in ECCS analysis.

The results are reported in individual plant Safety Analysis Reports (SAR). Also included in the individual plant Safety Analysis Report is a spectrum of postulated break sizes.

The limiting single failure is discussed in WCAP-8342[7], where the results of several sensitivity studies are reported.

The various features of the W evaluation model are discussed in more detail in the referenced Westinghouse reports describing W computer code models.

REGULATORY REQUIREMENT:

1. The Initial Stored Energy in the Fuel. The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest

calculated cladding temperature (or, optionally, the highest calculated stored energy). To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

FEATURES OF W EVALUATION MODEL:

The initial stored energy used in the W evaluation model is the maximum possible which includes the combination of worst-time-in-life based on fuel burn-up considerations and fuel densification[8] effects. This results in the highest calculated clad temperatures. The initial fuel temperatures (which reflect the initial stored energy) are calculated on a plant by plant basis. Effects of burn-up and temperature are considered in determing UO_2 thermal conductivity and is discussed in detail in Appendix C of the SATAN VI[1] report.

Gap conductance effects are evaluated considering burn-up, fuel densification and expansion, the mixture composition and pressure of the gaps within the fuel rod, initial cold gap dimensions with tolerances and cladding creep. A detailed discussion of the gap conductance model used in the W evaluation model is presented in the LOCTA-IV[2] WCAP. This model is also used in SATAN VI and WFLASH codes.

REGULATORY REQUIREMENT:

2. Fission Heat. Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.

3. <u>Decay of Actinides</u>. The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.

4. <u>Fission Product Decay</u>. The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standard - "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors", Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971). The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.

<u>FEATURES OF W EVALUATION MODEL:</u>

The power decay model in the Westinghouse evaluation for large break ECCS analysis is included in two computer codes, SATAN VI[1] for the blowdown blowdown period and LOCTA IV[2] for the refill-reflood periods of a hypothetical LOCA. The SATAN VI code calculates fission heat based on reactivity and reactor kinetics model. The effects of rod trip and insertion are conservatively neglected in the analysis for large area breaks. Fission product decay model in SATAN VI is at least equal to 1.2 times the values for infinite operating time in the ANS Standard referenced above. Also included in the SATAN VI model are actinide decay and residual fission decay. The contribution due to higher actinide isotopes is negligible, about .06% of core initial power, and therefore not included. All of the above contributions are added to provide a total power decay transient during the blowdown period that is conservative.

LOCTA-IV utilizes the calculated power decay transient during blowdown as one of its inputs for clad temperature calculations. For the refill-reflood periods (large break analysis only), LOCTA IV calculates the power decay transient starting at the end of blowdown or end of SATAN VI problem time. The LOCTA IV calculates fission product decay based on 1.2 times the ANS standard for infinite operating time. Actinide decay is calculated based on a three region core reflecting the end time of a fuel cycle which results in maximum calculated actinide decay rates. The actinide decay rates for this three region core basis are higher (or equal to) than those that would occur in the hot assembly at the time in the fuel cycle yielding the highest calculated fuel temperatures during a LOCA. (This three region core basis is also utilized in SATAN VI and WFLASH). Power from residual fission (after end of blowdown) represents a negligible contribution to total power (less than 0.4% of initial power) and is neglected in LOCTA IV.

In the WFLASH analysis the reactor core power remains at its initial value (prior to the postulated accident) until the reactor is tripped. An appropriate time delay is included in the reactor trip time. Following reactor trip the core decay model is identical to that described above for LOCTA IV.

The fraction of the locally generated gamma energy that is deposited in the fuel (including cladding) is 0.95: This value is justified in the LOCTA IV WCAP.

<u>REGULATORY REQUIREMENT</u>:

5. <u>Metal-Water Reaction Rate</u>. The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L. C., "Studies of Metal Water REactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction", ANL-6548, page 7, May 1962). The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the insdie of the cladding shall also be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less than 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

FEATURES OF W EVALUATION MODEL:

The rate phenomena of metal-water reaction is calculated using the Baker-Just equation, as required. When rupture is calculated to occur, the inside surface of the clad is included in the oxidation reaction model, 1.5 inches each way from the location of the rupture. The reaction is not assumed to be steam limited. This model is utilized in SATAN VI, WFLASH, and LOCTA IV.

REGULATORY REQUIREMENT:

6. Reactor Internals Heat Transfer. Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

FEATURES OF W EVALUATION MODEL:

The SATAN VI and WFLASH computer codes utilize a lumped parameter model for purposes of calculating metal heat transfer from piping, vessel walls, and non-fuel internal hardware. The WREFLOOD[5] code utilizes a lumped parameter approach to calculate metal heat transfer in the downcomer, lower plenum and the steam generator. For reflood, metal heat transfer from the piping and other non-fuel internal hardware has a small effect on the transient because the steam generators superheat the fluid passing through it.

REGULATORY REQUIREMENT:

7. Pressurized Water Reactor Primary-to-Secondary Heat Transfer. Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors).

FEATURES OF W EVALUATION MODEL:

This effect is accounted for, using appropriate heat transfer correlations described in SATAN VI[1], WFLASH[2] and WREFLOOD[5] WCAPs respectively.

REGULATORY REQUIREMENT:

B. SWELLING AND RUPTURE OF THE CLADDING AND FUEL ROD THERMAL PARAMETERS

Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. 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. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.

The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

FEATURES OF W EVALUATION MODEL:

The effects of swelling and rupture of the cladding are conservatively included in the Westinghouse Evaluation Model based on applicable data. The models are described in detail in the LOCTA IV report[2]. Swelling prior to burst is based on the work of Hardy[10] and expansion after burst is based on Westinghouse single rod burst tests[11] (SRBT). If burst is calculated to occur the effects of flow blockage are considered based on Westinghouse multi-rod burst tests[12] (MRBT). Time dependent gap conductance is calculated in LOCTA IV including effects of swelling and rupture.

SATAN VI and WFLASH codes have identical models as compared to LOCTA IV except that swelling prior to burst is not considered. This latter assumption is conservative because it increases SATAN VI (and WFLASH) heat release as compared to LOCTA IV heat release because of increased gap conductance in SATAN VI (and WFLASH) and hence tends to increase peak clad temperatures.

REGULATORY REQUIREMENT:

C. BLOWDOWN PHENOMENA

1. Break Characteristics and Flow

a. In anlayses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

FEATURES OF W EVALUATION MODEL:

Double-ended guillotine breaks are analyzed. The discharge coefficient for the instantaneous double-ended guillotine break is varied over the appropriate range. Longitudinal splits in the largest pipes are also included in the analysis for a range of breaks from the smallest break that the plant charging system can make up to the full cross-sectional pipe area. SATAN VI[1] is used for the blowdown phase of the large area break while WFLASH[3] is used for the small area break. These results are reported in the individual plant SAR.

REGULATORY REQUIREMENT:

b. Discharge Model

For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture", Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965). The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperature calculated by this variation has been achieved.

FEATURES OF W EVALUATION MODEL:

The Moody flow model is employed in SATAN VI[1] and WFLASH[3] for two phase flow discharge. The discharge coefficient is varied over a range from 0.6 to 1.0. One separate calculation is used to determine the peak clad temperature for a single value of a discharge coefficient in the range of 0.6 to 1.0 for a postulated break area. Hence at least three separate calculations are performed to span the discharge coefficient range (0.6 to 1.0) whereby each calculation results in a single value of peak clad temperature. If these calculations show that the maximum clad temperature for the hypothetical accident is found at a lower value of discharge coefficient, the range of discharge coefficients is extended by performing addition calculations. This is performed on a plant by plant basis and results are reported in the individual Safety Analysis Reports.

REGULATORY REQUIREMENT:

c. End of Blowdown

(Applies Only to Pressurized Water Reactors). For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the "end of bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective.

The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining "end of bypass" include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

FEATURES OF W EVALUATION MODEL:

For large break analysis the end-of-bypass time is calculated using the drift-flux flow model incorporated in SATAN VI[1]. This calculation determines the time at which down flow is predicted in the downcomer. Prior to this time, 100% of the injected accumulator water is assumed to be bypassed. This method is the first of the two methods described above in the regulatory requirement.

For small break analysis the reactor core remains at least partially covered during the complete LOCA transient because of the relatively small magnitude of break flow compared to flow pumped to the core inlet from the coasting down intact loop pumps. Hence an "end of bypass" calculation is not needed for small break analysis. In addition the entrainment mechanisms in the reactor downcomer are negligible by the time accumulator injection is iniated.

REGULATORY REQUIREMENT:

d. Noding Near the Break and the ECCS Injection Points

The noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

FEATURES OF W EVALUATION MODEL:

The proper noding in the regions near the postulated rupture location and near the injection locations were determined from sensitivity studies. These are reported in WCAP-8342 [7].

REGULATORY REQUIREMENT:

2. Frictional Pressure Drops. The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation (Baroczy, C. J., "A Systematic Correlation for Two-Phase Pressure Drop", Chem. Enging. Prog. Symp. Series, No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J. R. S., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Int. J. of Heat & Mass Transfer, 7, 709-724, 1964) for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R. C. Nelson, D. B., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Transactions of ASME, 695-702, 1948) for pressures lower than 250 psia is acceptable as a basis for calculating realistic two-phase friction multipliers.

FEATURES OF W EVALUATION MODEL: The frictional pressure drop calculation for ECCS hydraulic analysis include a realistic variation of friction factor as a function of Reynolds number and realistic two-phase friction multipliers adequately verified by comparison with data. The SATAN VI[1], WFLASH[3], and WREFLOOD[5] codes utilize friction factor correlations that agree with the Moody friction factor chart in the Crane[13] manual. SATAN VI, WFLASH, and WREFLOOD uses a two-phase friction multiplier correlation developed by Heat Transfer and Fluid Flow Service (HTFS) at Harwell[4]. This correlation has been adequately verified by a wide range of experimental data over a large range of mass velocities and is described in Reference 4.

REGULATORY REQUIREMENT:

3. Momentum Equation. The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

FEATURES OF W EVALUATION MODEL:

For large break blowdown hydraulic analysis, calculated by SATAN VI[1], all seven terms specified above are included in the momentum conservation equation. Detailed discussion is presented in the SATAN VI[1] report.

For small break blowdown hydraulic analysis, calculated by WFLASH, terms (2), (3) and (4) are relatively small in magnitude compared to the elevation pressure drop term in particular and therefore are neglected. The WFLASH[3] report presents a comparison of code predictions to applicable experimental data and noted agreement is good.

REGULATORY REQUIREMENT:

4. Critical Heat Flux

a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlation acceptable for use in LOCA transients include, but are not limited to, the following:

(1) W 3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, 241-248, 1967.

(2) B&W-2. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Two-Phase Flow and Heat Transfer in Rod Bundles, ASME, New York, 1969.

(3) Hench-Levy. J. M. Healzer, J. E. Hench, E. Janssen, S. Levy "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, GE Company Private report, July 1966.

(4) Macbeth. R. V. Macbeth, "An Appraisal of Forced Convection Burnout Data," Proceedings of the Institute of Mechanical Engineers, 1965-1966.

(5) Barnett. P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R 463, 1966.

(6) Hughes. E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia,: IN-1412, Idaho Nuclear Corporation, July 1970.

c. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

d. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) GE transient CHF. B. C. Slifer, J. E. Hench, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, General Electric Company, Equation C-32, April 1971.

e. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

FEATURES OF W EVALUATION MODEL:

In SATAN VI and LOCTA IV the Macbeth correlation is used to calculate the effects of DNB. After DNB is calculated to occur, no return to nucleate boiling is permitted until the reflood portion of the transient yields a quenched condition.

REGULATORY REQUIREMENT:

5. Post-CHF Heat Transfer Correlations.

a. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

b. The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime," AECL-3281, revised December 1969), the Dougall-Rohsenow flow film boiling correlation (R. S. Dougall and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities,: MIT Report Number 9079-26, Cambridge, Massachusetts, September 1963), and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," U.S.A.E.C. Docket RM-50-1, page 25-1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes. In addition the transition boiling correlation of McDonough, Milich, and King (J. B. McDonough, W. Milich, E. C. King, "Partial Film Boiling with Water at 2000 psig in a Round Vertical Tube," MSA Research Corp., Technical Report 62 (NP-6976), (1958) is suitable for use between nucleate and film boiling. Use of all these correlations shall be restricted as follows:

(1) The Goeneveld correlation shall not be used in the region near its low-pressure singularity,

(2) the first term (nucleate) of the Westinghouse correlation and entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300°F,

(3) transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300°F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

FEATURES OF W EVALUATION MODEL:

For post DNB heat transfer the Westinghouse correlation of steady-state transition boiling is employed. The correlation presented in the Westinghouse Redirect/Rebuttal Testimony has been modified slightly in a conservative manner as described in the LOCTA IV[2] report. Its use is subject to the restrictions mentioned above.

REGULATORY REQUIREMENT:

6. Pump Modeling. The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump heat may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent.

FEATURES OF W EVALUATION MODEL:

The reactor coolant pump model is based on two-phase homologous treatment. The model is verified by applicable Aerojet Nuclear Company (ANC) two-phase data. Further discussion is found in the SATAN VI report[1]. Identical models are incorporated in WFLASH and SATAN-VI.

REGULATORY REQUIREMENT:

7. Core Flow Distribution During Blowdown.
(Applies only to pressurized water reactors.)

a. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

b. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

FEATURES OF W EVALUATION MODEL:

For large break analysis, the flow rate through the hot assembly is calculated using the SATAN-VI code. For the W ECCS evaluation model, the reactor core in SATAN-VI is modelled utilizing two parallel fuel channels. One channel represents the hot assembly in the core and the second channel represents the remaining channels (or average channel) in the core. Crossflow is calculated between the hot assembly and the average assembly (between the two parallel fuel channels). Flow blockage is also calculated in SATAN-VI when it is calculated to occur. The SATAN-VI calculated hydraulic transient (pressure, flow, density etc) in the hot assembly is transferred directly as input to LOCTA-IV for the heatup calculation in the hot assembly. LOCTA-IV code has a swelling, burst and blockage model to determine the flow around the hot rod in the hot assembly and hence a conservative peak clad temperature. The SATAN-VI average flow hydraulic transient can be input into LOCTA-IV for the heatup calculation in the remaining core. This latter procedure is used to determine the core-wide metal water generation and hence total H_2 generation.

The method for determining the core fluid enthalpy (large break analysis) to be used as input into LOCTA-IV is explained later in Section 4.0 of this report.

For small break analysis, using WFLASH, only the average core is modelled because hot assembly flow is larger than average assembly due to larger decay heat. The WFLASH calculated hydraulic core transient is transferred directly as input to LOCTA-IV for the heatup calculation in the hot assembly. LOCTA-IV then performs a fuel rod swelling and burst calculation.

REGULATORY REQUIREMENT:

D. POST-BLOWDOWN PHENOMENA; HEAT REMOVAL BY THE ECCS.

1. Single Failure Criterion. An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

2. Containment Pressure. The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not excced a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

FEATURES OF W EVALUATION MODEL:

The two requirements above are satisfied in a consistent manner. In order to assure that all installed pressure-reducing systems and processes are in operation, the availability of all diesel powered emergency generating facilities are used. Hence, the customary ECCS single failure of one diesel cannot be postulated. The resulting single failure has been determined to be the failure of one residual heat removal pump. The analyses which were performed to determine the single failure are reported in WCAP 8342[7].

The containment pressure is calculated conservatively low by using appropriate values of input to the COCO code[6]. Appendix A to this report presents the W ECCS Containment Backpressure model and describes the various inputs for COCO that are selected in a conservative manner to provide a conservatively low containment backpressure for ECCS analysis. Figure 1 presents a comparison of the calculated containment pressure in a typical PWR plant for ECCS analysis, using the model presented in Appendix A, compared to pressure transients calculated specifically for containment pressure integrity analysis and for IAC analysis. WCAP-8342[7] in part reports the results of studies performed to determine the sensitivity of the calculated containment pressure to COCO input data.

REGULATORY REQUIREMENT:

3. Calculation of Reflood Rate for Pressurized Water Reactors. The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise the pump rotor shall be assumed to be running free. The ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971; "PWR Full Length Emergency Cooling Heat

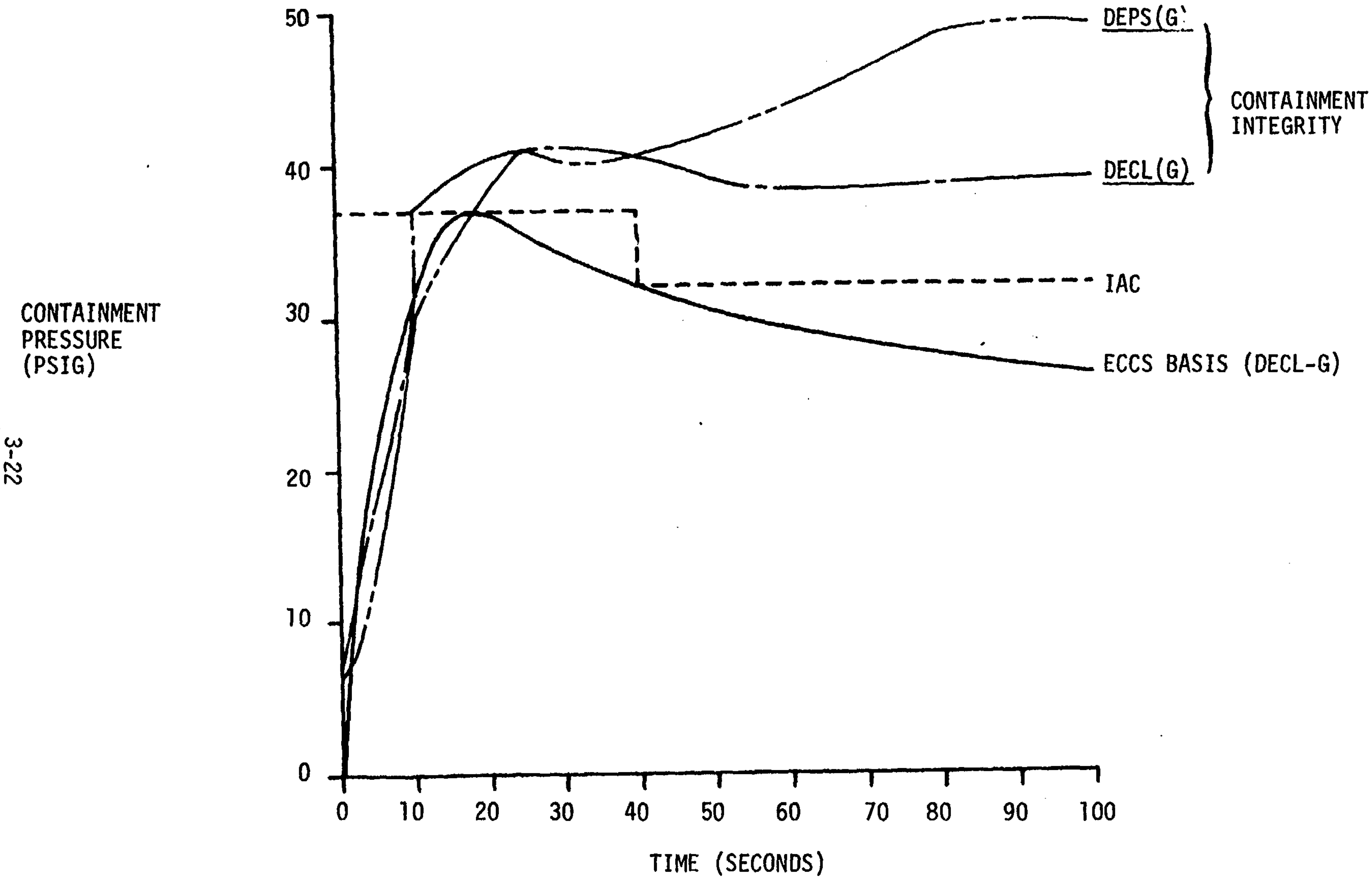

FIGURE 1 - COMPARISON OF CALCULATED CONTAINMENT PRESSURE TRANSIENT FOR ECCS BACKPRESSURE vs. CONTAINMENT INTEGRITY DESIGN

Transfer (FLECHT) Group I test Report," Westinghouse Report WCAP-7435, January 1970; "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972).

The effects on reflooding rate of the compressed gas in the accumulator which is discharged following accumulator water discharge shall also be taken into account.

FEATURES OF W EVALUATION MODEL:

The refill portion of the LOCA analysis is performed, in part by SATAN VI[1] and, in part, by WREFLOOD[5]. This is discussed in Section 4.0 of this report. The reflood portion of the LOCA transient is calculated using the WREFLOOD code[5]. This calculation conservatively assumes that the reactor coolant pumps have locked rotor resistance. This was determined by analyses reported in WCAP 8342[7]. The entrainment correlation was determined in accordance with applicable FLECHT data. The effects of compressed gas in the accumulators are discussed in WCAP 8342[7].

REGULATORY REQUIREMENT:

4. Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors. The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption.

FEATURES OF W EVALUATION MODEL:

The effects of steam/water mixing are included in the LOCA analysis. Thermal-hydraulic interaction between steam flow and water injection in the unbroken loops are included during refill and reflood periods of LOCA. The model is described in the WREFLOOD[5] code and has been verified with available data.

REGULATORY REQUIREMENT:

5. Refill and Reflood Heat Transfer for Pressurized Water Reactors.

For reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results ("PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971). The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied; presently available FLECHT heat transfer correlations ("PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972) are not acceptable. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.

During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer.

FEATURES OF W EVALUATION MODEL:

For reflood rates of 1"/sec or greater, reflood heat transfer coefficients are based on a modification of the previous FLECHT correlation (WCAP-7931) which removes the concerns cited by the AEC. The correlation used in the W evaluation model is presented in the LOCTA-IV[2] WCAP.

For reflood rates less than 1"/sec convection heat transfer coefficients are based only on steam cooling. Swelling and rupture effects are included by computing flow redistribution in WREFLOOD[5]. The steam flow is calculated in WREFLOOD including latter effects and is input into LOCTA-IV to calculate a steam-cooling heat transfer coefficient.

REGULATORY REQUIREMENT:

6. Convective Heat Transfer Coefficients for Boiling Water Reactor Rods Under Spray Cooling. Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:

   a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

   b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 Btu-hr^{-1}-ft^{-2}-$°F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

   c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu-hr^{-1} -ft^{-2}- $°F^{-1}$ shall be applied to all fuel rods.

FEATURES OF W EVALUATION MODEL:

This requirement is not applicable to a Pressurized Water Reactors PWR.

REGULATORY REQUIREMENT:

7. The Boiling Water Reactor Channel Box Under Spray Cooling. Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7 x 7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

   a. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

   b. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\text{-}{}^{\circ}F^{-1}$ shall be applied to both sides of the channel box.

   c. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis ("Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Company Report NEDO-10329, April 1971).

FEATURES OF W EVALUATION MODEL:

This requirement is not applicable to the Pressurized Water Reactors PWR.

3.1 SUMMARY

The above discussions have presented a detailed comparison of the Westinghouse evaluation model features with the requirements of Appendix K to 10CFR50 and demonstrates compliance with these requirements. The Westinghouse evaluation model is used to analyze ECCS performance in Westinghouse Pressurized Water Reactor (PWR) plants (with Zircaloy cladding) and to demonstrate conformance with the requirements presented in paragraph (b) of 10CFR50.46 (summarized in Table 1). A complete description of the method of analysis (using the Westinghouse evaluation model) used to determine compliance with the criteria in 10CFR50.46 for any particular W plant with Zircaloy cladding is presented in the next Section (4.0) of this report.

4.0 METHOD OF ANALYSIS

This section describes the method of analysis whereby a Westinghouse PWR plant's ECCS performance is evaluated with respect to the five criteria presented in paragraph (b) of 10CFR50.46. These criteria are listed below.

1. Peak Cladding Temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. Maximum Cladding Oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph, total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

3. Maximum Hydrogen Generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. Long-Term Cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The method of analysis for evaluating each of these criteria is described below.

4.1 CALCULATION OF PEAK CLAD TEMPERATURE

The calculation of peak clad temperature is performed by modelling the hottest fuel assembly (from the reactor core) in the LOCTA-IV code. The hot fuel assembly is subdivided into three regions: 1) the hottest rod, 2) adjacent rod to the hottest rod, and 3) the average fuel channel in the hot assembly. The peak clad temperature occurs on the hottest rod. The LOCTA-IV code is used in conjunction with other computer codes which determine necessary thermal-hydraulic boundary conditions for the LOCTA-IV fuel rod heatup analysis.

The method of analysis to determine peak clad temperature is divided into two types of analysis: 1) large break LOCA, and 2) small break LOCA. The method of analysis for large and small break LOCA is compared and described below.

The large break LOCA transient may be conveniently divided into three time periods: blowdown, refill and reflood. Also there are three distinct physical parts of the transient to be analyzed for each time period: thermal-hydraulic transient in the Reactor Coolant System (RCS), pressure and temperature within the containment and fuel and clad temperature within the hottest fuel rod. These considerations lead to a system of computer models designed to treat the LOCA transient. The LOCTA-IV code is used throughout the entire transient to compute fuel and clad temperatures in the hottest fuel rod. Likewise the COCO code is used for the complete containment pressure history for dry containments. The LOTIC code is used for ice containment pressure history. The SATAN-VI code is employed for the thermal-hydraulic transient during blowdown while the WREFLOOD code computes this transient during refill and reflood. See Figure 2.

For small breaks, the reactor does not empty and thus the core is recovered during blowdown. For these cases the WFLASH code is employed for the thermal-hydraulic transient while the LOCTA-IV code is again used for

PHYSICAL PART / BREAK SIZE	THERMAL-HYDRAULIC TRANSIENT IN REACTOR COOLANT SYSTEM	PRESSURE AND TEMPERATURE IN CONTAINMENT	FUEL AND CLAD TEMPERATURE IN HOTTEST ROD
Large	SATAN and WREFLOOD	COCO or LOTIC	LOCTA IV
Small	WFLASH	Not Required	LOCTA IV

Figure 2. Role of Westinghouse Computer Codes in Loss-of-Coolant Analysis

the clad temperature. Because the highest clad temperature occurs during blowdown, when the break flow is choked (sonic), containment pressure has no influence on ECCS performance and thus need not be considered.

4.1.1 LARGE BREAK ANALYSIS

The SATAN-VI code is the first used in the series of calculations which ultimately result in peak clad temperature. Inputs to this model include reactor power and initial conditions, system geometry and hydraulic data, reactor coolant pump characteristic curves, fuel kinetics data, fuel rod conditions, safety injection (SI) performance, and setpoints for reactor trip and safety injection. Containment pressure is input also in the determination of break flow for the period of non-critical flow at the end of blowdown. The fluid model within the SATAN-VI code solves the conservation equation of mass, momentum and energy and the equations of state to determine fluid pressure, enthalpy, density and mass flow rate as a function of time for each SATAN-VI element (control volume).

Figure 3 presents the SATAN-VI 46 element model that is used in the Westinghouse evaluation model. This model was determined based on sensitivity studies (Reference 7) to SATAN-VI noding in the core, steam generator, reactor vessel, and break.

Other models within the SATAN-VI code simulate quantities of interest such as average and hot assembly core conditions, reactor coolant pump performance, plant power transient, ECCS injection, break flow rate and reactor trip and safety injection signal. These models are described fully in Reference 1.

For the purpose of ECCS analysis, items of interest computed during blowdown include fluid conditions entering and within the reactor core - particularly the hot assembly - and the mass and energy flow to the

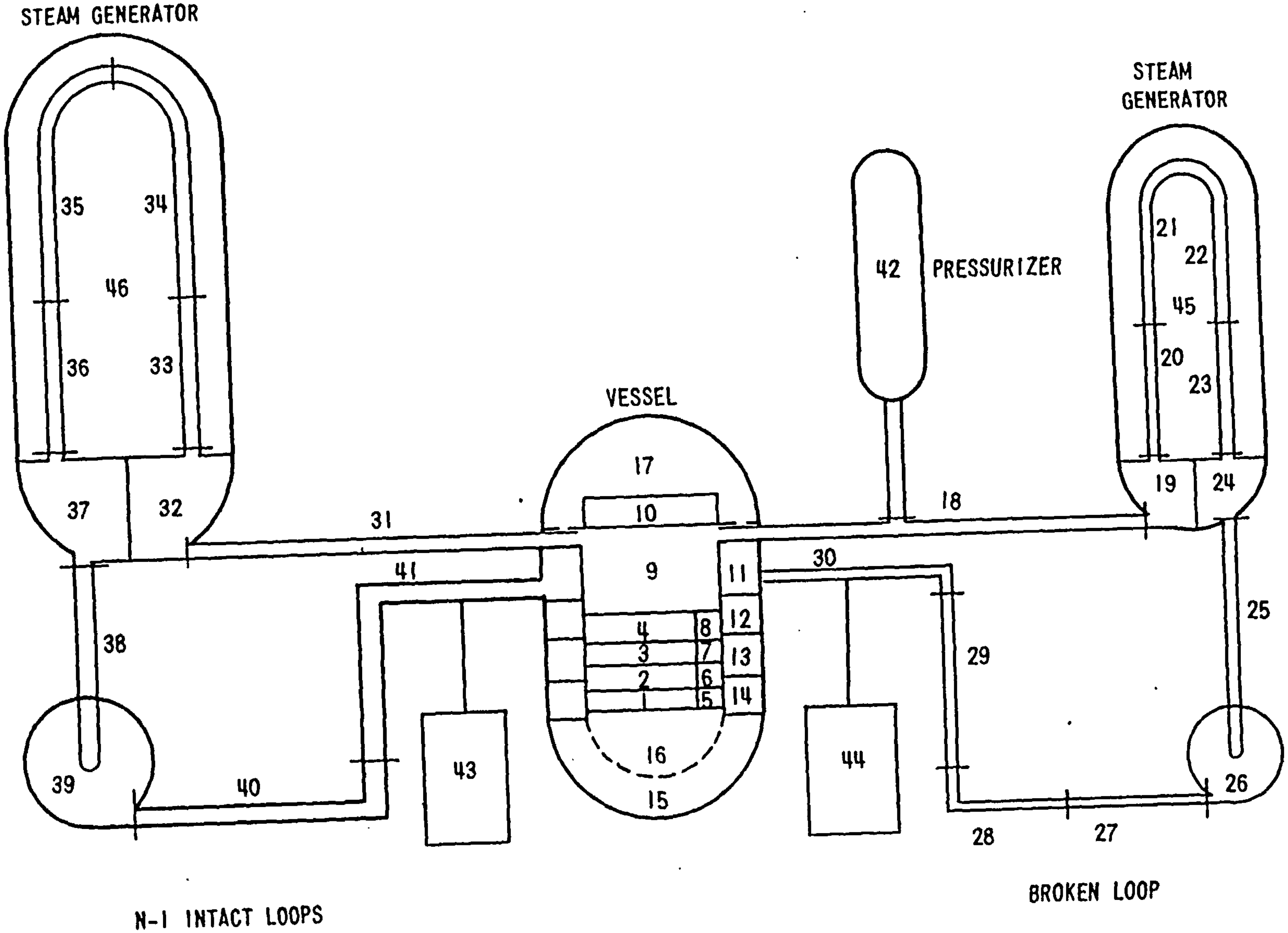

FIGURE 3 - SATAN-VI MODEL FOR W PWR (46 ELEMENT)

containment. At the end of the SATAN VI calculation, it is important to know the RCS and accumulator inventories in order to compute the time required to recover the bottom of the core.

The SATAN VI code is used from the initiation of the accident to the time designated as "End-of-SATAN". This time is defined as the earliest of either downflow in the downcomer region greater than ECCS flow or zero break flow on the vessel side or bottom of core recovered by ECCS water. After this time, the SATAN VI code is no longer used and the WREFLOOD code is applicable. This is shown in Figure 4.

Prior to the end of SATAN, an "end of bypass time" is determined as the first time when ECCS water begins to go down the downcomer. Refill is considered to begin at end of bypass. The water flow down the downcomer is determined from the total flow with the drift flux model as described in the SATAN VI WCAP[1]. In particular, liquid flow may be down while steam flow or total flow is up.

The purpose of the "end of bypass" time is to provide assurance for Appendix K analyses that all water injected up to that time shall not be included in the calculated reactor vessel inventory at the end of blowdown. Accordingly, the SATAN VI code includes an accumulator (and SI) bypass model which performs an inventory calculation to determine how much accumulator water must be bypassed according to the Appendix K rule and how much water is actually bypassed in the SATAN VI calculation. Any deficit in accumulator bypass is subtracted from the vessel inventory at the time of the switch from the SATAN VI blowdown code to the WREFLOOD reflooding code. The SATAN VI calculation is not affected by the bypass inventory calculation. This model is described in detail in Ref. 1.

For the refill calculation, WREFLOOD initializes the lower plenum inventory for reflood by determining the available amount of liquid that exists based on "end of SATAN" condition and prevailing containment pressure and subtracts from that the required bypass deficit per the Appendix K rule. Liquid in the intact cold leg pipes and inlet nozzles, broken

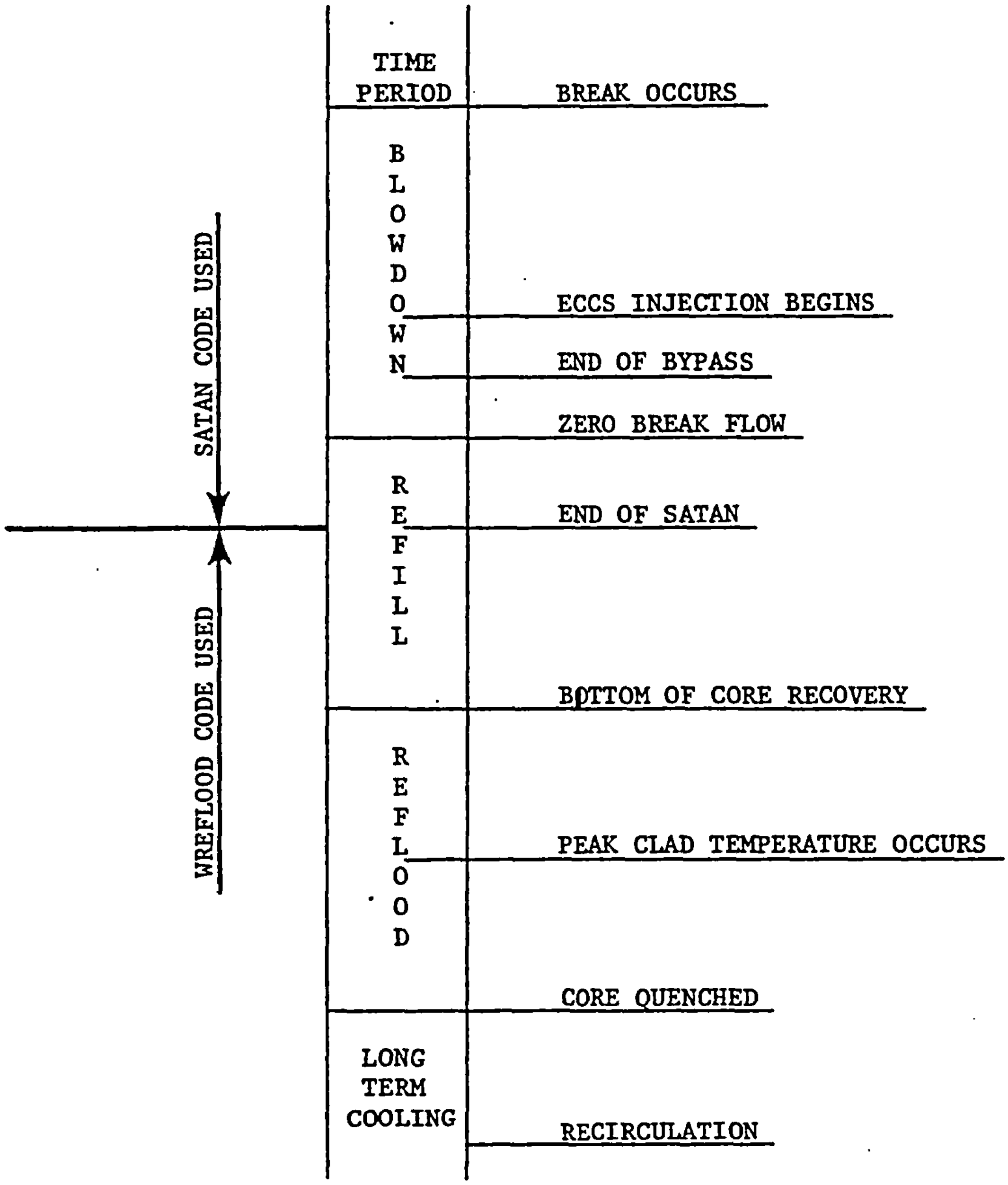

Figure 4. Sequence of Events for Large Break Loss-of-Coolant Analysis

cold leg nozzle, downcomer and lower plenum is considered available for refill. Negative inventory is disallowed. A fluid transient time from the ECCS injection point to the lower plenum is included. This inventory is increased at a rate determined by the ECCS flow rates until bottom of the core is recovered. At that time reflood begins.

Inputs to the reflood calculation in WREFLOOD include system geometry and hydraulic data, reactor coolant pump characteristic curves, ECCS performance data, core heat flow during reflood as well as steam generator and accumulator conditions at the beginning of the WREFLOOD calculation. The latter two quantities are determined directly from the "End of SATAN" conditions. Reactor coolant pump speed may also be determined directly from SATAN VI. However for Appendix K analyses, a locked rotor (zero speed) pump resistance is used. A final quantity determined directly from SATAN VI is the bypass deficit discussed above.

The primary conservation equation in WREFLOOD is the momentum equation. This equation determines local pressure changes around the reactor coolant loop due to spatial acceleration (area change and density change) and viscous losses (form and function). Mass velocity is considered uniform except at mixing or separation points.

Enthalpy changes occur due to heating of the water in the lower plenum and downcomer, addition of stored energy and residual heat in the reactor core, addition of heat in the steam generator and mixing at the injection point. Other models within WREFLOOD simulate core heat release, reactor coolant pump performance, residual heat, ECCS injection performance (accumulators and pumps) and break flow. These models are described fully in Ref. 2.

The WREFLOOD code consists of a fixed vessel model and two geometry loops. Figure 5 presents a schematic of the WREFLOOD Model used in the Westinghouse evaluation model.

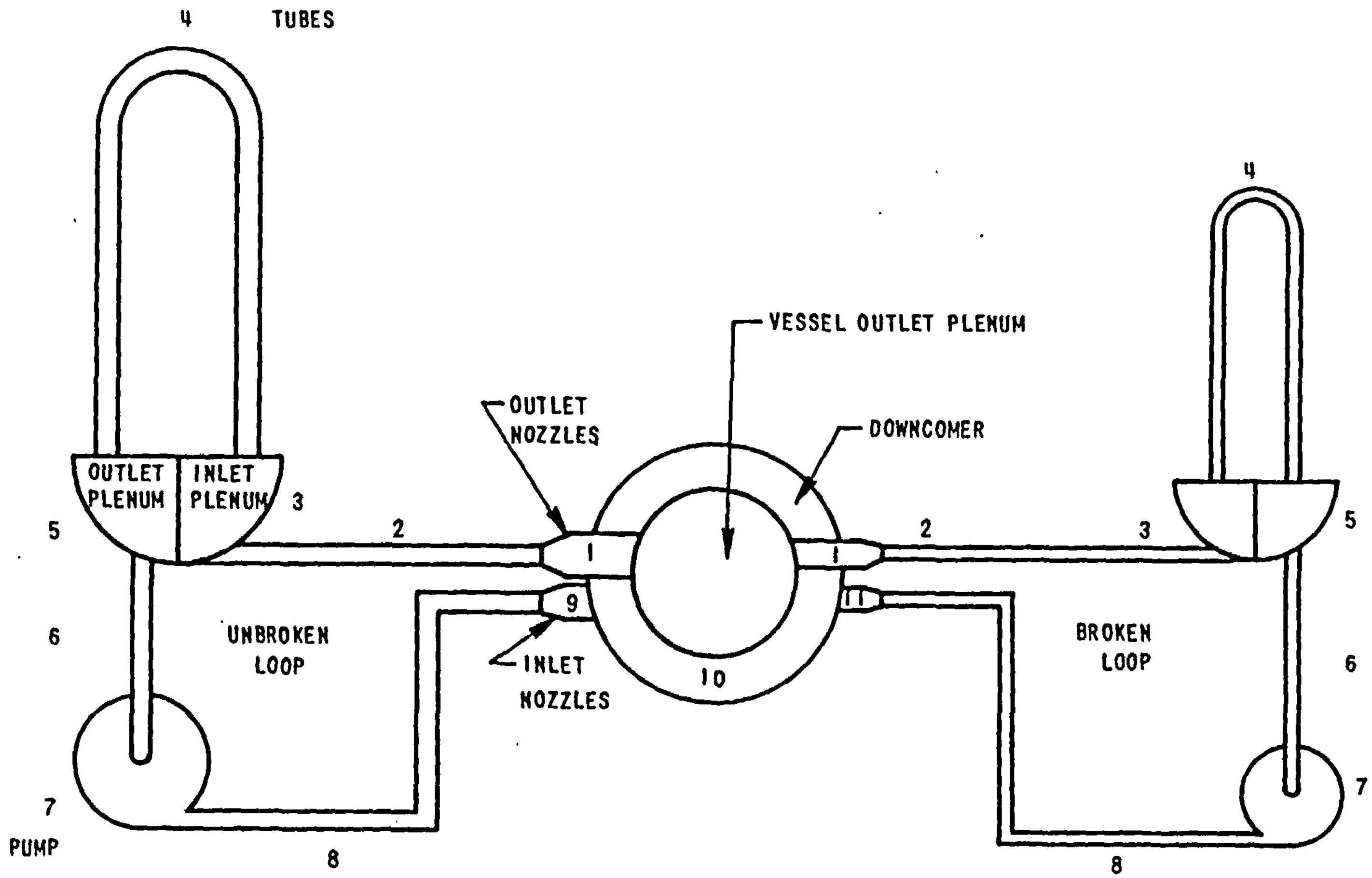

FIGURE 5 - SCHEMATIC OF WREFLOOD MODEL FOR W PWR

A key purpose of the WREFLOOD code in ECCS application is to determine the core flooding rate. This is the rate at which liquid enters the bottom of the core. A portion of this liquid is vaporized in the core and this vapor can entrain additional liquid as it exits the top of the core. The remainder of the liquid accumulates within the core, and the water level is increased. The fluid which exits the top of the core must be vented through the coolant loops and reactor coolant pump. The driving head for venting is established by the downcomer water level and the core water level. The mass flow rate to be vented is set by the flooding rate and the carryover rate fraction. The volume of steam to be vented depends on the local pressure. Finally the local pressure depends on the containment pressure. In accordance with the Appendix K requirements, a conservatively low containment pressure must be used for ECCS evaluation.

The containment pressure may be provided for use in WREFLOOD via two methods. A constant back pressure may be specified or a simulataneous calculation of containment pressure can be performed using the COCO code. In either case, the value is insured to be conservatively low via appropriate assumptions in the containment pressure analysis.

The linking of WREFLOOD and COCO is accomplished without sacrifice of either accuracy or flexibility. The codes are linked intimately, i.e. both codes are executed simultaneously; problem times for the two codes are locked into phase, and the relevant interface parameters are continuously exchanged for the current problem time. From the user's standpoint, each of the codes is practically unchanged from the form in which it has been previously used. There has been no degradation of the capability or flexibility of either code. Thus, the two codes retain the same mathematical form that they had when they were used separately for LOCA analysis. The parameters exchanged between the codes are indicated in Figure 6, which presents interface data.

For the purpose of ECCS analysis, items of interest computed by the WREFLOOD code, include the time at which the bottom of the core is

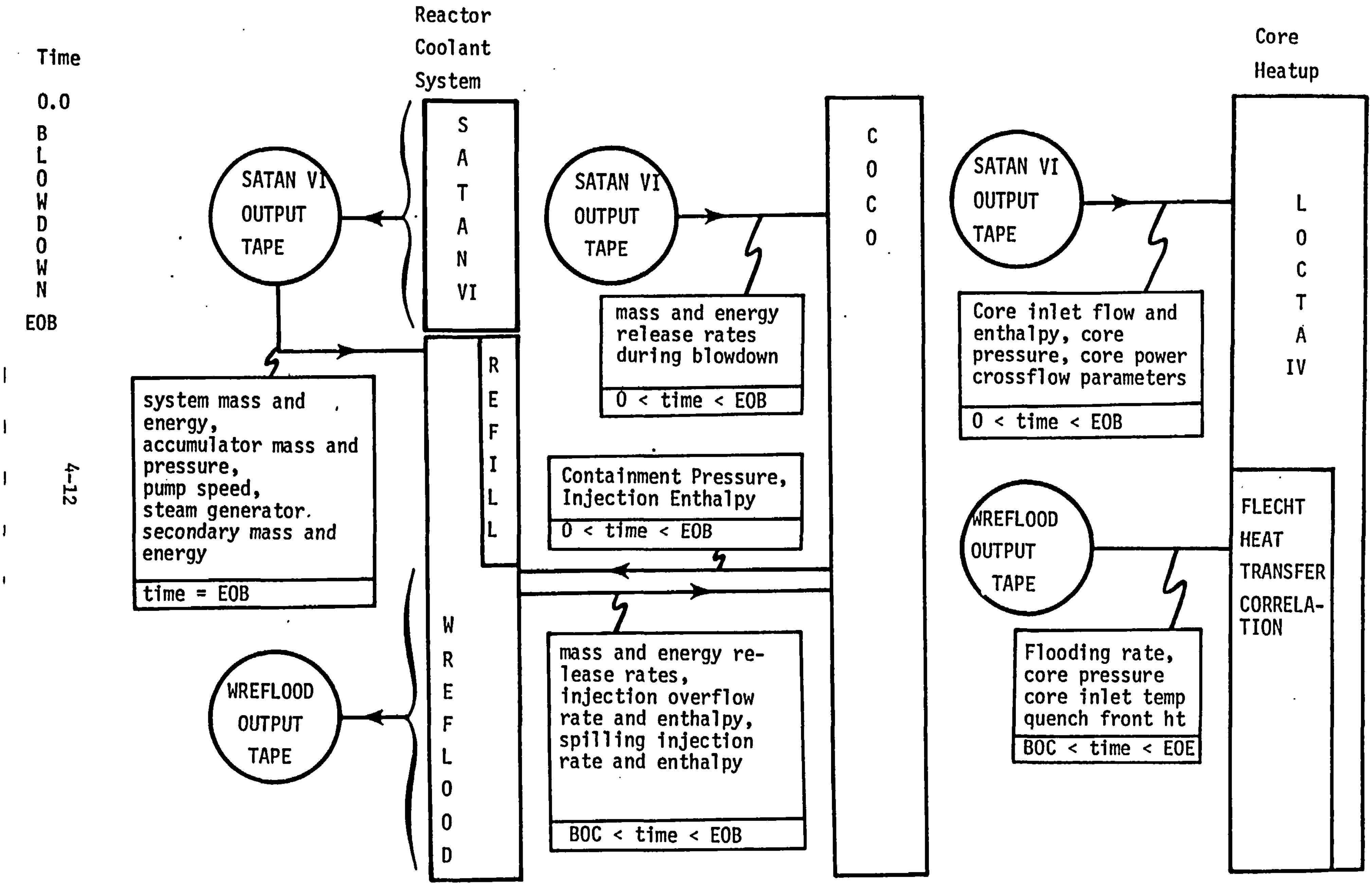

FIGURE 6 - CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL

recovered, fluid conditions entering the core - particularly the hot assembly and the mass and energy flow to the containment.

The WREFLOOD code is used from the end of SATAN until the clad temperature has peaked. The remainder of the transient indicates a monotonic reduction in temperature.

The Westinghouse containment pressure transient code, COCO, has been used extensively for containment pressure-temperature design analysis. The application of COCO to the problem of ECCS back pressure analysis is somewhate novel, but requires no major changes in the mathematical formulation of the various models in the code.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into systems. The first system consists of the air-steam phase, while the second is the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam, and subcooled or saturated water. Switching between states is handled automatically by the code. COCO provides analytical models for various containment cooling systems including containment spray, fan coolers, and structural heat sinks.

The overall containment model including containment free volume, spray and fan cooler heat removal capabilities and containment structural heat sinks will be determined from the individual plant parameters. Suitable conservatism will be applied to each of these parameters on a case by case basis. All containment cooling systems are assumed operable, and start times for individual components will be chosen from the individual plant parameters, on a basis which is consistent with the start time chosen for pumped safety injection. Heat transfer coefficients for structural heat sinks are based on the work of Tagami, with suitable conservatism applied.

The initial containment conditions, pressure, temperature, and relative humidity are the minimum during normal operation. The temperature of water in the refueling water storage tank, and the temperature of service water will be picked on a consistent basis which provides the greatest conservatism for ECCS analysis.

Appendix A describes the Westinghouse containment backpressure model for ECCS evaluation.

Mass and energy discharge rates during the blowdown portion of the LOCA are available from the SATAN VI output tape generated for the ECCS blowdown. Thus, blowdown as predicted by the SATAN VI code directly provides the initialization of the containment pressure at the start of the WREFLOOD - COCO calculation. During the reflooding portion of the transient, mass and energy release rates are transferred to COCO from WREFLOOD on an interactive basis.

The LOCTA IV code is used to obtain peak clad temperature in the hottest rod. Inputs to this code include initial conditions along the fuel rod including the clad, gap, and pellet. Of particular importance in ECCS analyses are pellet initial temperature and linear power. The pellet initial temperature is chosen at the worst possible time in life. It includes fuel densification and uncertainties per the Westinghouse densification model[8]. The appropriate linear power to be considered in input for ECCS analysis is the maximum value obtained from operation of the plant within the technical specifications or alternately a parametric study using the ECCS evaluation model may be performed which determine the maximum value of linear power which meets the 10CFR50.46 acceptance criteria. The value so determined would constitute an "ECCS limit" for the technical specifications. A choice between the specification methods above would be based on specific plant design and operation.

The hot fuel assembly is divided into three regions. The hot rod is analyzed in order determine peak clad temperature. A rod adjacent to the hot rod is analyzed to determine the amount of radiation heat transfer from the hot rod to non-burst adjacent rods. The average rod in the hot assembly is analyzed in order that the heat release may be optionally used to determine fluid properties in the hot assembly during blowdown or reflood.

For the determination of hot assembly fluid properties two methods are incorporated in LOCTA IV. In the first method, fluid properties in the hot assembly are determined from the hot assembly average rod heat release to the fluid. The power in the hot assembly is determined by the assembly peaking factor and the number of fuel rods in the assembly. The fluid properties at the inlet of the hot assembly are taken from SATAN VI[1] output.

Information from the SATAN VI code supplied to LOCTA IV includes hot assembly inlet flows and enthalpies, pressure and depressurization rates, quantities required for the calculation of crossflow, and the power generated in the fuel during blowdown. The energy and continuity equations are solved at each node for the fluid as it moves up or down the hot assembly, using as boundary conditions SATAN VI supplied values of flowrate and enthalpy. The following effects are taken into account in the fluid energy equations:

1. energy changes due to heat release from the hot assembly,
2. energy changes due to depressurization,
3. energy changes due to changes in density.

Crossflow due to blockage is calculated from quantities supplied by SATAN VI and is accounted for in LOCTA IV as a source term in the continuity equation. The effect of crossflow is thus to add or subtract mass from the hot assembly.

In the second method, SATAN VI hot assembly fluid properties are used directly as LOCTA IV hot assembly fluid properties. Since the axial noding of LOCTA IV is finer than that of SATAN VI, the mass velocity, pressure and enthalpy at each LOCTA IV node are linearly interpolated both in time and space from SATAN VI information.

Flow rates are defined at each flow path in SATAN VI. Mass velocity in each LOCTA IV node are calculated by interpolating this flow rate.

For the pressure, they are calculated in SATAN VI at the center of each SATAN VI element (control volume). By interpolation/extrapolation, the pressure at each elevation is calculated.

Enthalpies defined in SATAN VI elements are considered to be SATAN VI element enthalpies. From the SATAN VI enthalpy information at each SATAN VI point, the enthalpy in each LOCTA IV node is interpolated.

The hot assembly heat release is used in LOCTA IV when the core fluid conditions determined in SATAN VI and WREFLOOD (OR WFLASH) are not appropriate for the hot rod clad temperature calculation. This will occur in SATAN VI (or WFLASH) when the hot assembly is not simulated. It can occur in SATAN VI when the SATAN VI hot assembly power is less than the LOCTA IV hot assembly power. It can occur in WREFLOOD when the flooding rate is less than 1"/sec and a steam cooling calculation is necessary.

The hot rod adjacent rod and hot assembly are modeled with axial nodes placed at intervals along the rods. Additional nodes are placed at 3 inch intervals in the vicinity of the highest power spot. These additional nodes are used to model the burst region.

The fuel rod thermal model solves the transient heat conduction equation for the fuel and cladding. The following effects are taken into account:

1. Power generation and flux depression effects within the fuel.

2. Heat generation within the clad.

3. Variations in fuel and clad thermal properties due to temperature changes and zirconium oxide buildup.

Temperature nodes in the fuel and temperature nodes in the clad are used to calculate the radial temperature distribution within the fuel rod. Axial conduction in the clad is included in the calculations using the approximation that the axial temperature gradient is that existing at the start of each time step. The fuel rod noding model (axial and radial) is presented in WCAP 8342[7] and was determined based on sensitivity studies.

The power assumed to exist in the core at the time of the accident is at least 1.02 times the licensed power level of the plant being analyzed. As mentioned previously, the hot rod peaking factor is the maximum allowed by technical specifications. The axial power distribution assumed to exist in the core at the time of the accident is chosen so as to maximize calculated peak clad temperatures. Power distributions skewed to the top and bottom of the core, as well as the standard cosine power shape, are analyzed for the worst break size.

The burnup which yields the highest calculated stored energy is selected to determine initial values for fuel gap size, gas composition, and gap pressure using standard fuel design methods. These quantities are input to LOCTA IV which then calculates the corresponding gap conductance and fuel temperature. Additional temperature uncertainties and effects due to densification (in accordance with the Westinghouse Densification Model)[8] are added by increasing the gap width to increase the fuel average temperature. During blowdown prior to burst the gap conductance is calculated as a function of cladding and fuel thermal expansion, elastic deflection due to internal stresses, and temperature and pressure of the gases within the gap. Plastic swelling prior to burst is also included.

Heat generation due to zirconium-water reaction and changes in cladding properties due to oxide buildup are calculated on the outside of the hot adjacent, and average rods using the Baker-Just rate equation. If and when bursting has been calculated to occur, additional metal water reaction and oxide buildup is calculated on the inside of the cladding within a region extending 1.5 inches on either side of the burst point. The rod-to-steam heat transfer regimes considered in LOCTA IV are:

Forced Convection to water
Nucleate Boiling
Transition Boiling
Forced Convection to Steam
Radiation to Steam
Reflood Heat Transfer (FLECHT)

In addition rod-to-rod radiation is considered; this is significant primarily in the burst region.

A detailed description of the heat transfer model is presented in the LOCTA IV WCAP.

A summary of the code interfaces (SATAN VI, WREFLOOD, COCO and LOCTA IV) are presented in Figure 6.

4.1.2 SMALL BREAK ANALYSIS

For small break analysis, the peak clad temperature occurs during blowdown. Hence many of the feature used for large breaks are not required.

The WFLASH code is similar to SATAN VI in terms of input data and models. One difference is that phase separation is important for the larger transients associated with small breaks and this is incorporated is the WFLASH code. Figure 7 presents the WFLASH model used in the W evaluation model. The interface between WFLASH and LOCTA IV is shown in Figure 8. A detailed description of the WFLASH model and code options for small break LOCA analysis is presented in Appendix A of the WFLASH[3] report.

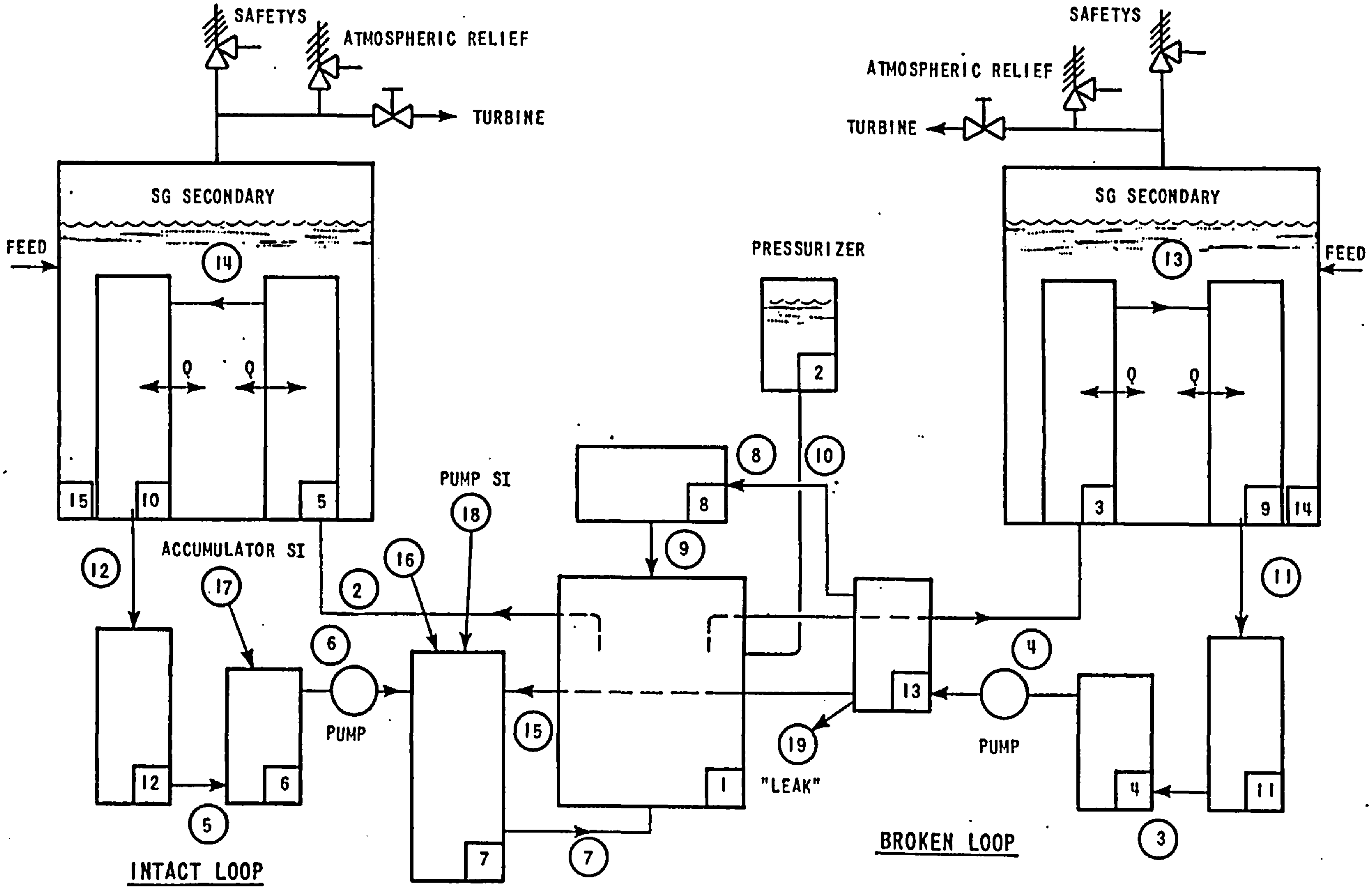

Figure 7. WFLASH Model for PWR

7496-1

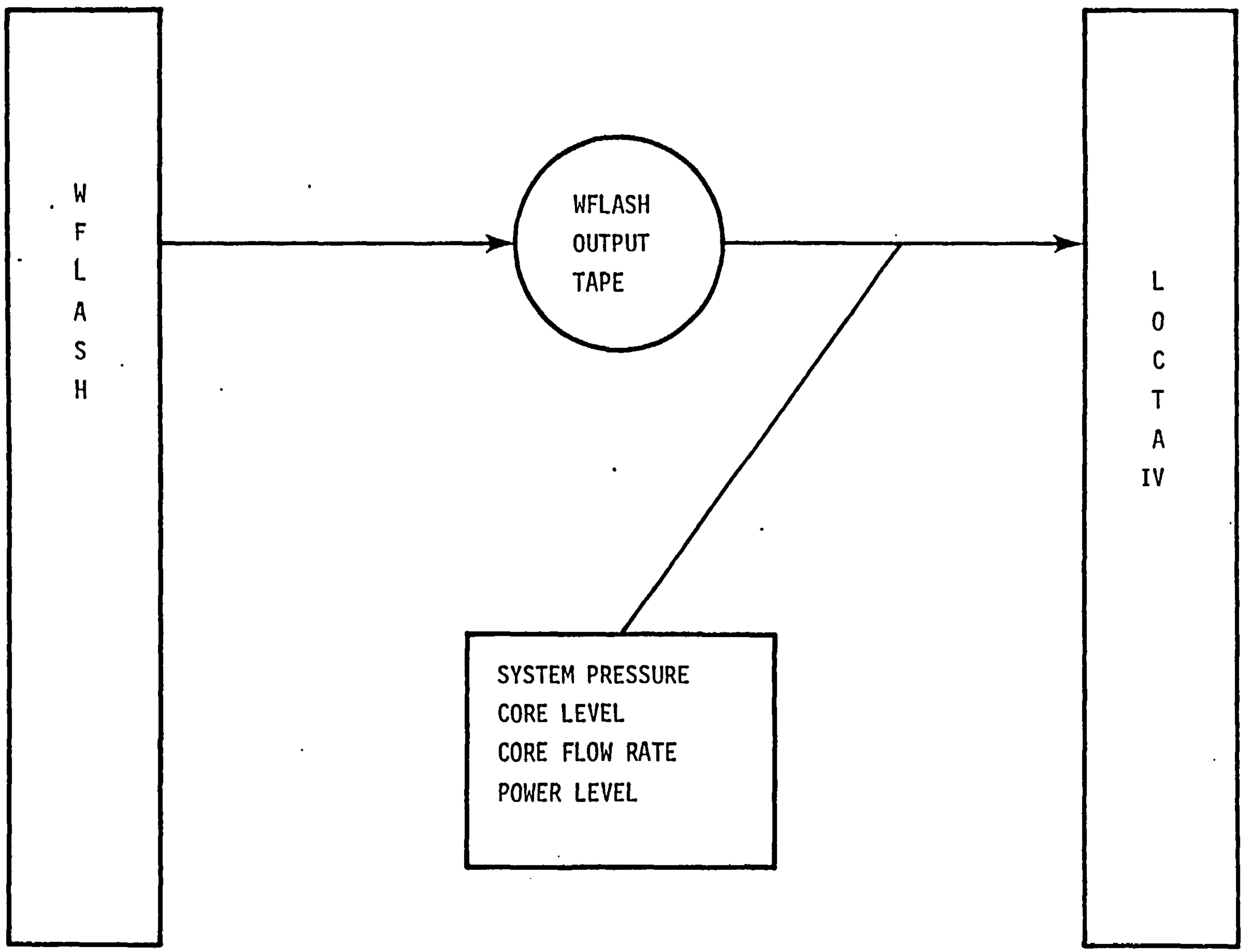

FIGURE 8 - CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL

The LOCTA IV code is used to calculate the clad temperature transient in the hot assembly for small breaks from which the peak clad temperature (for small break range only) can be determined.

4.2 CALCULATION OF MAXIMUM CLADDING OXIDATION

Cladding oxidation thickness is calculated in LOCTA IV based on the Baker-Just metal-water reaction as required by Appendix K of 10CFR50. The method of analysis for the calculation of cladding oxidation is identical to that presented in Section 4.1 and is performed in LOCTA IV when the clad temperature transient is calculated. This 10CFR50.46 requirement is usually less limiting than the peak clad temperature limit. The maximum calculated cladding oxidation occurs on the hot rod of the hot assembly and does not exceed 0.17 times the total cladding thickness before oxidation.

4.3 CALCULATION OF MAXIMUM HYDROGEN GENERATION

Hydrogen generation is calculated in LOCTA IV as a byproduct of the Zr-water (metal-water) reaction using Baker-Just equation. The 10CFR50.46 requirement, of $\leq$ 1% metal-water reaction, refers to a core wide basis. The method of analysis for calculating the maximum hydrogen generation on a core wide basis is similar to the methods presented in Section 4.1 of this report except that a series of LOCTA IV calculations are made by varying the radial peaking factors in each calculation such that various representative radial power regions in the core can be analyzed for the local metal water reaction and hence the hydrogen generation. Each representative radial region is analyzed with one LOCTA IV calculation.

The highest radial power region uses the SATAN VI thermal-hydraulic transient from the hot assembly and the other radial power regions use the SATAN VI average assembly thermal-hydraulic transient.

The total core-wide hydrogen generation is calculated by convoluting the results of the radial power region analysis with the appropriate radial power distribution. This 10CFR50.46 requirement is usually not limiting compared to the 2200°F limit.

4.4 COOLABLE GEOMETRY

The hottest rod in the entire core is analyzed and shown to have margin between computed peak clad temperature and clad melting point. The majority of the rods in the core are substantially cooler and hence no gross core migration is possible.

Changes in geometry due to bursting is calculated in the Westinghouse Evaluation Model based on experimental data. These regions are also shown to be coolable and thus meet this criteria.

4.5 LONG-TERM COOLING

After successful initial operation of the ECCS, the reactor core is recovered with borated ECCS water. This ECCS water has enough boron concentration to maintain core shutdown. Decay heat is removed by a continuous supply of water from the ECCS. This supply initially comes from the refueling water storage tank (RWST). After RWST is empty the ECCS pumps enter a recirculation mode wherein water is drawn from the containment sump and is cooled in the residual heat removal heat exchangers. Hence long term cooling of the core is maintained by the ECCS. The core is maintained in a shutdown state by borated water. Description of the residual heat removal system is provided in the plant SAR.

5.0 CONCLUSIONS

The Westinghouse ECCS evaluation model satisfies the requirements of Appendix K to 10CFR50. This model will be used to analyze ECCS effectiveness for Westinghouse PWR plants (with Zircaloy cladded cores and present type ECCS) in accordance with the requirements of the Acceptance Criteria set forth in 10CFR50.46. These requirements, the criteria themselves and the specified features of the required evaluation models, are considered to be very conservative. This combined with the fact that the design of the ECCS is in accordance with the requirements of 10CFR50.46, ensure that the performance of the ECCS will be adequate to protect the public health and safety. It must be noted, however, that loss-of-coolant accidents are highly unlikely events and should be considered as such..

In the past, Westinghouse and the nuclear industry have performed several experiments relevant to LOCA technology that has increased the understanding of LOCA phenomena. This was reflected in some features of 10CFR50 Appendix K as compared to the Interim Acceptance Criteria of June 1971. Westinghouse will continue its ECCS experimental program to continually aid in understanding LOCA events. New knowledge from these experimental programs will be reflected in future W analyses.

6.0 REFERENCES

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APPENDIX A

WESTINGHOUSE CONTAINMENT PRESSURE MODEL FOR ECCS EVALUATION

Presented here is the Westinghouse containment pressure model for ECCS analysis that insures a conservatively low containment backpressure for the ECCS calculation. Section A.1 discusses input assumptions for the containment pressure code (COCO or LOTIC) such as containment initial condition, containment free volume, active and passive heat sinks, condensing heat transfer coefficients and gap coefficient. Sections A.2 and A.3 present data that is supplied to Westinghouse by the utility or architect engineer for dry and ice containment designs, respectively, that are used as input to the containment pressure codes.

A.1 WESTINGHOUSE CONTAINMENT BACKPRESSURE FOR ECCS EVALUATION

I. Input Information for Model

A. Initial Containment Internal Conditions

The minimum containment gas temperature, minimum containment gas pressure, and maximum humidity encountered under limiting normal operating conditions will be used in the containment model.

B. Initial Outside Containment Ambient Conditions

An appropriate low ambient temperature external to the containment will be used in the containment model.

C. Containment Volume

The maximum net free containment volume will be utilized in the containment backpressure model. This calculation will be performed by the utility or architect engineer and supplied to Westinghouse (See Sections A.2 and A.3)

II. Active Heat Sinks

A. Spray and Fan Cooling Systems

The assumptions for containment cooling systems that are to be utilized in the containment backpressure model will include the assumption of full containment safety systems operating at their maximum available heat removal capacity. In addition, minimum temperature of the stored water used for the spray trains and cooling water for the fan coolers based on technical specification limits will be assumed.

III. Passive Heat Sinks

A. Identification

The heat sinks considered in the containment evaluation model will be established by identifying those passive heat sinks such as structures and components that exist in the containment that would influence the pressure response. This evaluation of heat sinks is to be performed by the utility or the architect engineer and supplied to Westinghouse. (See Sections A.2 and A.3)

B. Heat Transfer Coefficients

The following conservative condensing heat transfer coefficients for heat transfer to the exposed static heat sink during the blowdown and post-blowdown phases of the accident will be used in the containment model.

1. At the end of blowdown, assume a maximum condensing heat transfer coefficient five times higher than that calculated using the Tagami correlation.

$$h_{max} = 75\left(\frac{Q}{Vt_p}\right)^{0.6}$$

where

h_{max} = maximum heat transfer coefficient, Btu/hr-ft^2, °F
Q = primary coolant energy, Btu
V = net free containment volume, ft^3
t_p = time interval to end of blowdown, sec.

Prior to the end of blowdown, assume a parabolic ($\sqrt{t/t_p}$) increase from the stagnant heat transfer coefficient to the peak value specified above.

2. During the long-term stagnation phase of the accident, characterized by low turbulence in the containment atmosphere, assume condensing coefficients equal to 1.0 times that obtained from the Tagami data and represented by the expression:

$$h = 2 + 50X$$

where

X = steam to air weight ratio.

3. During the transition phase of the accident between the end of blowdown and the long-term post-blowdown phase an exponential transient is presented below:

$$h = h_{STAG} + (h_{max} - h_{STAG})\ e^{-0.05(t - t_p)}$$

Westinghouse believes that values for the gap heat transfer coefficient between steel and concrete, for conditions applicable to this analysis, lie in the range 10-100 Btu/hr-°F-ft^2. Preliminary results of available experimental data support this conclusion. However, until more complete data set is available to further support this conclusion, A gap heat transfer coefficient based on steel-to-steel data will be used. A value h = 300 btu/hr-°F-ft^2 will be used in the interim[1-4]. This is clearly a very conservative value.

A.2 CONTAINMENT DATA REQUIRED FOR ECCS EVALUATION FOR DRY CONTAINMENT

I. Conservatively High Estimate of Containment Net Free Volume ______ft^3

II. Initial Conditions

- A. Lowest Operational Containment Pressure ______psia
- B. Lowest Operational Containment Temperature ______°F
- C. Lowest Refueling Water Storage Tank Temperature ______°F
- D. Lowest Service Water Temperature ______°F
- E. Lowest Temperature Outside Containment ______°F

III. Structural Heat Sinks*

- A. For each Surface
  1. Description of Surface
  2. Conservatively High Estimate of Area Exposed to Containment Atmosphere ______ft^3
- B. For each Separate Layer of each Surface
  1. Material
  2. Conservatively Large Estimate of Layer Thickness ______ft
  3. Conservatively High Value of Material Conductivity ______BTU/hr-°F-ft
  4. Conservatively High Value of Volumetric Heat Capacity ______BTU/ft^3-°F

IV. Spray System

- A. Runout Flow for a Spray Pump ______gpm
- B. Number of Spray Pumps Operating with No Diesel Failure ______
- C. Number of Spray Pumps Operating with One Diesel Failure ______
- D. Fastest Post Accident Initiation of Spray System ______secs

*Structural Heat Sinks should also account for any surfaces neglected in Containment Integrity Analysis.

V. Safeguards Fan Coolers (if any)

A. Number of Fan Coolers Operating with No Diesel Failure ______

B. Number of Fan Coolers Operating with One Diesel Failure ______

C. Fastest Post Accident Initiation of Fan Coolers ______secs

D. Performance Data

1. If Fan Coolers Cooled with Service Water, Provide a Curve of Heat Removal Versus Containment Temperature for Lowest Service Water Temperature

2. If Fan Coolers Cooled with Component Cooling Water

a. Provide a Family of Heat Removal Versus Containment Temperatures Curves Covering the Range of Component Cooling Temperatures from the Lowest Service Water Temperature to the Highest Expected Component Cooling Temperature

b. Component Cooling Heat Exchanger UA ____10^6xBTU/hr-°F

c. Component Cooling Water Flow per Component Cooling Heat Exchanger ______gpm

d. Service Water Flow to Component Cooling Heat Exchanger ______gpm

e. Number of Component Cooling Heat Exchangers Operating with No Diesel Failure ______

f. Number of Component Cooling Heat Exchangers Operating with One Diesel Failure ______

A.3 CONTAINMENT DATA REQUIRED FOR ECCS EVALUATION ICE CONDENSER CONTAINMENT

I. Conservatively High Estimate of Containment Net Free Volume ______ft^3

The distribution between upper, lower and dead ended compartments should also be given.

II. Initial Conditions

A. Lowest Operational Containment Pressure ______psia

B. Lowest Operational Containment Temperature for the Upper, ______°F
Lower and Dead Ended Compartments ______°F

C. Lowest Refueling Water Storage Tank Temperature ______°F

D. Lowest Service Water Temperature ______°F

E. Lowest Temperature Outside Containment ______°F

F. Lowest Initial Spray Temperature ______°F

III. Structural Heat Sinks*

A. For each Surface

1. Description of Surface
2. Conservatively High Estimate of Area Exposed to Containment Atmosphere ______ft^2
3. Location in Containment by Compartment

B. For each Separate Layer of each Surface

1. Material
2. Conservatively Large Estimate of Layer Thickness ______ft
3. Conservatively High Value of Material Conductivity ______BTU/hr-°F
4. Conservatively High Value of Volumetric Heat Capacity ______BTU/ft^3

IV. Spray System

A. Runout Flow for a Spray Pump ______gpm

B. Number of Spray Pumps Operating with No Diesel Failure ______

C. Number of Spray Pumps Operating with One Diesel Failure ______

D. Fastest Post Accident Initiation of Spray System ______secs

E. Distribution of the Spray Flow to the Upper and Lower Compartments (should have conservatively high flow to the lower compartment)

V. Deck Fan

A. Fastest Post Accident Initiation of Deck Fans ______secs

B. Conservatively High Flow Rate Per Fan ______cfm

VI. Conservatively Low Hydrogen Skimmer System Flow Rate ______cfm

*Structural Heat Sinks should also account for any surfaces neglected in Containment Integrity Analysis.

A.4 APPENDIX A REFERENCES

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