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LOSS OF COOLANT: CONTROL OF CONSEQUENCE BY EMERGENCY CORE COOLING^[a]

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I. INTRODUCTION

For certain postulated accidents in central-station nuclear power plants, engineered safety systems are intended to mitigate potential consequences to both the public and the plant. These engineered safety systems are intended to cover a range of possible plant failures including the design basis accident (DBA) which, for certain liquid-cooled reactors, is postulated to result from a large rupture of one of the primary system coolant pipes. The pipe rupture and the associated accident are identified as the loss-of-coolant accident (LOCA). This accident and its consequences are controlled by emergency core cooling (ECC) systems which are standby auxiliary cooling systems.

This paper is directed to the current state-of-understanding of those factors which, as identified by the analysis, are requirements for and the performance of emergency cooling systems. Emphasis is placed on light water-cooled pressurized water and boiling water types. The approach taken, to determining the design requirements and performance, in general, is applicable to other reactor systems having liquid coolant at elevated temperatures and pressures.

The body of the paper is divided into three parts as follows: The first part (Section II) describes the process of ECCS performance criteria into design requirements and uses a "fault tree" technique to identify systematically the phenomena and system behavior in the LOCA which are important to the selection of ECC design requirements. The second part (Section III) presents "performance maps" of current emergency core cooling systems for both pressurized and boiling water reactors. These performance maps are derived from postblowdown core behavior based on information obtained from experimental programs on emergency core cooling, metal-water reaction rates, and embrittlement. Part three (Section IV) presents a survey of the research and development, on an international basis, addressed to the various areas of hydraulic, thermodynamic, and mechanical effects surrounding the performance of emergency core cooling systems in light water-cooled reactors. The concluding section (Section V) summarizes the more important factors in determining the selection of ECC system design requirements. Additionally, the relevant research and development planned and needed for improving the analytical base for ECC design and evaluation are cited.

Appendices are included which provide (a) a summary description of contemporary PWR and BWR primary systems and the associated emergency core cooling systems, and (b) a summary description of the loss-of-coolant process and ECC for both PWR's and BWR's.

II. FACTORS AFFECTING ECCS PERFORMANCE AND DESIGN REQUIREMENTS

The basic criterion for limiting the consequences of an LOCA in contemporary light water-cooled reactors is as follows:

A reactor core, in the unlikely event of a loss-of-coolant accident, must be maintained in a definable heat transfer geometry in order to assure the potential for continued removal of after-heat.

Subsequent criteria can then be derived that dictate the upper limits for certain properties of reactor core materials that must not be exceeded, assure that the structural integrity of core internals is maintained. Property limits of core materials include maximum temperature, and possibly time-at-temperature, limits which the hotter regions of the core can be allowed to exceed.

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[a] Work performed under auspices of the United States Atomic Energy Commission under Contract No. AT(40-1)-1088.

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These criteria must be reduced to performance and functional design requirements for removal of a specific sensible heat from the reactor core by auxiliary cooling. This section reviews the processes which could be used by the designer to translate the performance requirements into functional requirements and ultimately into suitable ECC system design requirements.

The first step in the translation process would be for the designer to evaluate LOCA thermal-hydraulic behavior for a particular plant design in order to determine the demand requirements for ECC. These evaluations would be performed for a range of break sizes and locations with particular emphasis on the influence that the blowdown hydraulic forces might have on the effectiveness of the emergency core coolant. In these evaluations, the designer would obtain the histories of the mass and the fluid states in the core by using digital computer codes that are capable of tracking the mass, momentum, and energy redistribution of the coolant as the system depressurizes. The core flow rate and the state of the fluid at the core inlet would be used in additional calculations of the core thermal response to determine the extent to which the energy generated within the fuel is transferred to the coolant. The core thermal response calculations would produce histories of core temperature.

These cladding temperature histories, obtained for a variety of assumed primary system ruptures, would provide the designer with the time after rupture that the ECC systems would have to be actuated in order to be effective. For a postulated break in the primary system which reflooding is the basic ECC action, some of the factors which the PWR designer would consider in selecting requirements for an ECC system are the following:

- (1) The cladding temperature and rate of temperature rise in the latter phase of blowdown prior to reflooding
- (2) The volume and injection rate of water which must be supplied to fill the lower plenum and initiate core reflooding at a prescribed time
- (3) The magnitude and duration of system decompression forces which could act on the injected water and prevent it from reaching the lower plenum
- (4) Core thermal behavior once reflooding is initiated, and also the potential system effects which permit or inhibit the design flooding rate to be achieved.

Acquisition of this information with adequate precision to permit quantification of ECC effectiveness places significant demands on the completeness and correctness of the analysis methods used to characterize the thermal processes occurring during a loss-of-coolant accident with emergency core coolant injection. The analysis methods available to the designer have been partially verified by experiment in certain areas, and, as discussed in Section IV, further effort is being addressed to the verification process.

To assist in determining the adequacy of ECC performance requirements developed using the current analysis methods, systematic investigations can be conducted to determine whether uncertainties in the analysis methods could lead to underpredictions of cladding temperatures, emergency coolant delivery rates, and heat removal rates. Such investigations can be conducted to show where improvements in the completeness and precision of the analysis will lead to quantification of potential conservatism and margins in ECC system designs. Ultimately the refinement and experimental verification of analysis techniques can lead to an increased level of confidence in the determination of (a) the adequacy of ECC system requirements, the existence of margins and (b) the potential for unneeded system complexity which could be a causative factor in accidents.

As an illustration of probing the uncertainties in the analysis techniques a fault tree evaluation method has been developed which exhibits the potential for error in the modeling methods which could lead to an erroneous conclusion on reactor performance. The tree presented was generated for discussion purposes and is only developed to the level needed to identify specific elements of the analysis. In general, greater detail is provided for those elements of the tree within the time interval of a LOCA in which ECC action is involved.

Since each plant design responds somewhat differently to a particular pipe break area during a postulated LOCA, a particular type of plant and pipe break was necessarily selected for the study. The example fault tree developed applies to a typical PWR plant subjected to the hypothetical design basis accident which for this plant is a double-ended break in an inlet pipe.

The material presented in this section of the paper includes the following:

- (1) Fault tree attributes, logic, and symbolism
- (2) A development of each branch of the example tree and a qualitative discussion of each "fault" relative to its potential for causing a significant error in analysis results

performance and functional design requirements for removal of a specified amount of heat from the core by emergency core cooling. This section reviews the processes which could be used by the designer to translate these requirements into suitable ECC system design requirements.

It would be for the designer to evaluate LOCA thermal-hydraulic behavior for the design demand requirements for ECC. These evaluations would be performed for a wide range of emphasis on the influence that the blowdown hydraulic forces might have on the system. In these evaluations, the designer would obtain the histories of the mass flow rates and temperatures from computer codes that are capable of tracking the mass, momentum, and energy balances. The core flow rate and the state of the fluid at the core inlet would then be used to determine the thermal response to determine the extent to which the energy generated and stored in the core would be removed. The core thermal response calculations would produce histories of the cladding

temperatures for a variety of assumed primary system ruptures, would provide an index of the effectiveness of the ECC. It would have to be actuated in order to be effective. For a postulated break in a PWR in the primary system, the factors which the PWR designer would consider in selecting requirements for

the ECC would be the rate of temperature rise in the latter phase of blowdown prior to

the time at which water which must be supplied to fill the lower plenum and initiate

the ECC. The analysis would include the effects of decompression forces which could act on the injected water and the lower plenum

and the time at which the ECC is initiated, and also the potential system effects which permit the ECC to be achieved.

The adequacy of the precision to permit quantification of ECC effectiveness would place a question on the correctness of the analysis methods used to characterize the thermal-hydraulic behavior of the system with emergency core coolant injection. The analysis methods currently used are based on data verified by experiment in certain areas, and, as discussed in Section IV, considerable work is in process.

The ECC performance requirements developed using the current analysis methods, and the methods to determine whether uncertainties in the analysis methods could result in unacceptable emergency coolant delivery rates, and heat removal rates. Such investigations can also be used to determine the completeness and precision of the analysis will lead to quantification of the effectiveness of the system designs. Ultimately the refinement and experimental verification of the analysis methods will lead to a level of confidence in the determination of (a) the adequacy of ECC systems and the required system complexity which could be a causative factor in accident initiation.

In the analysis techniques a fault tree evaluation method has been used to determine the effectiveness of the analysis methods which could lead to an erroneous conclusion on required ECCS. The example fault tree developed for discussion purposes and is only developed to the level needed to identify the major elements of the tree within the time period of the accident.

It is different to a particular pipe break area during a postulated LOCA, a particular area of the system is selected for the study. The example fault tree developed applies to a typical four-loop system accident which for this plant is a double-ended break in an inlet pipe.

The paper includes the following:

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- (3) Comments on the magnitude of the error required to cause an unexpected violation of the stated limits
- (4) A summary of the overall PWR analysis fault tree identifying what appear to be the most significant end faults for the tree
- (5) A discussion of how the example tree could be modified for application to analysis of a BWR.

FAULT TREE ATTRIBUTES, LOGIC, AND SYMBOLISM

Specific attributes of the fault tree evaluation method used in this paper to highlight the important fact "loss-of-coolant: control of consequence by emergency core cooling", are as follows:

- (1) It establishes the effect-cause relationships starting with some undesired event and continuing through event sequences until the possible initiating elements are found.
- (2) It involves a deductive method of reasoning. Generally, the rules of development are straightforward. For a given event, the conditions necessary to cause the event and those sufficient to cause it are required.
- (3) It involves a recorded process which uses logic symbology to show relationships in a form which follows Boolean rules and relationships (basic laws of thought) and in the language of the system being evaluated.
- (4) It is a communication tool, expressed in terms of the system, understandable by designer and management alike.

Fault tree analysis is a systematic procedure that identifies and records the various combinations of conditions, actions, or events that can result in a predefined system state or undesired event. The development of a fault tree is a definition of an undesired condition. The system is then methodically analyzed to determine all the logical and functional fault events that can cause the undesired event. The principal contributory fault events are then further developed to establish the logical chain of subevents. All fault events are coupled through logic gates. Figure 1 summarizes the steps in constructing the tree and defines each of the logic gates.

TYPICAL EXAMPLE OF A FAULT TREE FOR ANALYSIS OF THE DESIGN BASIS ACCIDENT IN A PWR

The top structure for an example PWR fault tree is shown in Figure 2. The undesired event is "analysis of inadequate ECC requirements-cladding temperature or embrittlement limits exceeded". The "and" gate and the structure immediately below the undesired event reflect the fact that the remaining structure of the tree is developed to investigate sources of errors in the analysis which is used to determine the requirements for and the adequacy of ECC system design basis accident situation.

The investigation near the top of the tree was separated into analysis of events that would cause high cladding temperatures prior to core reflooding and analysis of events that would cause high temperatures after core reflooding is initiated. The investigation places emphasis on two major aspects of an ECC system design--first, the time in the accident sequence at which ECC action is initiated, and second, the effectiveness of the energy removal process once ECC action is initiated.

The five major branches of the fault tree, shown on Figure 2, are developed individually in the remainder of the paper. In general, each branch of the tree is developed only to a level suitable for discussing potential errors or deficiencies in portions of the analysis. Once the desired level is achieved the fault is shown as a diamond which indicates that further development of the tree is required to define the primary faults. Each diamond is briefly discussed to provide an indication of the relevance of the identified event to ECC system design and performance requirements. The specific statements are developed in the evaluation of the behavior of a four-loop PWR during a design basis accident. The term "predicted" on the fault tree indicates an event or condition that is based on the accident concept portrayed by existing safety analysis codes.

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EVENT REPRESENTATIONS

The rectangle identifies an event that results from the combination of fault events through the input logic gate.

The circle describes a basic fault event that requires no further development. Frequency and mode of failure of items so identified are derived from empirical data.

The triangles are used as transfer symbols. A line from the apex of the triangle indicates a transfer in and a line from the side denotes a transfer out.

The diamond describes a fault event that is considered basic in a given fault tree. The possible causes of the event are not developed either because the event is of insufficient consequence or the necessary information is unavailable.

The house indicates an event that is normally expected to occur.

LOGIC OPERATIONS

The AND gate describes the logical operation whereby the coexistence of all input events is required to produce the output event.

The OR gate defines the situation whereby the output event will exist if one or more of the input event exists.

The INHIBIT gate describes a causal relationship between one fault and another. The input event directly produces the output event if the indicated condition is satisfied. The conditional input defines a state of the system that permits the fault sequence to occur, and may be either normal to the system or results from failures.

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Fig. 1 Fault tree symbolism.

Analysis fails to identify
adequate ECC requirements—
cladding temperature
or embrittlement Limits
exceeded

and

Design
basis
accident
occurs
in large
PWR

Error in
calculational
techniques

Predictions
show
temperature
and
embrittlement
limits not
exceeded

or

Analysis in error—
Limits violated
prior to core
reflooding

or

Analysis in error—
Limits violated
during core
reflooding

or

①

②

③

④

⑤

Analysis overpredicts
energy removal rates
or underpredicts
stored energy

Analysis
underpredicts
time to initiate
reflooding

Analysis underpredicts
core temperature
at time of reflooding
initiation

Analysis underpredicts
power generation
during core reflooding

Analysis overpredicts
energy removal rates
during core reflooding

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Fig. 2 Tree development for cladding temperature or embrittlement limits exceeded because analysis fails to identify adequate ECC requirements.

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BRANCH 1--ANALYSIS OVERPREDICTS ENERGY REMOVAL RATES OR UNDERPREDICTS STORED ENERGY

Branch 1, shown on Figure 3, identified potential analysis errors which might result in the cladding temperature underpredicted prior to core reflooding. For the development of this branch the blowdown and refill times are assumed predicted.

This branch is not developed in extensive detail because large errors would have to exist in the analysis for a fault to occur and such errors are considered unlikely. Only potential faults associated with the analysis underpredicting energy removal rates or overpredicting energy removal rates are shown. The period of interest for this branch is about 10 to 30 seconds after rupture. This period, based on current predictions, can be separated into three equal time increments: (a) ten seconds for the system to depressurize to 600 psi, (2) ten seconds for the system to further depressurize to the containment pressure, and (3) ten seconds for the accumulators to refill the lower plenum after blowdown is complete. The expected maximum cladding temperature during this period is several hundred degrees Fahrenheit below the temperature limit; however the possibility of exceeding the cladding temperature limit could be exceeded. For example, if the energy stored within the fuel is as predicted, but no heat transfer occurs, cladding temperatures potentially could rise to above 2500°F by the time reflooding is initiated. Without blowdown heat transfer, temperatures would reach about 2000°F in 4 or 5 seconds as a result of energy redistribution in the fuel rod. After this time, a relatively slow heatup rate exists which is primarily dependent on power generation. At about 30 seconds after rupture the adiabatic heatup rate is about 20°F/sec in the high-power density reactor core.

On the left side of Figure 3 under "analysis overpredicts blowdown heat transfer" the tree is terminated with "analysis overpredicts surface heat transfer coefficients". If the tree had been developed further the following potential faults in the analysis would have been included:

- (1) Analysis overpredicts the time to CHF
- (2) Analysis overpredicts heat transfer during transition boiling
- (3) Analysis overpredicts energy removal rates during the period when film boiling and convection correlations are applied.

Development of the tree below each of the various heat transfer processes noted would have reflected errors resulting from inapplicability of the correlations to a transient environment or from errors in the calculation of core mass flow rate. Errors in core mass flow rate or fluid state calculations may result from errors in the break flow model, the pipe friction model, hydraulic resistance model, or the steam generator model, and from errors in calculations of the flow maldistribution or flow or blockage.

Sensitivity studies are either not available or not sufficiently complete to quantify the magnitude of the errors in the calculations which would be required to cause a fault on Branch 1 due to analysis errors associated with the analysis overpredicting surface heat transfer coefficients. However, the authors' opinion is that for this fault to occur actual heat transfer coefficients used after core dryout would have to be at least 30 to 40% less than predicted and the transition cooling correlations in the codes would have to be applied for a longer period of time than is actually appropriate, probably 30 to 60 seconds. Currently planned research programs in the blowdown heat transfer area are expected to show that not only is this branch of the tree unlikely but that current analysis underpredicts blowdown heat transfer.

If the right side of Figure 3 under "analysis underpredicts stored energy or power generation rate" had been developed further, the need to examine models or assumptions in the analysis associated with the following areas would be evident:

- (1) Gap conductance
- (2) UO₂ physical properties (thermal conductivity and specific heat)
- (3) Prerupture power density
- (4) Decay power generation rates.

The gap conductance is probably the most influential and the most difficult parameter to quantify and, therefore, the most likely fault source. Gap conductance is a function of such factors as initial contact resistance between the pellet and the cladding, cladding temperature, burnup, and fuel densification. Calculated values vary from about 1500 to 8000 Btu/hr-ft²-°F.

ENERGY REMOVAL RATES OR UNDERPREDICTS STORED ENERGY

potential analysis errors which might result in the cladding temperature being development of this branch the blowdown and refill times are assumed to occur as

ve detail because large errors would have to exist in the analysis before this fault unlikely. Only potential faults associated with the analysis underpredicting energy removal rates are shown. The period of interest for this branch is about 30 seconds. be separated into three equal time increments: (a) ten seconds for the system to system to further depressurize to the containment pressure, and (3) approximately lower plenum after blowdown is complete. The expected maximum temperature Fahrenheit below the temperature limit; however the possibility does exist that the For example, if the energy stored within the fuel is as predicted, but no blowdown potentially could rise to above 2500°F by the time reflooding is initiated. In fact, es would reach about 2000°F in 4 or 5 seconds as a result of temperature a relatively slow heatup rate exists which is primarily dependent on decay heat the adiabatic heatup rate is about 20°F/sec in the high-power density regions of the

sis overpredicts blowdown heat transfer" the tree is terminated with the statement "Efficients". If the tree had been developed further the following potential errors in

IF

during transition boiling

oval rates during the period when film boiling and convection

ious heat transfer processes noted would have reflected errors resulting from the environment or from errors in the calculation of core mass flow rate or fluid state. calculations may result from errors in the break flow model, the pump model, the reactor model, and from errors in calculations of the flow maldistribution due to cross

e or not sufficiently complete to quantify the magnitude of the error in present a fault on Branch 1 due to analysis errors associated with the diamond "analysis However, the authors' opinion is that for this fault to occur actual heat transfer to be at least 30 to 40% less than predicted and the transition cooling correlations longer period of time than is actually appropriate, probably 30 to 40% longer. The blowdown heat transfer area are expected to show that not only is a fault on this analysis underpredicts blowdown heat transfer.

ysis underpredicts stored energy or power generation rate" had been developed options in the analysis associated with the following areas would have been made

nductivity and specific heat)

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influential and the most difficult parameter to quantify and, therefore, could be a such factors as initial contact resistance between the pellet and the cladding, power densification. Calculated values vary from about 1500 to 8000 Btu/hr-ft²-°F; the

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Analysis overpredicts
energy removal rates
or underpredicts
stored energy

or

Analysis overpredicts
blowdown heat
transfer

and

Stored
energy
and power
generation
as
predicted

Analysis overpredicts
surface heat transfer
coefficients

Analysis underpredicts
stored energy or
power generation
rate

and

Rod energy
removal
rates as
predicted

Analysis underpredicts
stored energy or power
generation rate higher
than predicted

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Fig. 3 Tree development for analysis overpredicts energy removal rates or underpredicts stored energy.

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values are lowest at the beginning of life and increase with burnup. For the higher values, large changes have little effect as values drop to about 1500 Btu/hr-ft²-°F, the time-temperature response of the cladding starts to be affected. I believe that the analysis is not likely to underpredict the stored energy by more than 20% which is the minimum error needed to cause a fault on this branch of the tree. The early LOFT tests are expected to provide a basis for establishing bounds on the stored energy calculations and showing that the fault is indeed unlikely.

The houses on Figure 3 are shown to identify assumptions required to make the event chains independent.

Branch 2--ANALYSIS UNDERPREDICTS TIME TO INITIATE REFLOODING

This branch of the tree, shown in Figure 4, addresses the problem of whether elements of the analysis which characterize LOCA behavior are sufficiently erroneous that limits could be exceeded because core reflooding does not occur as predicted. For this discussion the blowdown heat transfer, prerupture stored energy, and decay power generation (Branch 1) are assumed to occur as predicted. As would be expected, elements on this branch include items related to whether the lower plenum and how fast the vessel lower plenum could be filled by injection of the inventory in the accumulator. The diamonds and the inhibit gate are discussed in the following subsections. The inhibit gate is used to emphasize that a significant portion of the injected ECC (40% or more) bypasses the core and is expelled through the break during blowdown. The water level in the lower plenum and the blowdown time are not of particular significance to this fault.

- (1) Analysis Overpredicts Accumulator Delivery Rate. For the particular type of plant and accident considered, three accumulators, each with a nominal water volume of 850 ft³, are assumed available. The fourth accumulator, the one on the broken loop, is assumed to discharge into the containment. The nominal predicted discharge rate for the three accumulators is about 80 ft³/sec. The predicted behavior is for the accumulators to fill the lower plenum, downcomer, and core bypass area which is connected to the downcomer (a total volume of about 1750 ft³) in slightly more than 20 seconds after the blowdown is completed. The lower plenum would refill up to the bottom of the core in about 12 seconds after blowdown.

The analyses would overpredict the actual delivery rates if the hydraulic resistances between the accumulators and the primary system are higher than assumed, or if the accumulator gas pressure or water volume is lower than assumed. To be a significant problem on this branch of the tree, the error, in the author's opinion, would have to result in the predicted delivery rate being more than a factor of two greater than the actual accumulator delivery rate. A reduction of the delivery rate by two from that predicted would introduce about a ten second delay in the time required to fill an empty lower plenum or, for an adiabatic heatup rate, permit the temperature to rise about 200°F higher than predicted. Although extremely unlikely, a combination of errors involving the prerupture accumulator water level, the coolant bypassing the core to the break during blowdown, and the length of the blowdown time could occur such that, contrary to predictions, the lower plenum does not fill before the accumulators are emptied. The time would then be ample for temperature limits to be exceeded before the low pressure injection system (LPIS) could raise the water level to the core bottom and initiate reflooding. The delivery rate of the LPIS, which is eight to ten times less than the delivery rate for the three accumulators, would be inadequate to prevent this fault.

- (2) Analysis Underpredicts ECC Bypass to the Break During Blowdown. The possibility exists that mechanisms not properly accounted for in the analysis could result in the predictions of the injected ECC carried to the break being less than actually occurs. The primary concern is the analytical difficulty in characterizing the countercurrent flow situation in the downcomer region of the vessel. Steam flow at high velocity up the downcomer during blowdown may prevent water from falling to the lower plenum. However, even if 100% of the ECC injected is assumed to bypass the core and is expelled through the break, a fault does not occur for this branch of the tree unless the lower plenum is empty at the end of blowdown and the blowdown time from 600 psi to containment pressure is significantly longer than predicted. For example, in terms of the cladding temperature rise resulting from ECC bypass and a longer blowdown time, an error of a factor of two in the expected blowdown time would add, at most, 300°F to the cladding temperature at initiation of reflooding. However, because of the relatively low LPIS delivery rate, the limiting case would be failure of the accumulators to refill the lower plenum. This situation would probably occur with 100% bypass if the depressurization time from 600 psi to the containment pressure approaches 20 seconds. (At 80 ft³/sec a water volume of 1600 ft³ would be lost out of about 2500 ft³ total in the three accumulators.)

and increase with burnup. For the higher values, large changes have little effect. However, at 1000°F, the time-temperature response of the cladding starts to be affected. The authors underpredict the stored energy by more than 20% which is the minimum error that would be of the tree. The early LOFT tests are expected to provide a basis for establishing error bounds and showing that the fault is indeed unlikely.

to identify assumptions required to make the event chains independent.

TIME TO INITIATE REFLOODING

Figure 4. addresses the problem of whether elements of the analysis codes used to predict the time to initiate reflooding are so grossly erroneous that limits could be exceeded because core reflooding does not occur as predicted. Downward heat transfer, prerupture stored energy, and decay power generation (elements of the tree). As would be expected, elements on this branch include items related to when ECC injection into the vessel lower plenum could be filled by injection of the inventory in the accumulators. The analysis is discussed in the following subsections. The inhibit gate is used to emphasize that, unless a significant portion (40% or more) bypasses the core and is expelled through the break during blowdown, the time to initiate reflooding and the blowdown time are not of particular significance to this fault.

Accumulator Delivery Rate. For the particular type of plant and accident sequence, each with a nominal water volume of 850 ft³, are assumed available. The flow rate from one on the broken loop, is assumed to discharge into the containment. The predicted flow rate for the three accumulators is about 80 ft³/sec. The predicted time to fill the lower plenum, downcomer, and core bypass area which is about 1750 ft³ (a total volume of about 1750 ft³) in slightly more than 20 seconds after initiation. The lower plenum would refill up to the bottom of the core in about 12 seconds.

To predict the actual delivery rates if the hydraulic resistances between the accumulator and the break system are higher than assumed, or if the accumulator gas pressure is higher than assumed. To be a significant problem on this branch of the tree, the error, would have to result in the predicted delivery rate being more than a factor of two less than the accumulator delivery rate. A reduction of the delivery rate by two from the predicted rate would result in about a ten second delay in the time required to fill an empty lower plenum. This would permit the temperature to rise about 200°F higher than predicted. It is highly unlikely, a combination of errors involving the prerupture accumulator flow rate, the time to pass the core to the break during blowdown, and the length of the break would result such that, contrary to predictions, the lower plenum does not fill before the time to initiate reflooding. The time would then be ample for temperature limits to be exceeded. The lower plenum injection system (LPIS) could raise the water level to the core bottom and maintain it at that level. The flow rate of the LPIS, which is eight to ten times less than the delivery rate of the accumulators, would be inadequate to prevent this fault.

Bypass to the Break During Blowdown. The possibility exists that the analysis could result in the predictions of the injected water volume being less than actually occurs. The primary concern is the analytical prediction of the countercurrent flow situation in the downcomer region of the vessel. The possibility that the downcomer during blowdown may prevent water from falling to the lower plenum is addressed. If 100% of the ECC injected is assumed to bypass the core and is expelled through the break, it does not occur for this branch of the tree unless the lower plenum is empty at the time of the blowdown. The time from 600 psi to containment pressure is significantly less than the blowdown time. For example, in terms of the cladding temperature rise resulting from ECC injection, an error of a factor of two in the expected blowdown time would result in a cladding temperature at initiation of reflooding. However, because of the high flow rate, the limiting case would be failure of the accumulators to refill the lower plenum. This would probably occur with 100% bypass if the depressurization time from 600 psi to containment pressure approaches 20 seconds. (At 80 ft³/sec a water volume of 1600 ft³ is required to refill the lower plenum. At 80 ft³/sec a water volume of 1600 ft³ total in the three accumulators.)

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Analysis underpredicts
time to initiate
reflooding

or

System effects
delay core
reflooding

Analysis overpredicts
accumulator delivery
rate

Analysis underpredicts
ECC bypass to break
during blowdown

and

Analysis underpredicts
water loss from lower
plenum

Analysis underpredicts
blowdown time

Analysis underpredicts
time to initiate
reflooding

or

System effects
delay core
reflooding

Analysis overpredicts
accumulator delivery
rate

and

Analysis underpredicts
blowdown time

Analysis underpredicts
blowdown time

2

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Although hydraulic mechanisms resulting in ECC bypass of the core during blowdown are not precisely modeled in the analysis code, indications are that the calculated effectiveness of the ECC will be suitably conservative if 100% bypass is assumed to occur until the time in the blowdown process at which calculated momentum fluxes up the downcomer are below the value at which countercurrent flooding correlations² predict downflow of water.

- (3) Analysis Underpredicts Blowdown Time. For this branch of the tree the preceding discussion indicated that the actual blowdown time from 600 psi to the containment pressure would have to be at least a factor of two longer than predicted to be a fault. Blowdown experiments have shown that models, such as the Moody critical flow model³ derived from steady state experiments, do overpredict flow rates, and hence underpredict blowdown time, during the saturated portion of blowdown. However, the authors consider errors of two or more in blowdown time during the delivery period unlikely.
- (4) Analysis Underpredicts Water Loss from the Lower Plenum. If a system is slowly depressurized from operating conditions to containment pressure, about 40% of the fluid in the lower plenum flashes to steam and is lost from the lower plenum. A few percent also flashes due to heat transfer from surrounding surfaces. To empty the lower plenum at the end of blowdown, the other 60% of the fluid would have to be removed by entrainment. For large breaks accompanied by rapid depressurization rates, high-velocity reverse flow could occur in the core and empty the lower plenum.

Present analysis models, limited to a one-dimensional representation of the fluid flow, cannot directly accommodate entrainment in the lower plenum. For large breaks, homogeneous one-dimensional representation of the lower plenum does predict that the lower plenum is nearly empty at the end of blowdown. Thus, in general, the predicted results are not expected to deviate significantly from worst-case conditions. A significant error could possibly be introduced in the predictions of the water remaining in the lower plenum if the models are manipulated to accommodate modeling difficulties regarding the delivery phase of the accident. For example, if the model is changed so that the accumulators discharge cold water into the lower plenum during blowdown and then in effect remove some assumed fraction of the ECC at the end of blowdown to account for ECC bypass, the results may show significantly more water remaining in the lower plenum than would actually occur.

Branch 3--ANALYSIS UNDERPREDICTS CORE TEMPERATURE AT TIME OF REFLOODING INITIATION

This branch of the tree addresses the problem of determining whether errors can occur in the calculated time-to-behavior of the cladding such that the predicted temperatures are sufficiently lower than the actual temperatures there in the temperature limits being exceeded even if reflooding effectiveness is as predicted.

The tree developed for this branch is shown in Figure 5. It has essentially the same elements as those on Branch (Figures 3 and 4), but the acceptable error in maximum temperature rise during blowdown is several hundred degrees. This branch of the tree, based on predicted energy removal rates during reflooding, takes into account the fact that turnaround does not occur at the initiation of reflooding for certain designs and points out that a temperature exists which, if exceeded at the time of core reflooding, will result in violation of the upper limit. The faults on this branch of the tree are essentially the same as for the two branches previously discussed under the fault "analysis in error-limits violated prior to reflooding". If developed, the diamond on the right "analysis overpredicts energy removal rates during blowdown or underpredicts stored energy" would include all elements of Branch 1 of Figure 3. The inputs on the left side of Branch 3 are the same as shown on Branch 2. If the predicted temperatures at the start of core reflooding are about 1600°F, the magnitude of the acceptable errors discussed for inputs on Branches 1 and 2 would have to be reduced by about 40% to be appropriate for this branch. In the authors' opinion, faults on Branch 1 concerned with errors in predicted blowdown heat transfer (probably resulting from errors in calculated core flow areas and fluid state) and in prerupture stored energy would be more significant than the other errors on these two branches.

The remainder of the discussion for this branch summarizes events which could cause the predicted cladding temperature at reflood initiation to be lower than the actual cladding temperature. The maximum temperature achieved during core reflood will occur when the heat removal is equivalent to the power generation (decay power and metal-water reaction energy). A significant time will have elapsed after reflooding is initiated before overall maximum temperatures are reached (40 sec). During the first few seconds after reflooding starts, little change in the adiabatic temperature rise occurs at the core mouth. A rate of temperature rise of 20°F/sec alone potentially permits temperatures to rise by 100°F. After the first few seconds, water levels in the core are sufficiently high to provide sufficient steam generation to entrain fluid and carry it to the elevations. Typically, the energy removal rates at this time are still less than the power generation rate because sy-

ting in ECC bypass of the core during blowdown are not precisely
ations are that the calculated effectiveness of the ECC will be
is assumed to occur until the time in the blowdown process at
up the downcomer are below the value at which countercurrent
ow of water.

ime. For this branch of the tree the preceding discussion indicated
600 psi to the containment pressure would have to be at least a
to be a fault. Blowdown experiments have shown that models, such
derived from steady state experiments, do overpredict flow rates,
time, during the saturated portion of blowdown. However, the
e in blowdown time during the delivery period unlikely.

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pressure, about 40% of the fluid in the lower plenum flashes to
plenum. A few percent also flashes due to heat transfer from
lower plenum at the end of blowdown, the other 60% of the fluid
inment. For large breaks accompanied by rapid depressurization
occur in the core and empty the lower plenum.

one-dimensional representation of the fluid flow, cannot directly
lower plenum. For large breaks, homogeneous one-dimensional
does predict that the lower plenum is nearly empty at the end of
redicted results are not expected to deviate significantly from
error could possibly be introduced in the predictions of the water
e models are manipulated to accommodate modeling difficulties
e accident. For example, if the model is changed so that the
to the lower plenum during blowdown and then in effect remove
the end of blowdown to account for ECC bypass, the results may
ing in the lower plenum than would actually occur.

TEMPERATURE AT TIME OF REFLOODING INITIATION

blem of determining whether errors can occur in the calculated time-temperature
temperatures are sufficiently lower than the actual temperatures thereby resulting
reflooding effectiveness is as predicted.

on in Figure 5. It has essentially the same elements as those on Branches 1 and 2
maximum temperature rise during blowdown is several hundred degrees less. This
removal rates during reflooding, takes into account the fact that temperature
reflooding for certain designs and points out that a temperature exists which, if
result in violation of the upper limit. The faults on this branch of the tree are
eviously discussed under the fault "analysis in error-limits violated prior to core
ght "analysis overpredicts energy removal rates during blowdown or underpredicts
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e flow areas and fluid state) and in prerupture stored energy would be more likely

branch summarizes events which could cause the predicted cladding temperature at
adding temperature. The maximum temperature achieved during core reflooding
the power generation (decay power and metal-water reaction energy). In general,
ng is initiated before overall maximum temperatures are reached (40 to 80 sec).
ts, little change in the adiabatic temperature rise occurs at the core midplane and
ne potentially permits temperatures to rise by 100°F. After the first few seconds,
o provide sufficient steam generation to entrain fluid and carry it to the high
at this time are still less than the power generation rate because system effects,

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Fig. 5 Tree development for analysis underpredicts core temperature at time of reflooding initiation.

related to steam binding, limit reflooding rates. Temperatures continue to rise but at a lower rate. As time passes in the core moves to higher elevations, the energy removal rates increase, decay power levels decrease, and temp occurs.

The relationship of the maximum calculated cladding temperature to the temperature at reflood initiation is shown in Section III which presents a summary of sensitivity study results. For the particular example tree, the study shows that, if actual temperatures are about 2000°F at the start of core reflood rates are as predicted, temperature limits will be exceeded. This branch of the tree points out the need for precision in the analysis of system behavior during blowdown than would be required to show that faults on the (Branches 1 and 2) will not occur.

Branch 4--ANALYSIS UNDERPREDICTS POWER GENERATION DURING CORE REFLOODING

Branch 4 describes events which would cause cladding temperature and temperature-time limits to be exceeded if power generation rates are lower than actual rates. The tree developed for this branch is shown in Figure 6.

Fig. 6 Tree development for analysis underpredicts power generation during core reflooding.

This fault is straightforward and involves two source terms--decay heat power generation and metal-water reaction. The time period of interest is 30 to 120 seconds after rupture. For the plant examined (temperature at reflooding initiation about 2000°F) excessive power would result if the calculated values are low by 10 to 20%. No decay heat power are obtained using 120% of the ANS standard. Currently the conservatism of this value is established. The metal-water reaction rate is usually calculated using the Baker-Just equations.⁴ Posttest examination used in the Full-Length Emergency Cooling Heat Transfer Projects^{5,6} to determine oxide thickness indicate Baker-Just relationship results in overprediction of the metal-water reaction rate. For the time sequence example, the metal-water reaction does not become significant until temperatures are near the specified limit.

In the authors' opinion, predictions based on a power generation rate of 1.2 times the ANS standard and a multiplier in the Baker-Just energy calculation will prevent a fault on Branch 4.

ates. Temperatures continue to rise but at a lower rate. As time passes, the quench front energy removal rates increase, decay power levels decrease, and temperature turnaround

calculated cladding temperature to the temperature at reflood initiation for various which presents a summary of sensitivity study results. For the particular plant considered t, if actual temperatures are about 2000°F at the start of core reflooding and reflooding will be exceeded. This branch of the tree points out the need for considerable more r during blowdown than would be required to show that faults on the previous branches

POWER GENERATION DURING CORE REFLOODING

uld cause cladding temperature and temperature-time limits to be exceeded if predicted al rates. The tree developed for this branch is shown in Figure 6.

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t for analysis underpredicts power generation during core reflooding.

olves two source terms-decay heat power generation and metal-water reaction energy 30 to 120 seconds after rupture. For the plant examined (temperatures at the time of ive power would result if the calculated values are low by 10 to 20%. Most calculations 120% of the ANS standard. Currently the conservatism of this value has not been usually calculated using the Baker-Just equations.⁴ Posttest examination of bundles ng Heat Transfer Projects^{5,6} to determine oxide thickness indicate that use of the iction of the metal-water reaction rate. For the time sequence considered in this become significant until temperatures are near the specified limit.

ased on a power generation rate of 1.2 times the ANS standard and a multiplier of 1.0 ent a fault on Branch 4.

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Branch 5--ANALYSIS OVERPREDICTS ENERGY REMOVAL RATES DURING REFLOODING

This branch of the tree considers those errors that could exist in the predicted energy removal rates such that limits are exceeded unexpectedly during core reflooding. The fault tree for Branch 5 is shown in Figure 7. Figure 7 shows the development of the tree for "analysis overpredicts reflooding flow rate".

Fig. 7 Tree development for analysis overpredicts energy removal rates during core reflooding.

The time period of interest for this fault is the same as noted in the discussion on power generation; the seconds. The actual duration is, of course, a function of plant design, temperature levels, and ECC system performance. Potential sources of errors in analysis, which are considered in the upper section of the tree, include methods for determining water level in the downcomer (the driving head for reflooding), the major elements of the steam binding pressure resistance to steam flow from the core to the break, and the potential for errors in the experimental base used for reflooding heat transfer coefficients.

A general understanding of the present method used to derive the predicted heat transfer coefficients is used in the development of the tree below the statement "analysis overpredicts heat transfer coefficients". In safety analysis, predicted values are simply heat transfer coefficients derived from PWR-FLECHT tests.⁵ The applicable PWR-FLECHT results for a design problem are selected by identifying the appropriate system and core conditions existing at the time of reflooding with specific consideration given to containment pressure, cladding temperature, water temperature in the low pressure system, and the reflooding rate. The reflooding rate is usually calculated by using the accumulator and low pressure injection system mass fraction of the flow into the bottom of the core which is assumed to exit the bundle (derived from FLECHT results). The reflooding rate is the most important variable to be considered in selecting the correct heat transfer coefficients from the FLECHT results and it is also the most difficult to predict. Because of the importance of reflooding rate, the tree is developed further in this area (Figure 8) to identify specific errors that may produce errors in the predictions of reflooding rate. All other sources of errors are consolidated in the discussion "PWR-FLECHT results not typical or misapplied".

ENERGY REMOVAL RATES DURING REFLOODING

the errors that could exist in the predicted energy removal rates such that temperature are reflooding. The fault tree for Branch 5 is shown in Figure 7. Figure 8 shows further predicts reflooding flow rate".

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ment for analysis overpredicts energy removal rates during core reflooding.

is fault is the same as noted in the discussion on power generation; that is, 30 to 120 se, a function of plant design, temperature levels, and ECC system performance. The hich are considered in the upper section of the tree, include methods for evaluating the ing head for reflooding), the major elements of the steam binding problem (system to the break), and the potential for errors in the experimental base used to obtain the

ment method used to derive the predicted heat transfer coefficients is useful in following statement "analysis overpredicts heat transfer coefficients". In safety calculations, the coefficients derived from PWR-FLECHT tests.⁵ The applicable PWR-FLECHT results for ing the appropriate system and core conditions existing at the time reflooding is initiated, tainment pressure, cladding temperature, water temperature in the lower plenum, and ally calculated by using the accumulator and low pressure injection system delivery rates, n of the core which is assumed to exit the bundle (derived from FLECHT results), and upper plenum to the break. The reflooding rate is the most important variable that must t transfer coefficients from the FLECHT results and it is also the most difficult to obtain. rate, the tree is developed further in this area (Figure 8) to identify specific events which flooding rate. All other sources of errors are consolidated in the discussion related to the al or misapplied".

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Fig. 8 Tree development for analysis overpredicts reflooding flow rate.

PWR-FLECHT RESULTS NOT TYPICAL OR MISAPPLIED

For the diamond "PWR-FLECHT results not typical or misapplied" only the factors which are believed of reducing the heat transfer coefficients by more than 10% are considered. For PWR designs, an error of 10 transfer coefficients may result in temperature predictions being unrealistically low and cause a fault on the temperatures at reflooding initiation are above 1600°F. With regard to heat transfer coefficients obtained from tests not being typical, the following two areas of uncertainty would have been identified if the tree had been more level:

- (1) Flow maldistribution resulting from blockage or radial temperature gradients in a large core produce heat transfer coefficients lower than observed for PWR-FLECHT
- (2) Heat transfer coefficients for zircaloy-clad rods at low reflooding rates may be lower than heat transfer coefficients obtained from PWR-FLECHT tests which used stainless steel-clad rods.

As noted in Section IV both areas are being considered for investigation by additional experimental work.

In terms of misapplication of the FLECHT results, a 10% overprediction in the value of the heat obtained using FLECHT data could result if a gross error exists in calculations of the containment pressure. If pressures are 15 to 20 psi higher than the actual pressures. Also for low reflooding, if the FLECHT heat transfer is used, the predicted coefficient may be high by more than 10%.

Analysis Overpredicts Reflooding Flow Rate

The subbranch of the tree for low core reflooding rates was shown in Figure 8. In calculations of the core the pressure drop from the core inlet to the core outlet plus the pressure drop from the core outlet back to the head of water in the downcomer. A 17% error in reflooding rate calculations would be necessary to cause a 10% error in heat transfer coefficients. The three diamonds on the right relate to the supply of water to the downcomer factor in the reflooding rate calculations. In order for a fault to occur, the predicted downcomer water height would have to be about 40% higher than the actual height.

Further development of the tree below the diamond "accumulators empty before predicted" would include errors shown on Branch 2. The magnitude of the acceptable difference between predicted and actual behavior is considerably less on this branch than on Branch 1. On this branch, the predicted behavior for a predicted accretion rate of about 80 ft³/sec is for the lower plenum downcomer and downcomer bypass region (a total volume of 2250 ft³) to be filled before the accumulators are emptied. Since the total volume of the accumulators is only about 2250 ft³, a bypass of the core during blowdown and a blowdown time 4 or 5 seconds longer than predicted could cause this error.

Once the downcomer is full, for ECC entrainment to affect the water height in the downcomer, more incoming LPI flow would have to be lost. The potential for an analysis error resulting in an overprediction is unlikely because in actuality two systems are available and the predictions are based on the assumption that one system is available.

The event in the diamond of Figure 8, "analysis underpredicts mass flow rate from vessel", would occur if the core outlet flow to the core inlet flow is about 17% higher than observed in the FLECHT tests. At present, correlations used for reflooding evaluations do not treat the mass and energy transfer process rigorously. Instead, time-dependent data are used in the calculations to account for the fraction of the core inlet mass flow which exits the vessel. Vendors indicate that the mass fraction out of the vessel as derived from the FLECHT experiments is higher than that for reactor cores because of the small size of the PWR-FLECHT bundle and the fact that a hot section of the PWR is used rather than an average section. The authors' opinion is that the FLECHT mass fraction of the core inlet flow which exits the vessel represents a worst case value for predicting reflooding rates.

The three elements which could cause the predicted pressure drop between the upper vessel plenum and the annulus to be lower than the actual drop are identified in the diamonds on the left of Figure 8. Containment pressure predicted could have been identified by a fourth diamond; however, for a predicted pressure of 60 psia a 35% error is required to cause a fault. Each of the faults for the three diamonds under high system pressure drops is discussed in the following discussions.

- (1) Analysis Underpredicts Effect of Plugging by ECC Water in Pipes. For plant designs in which ECC is injected into the piping, the actual pressure drops will be complex functions of momentum and energy exchanges between the ECC and the steam flowing in the lines. Two-phase pressure drops of real significance in terms of core reflooding rates are most likely to occur during accumulator injection

MISAPPLIED

is not typical or misapplied" only the factors which are believed potentially capable more than 10% are considered. For PWR designs, an error of 10 to 20% in the heat transfer predictions being unrealistically low and cause a fault on this branch provided the 1600°F. With regard to heat transfer coefficients obtained from the PWR-FLECHT, areas of uncertainty would have been identified if the tree had been developed to one

from blockage or radial temperature gradients in a large core may be lower than observed for PWR-FLECHT

alloy-clad rods at low reflooding rates may be lower than heat transfer coefficients from FLECHT tests which used stainless steel-clad rods.

considered for investigation by additional experimental work.

FLECHT results, a 10% overprediction in the value of the heat transfer coefficient would mean a gross error exists in calculations of the containment pressure such that calculated actual pressures. Also for low reflooding, if the FLECHT heat transfer correlations are in error more than 10%.

reflooding rates was shown in Figure 8. In calculations of the core reflooding rate, the core outlet plus the pressure drop from the core outlet back to the downcomer is considered. A 17% error in reflooding rate calculations would be necessary to cause a 10% error in the diamonds on the right relate to the supply of water to the downcomer which is a key factor in order for a fault to occur, the predicted downcomer water height would have to be

the diamond "accumulators empty before predicted" would include all the potential for the acceptable difference between predicted and actual behavior would, however, be small. On this branch, the predicted behavior for a predicted accumulator discharge from the downcomer and downcomer bypass region (a total volume of about 1750 ft³) to the accumulator. Since the total volume of the accumulators is only about 2250 ft³, complete ECC shutdown time 4 or 5 seconds longer than predicted could cause this fault.

entrainment to affect the water height in the downcomer, more than 25% of the water height. The potential for an analysis error resulting in an overprediction of LPIS delivery is small and available and the predictions are based on the assumption that one system fails.

"analysis underpredicts mass flow rate from vessel", would occur if the ratio of the predicted to observed is 17% higher than observed in the FLECHT tests. At present, calculational methods do not model the mass and energy transfer process rigorously. Instead, time-dependent FLECHT is used for the fraction of the core inlet mass flow which exits the vessel. At present, several assumptions for the vessel as derived from the FLECHT experiments is higher than expected in larger PWR-FLECHT bundle and the fact that a hot section of the PWR core was simulated. The opinion is that the FLECHT mass fraction of the core inlet flow which exits the vessel is higher than observed at reflooding rates.

the predicted pressure drop between the upper vessel plenum and the downcomer is identified in the diamonds on the left of Figure 8. Containment pressure lower than observed at fourth diamond; however, for a predicted pressure of 60 psia a 35% error would be necessary. For the three diamonds under high system pressure drops is summarized in the

Plugging by ECC Water in Pipes. For plant designs in which ECC is used, pressure drops will be complex functions of momentum and energy balance for the steam flowing in the lines. Two-phase pressure drops of real fluids and plugging rates are most likely to occur during accumulator injection

when ECC flow rates are high. The authors' opinion is that if the angle of the ECC injection line is such that the ECC momentum is directed down the pipe the plugging problem would be less than for an ECC line with a 90 degree angle to the inlet piping. Although perhaps not significant, the assumption that one of the two LPIS systems fails to operate, which is customarily assumed in safety analysis, would not be conservative in calculating core reflooding rates provided the downcomer is filled by the accumulator.

- (2) Analysis Underpredicts Energy or Mass Transfer from Steam Generator Secondary System to Primary Loop. During core reflooding the fluid from the secondary side of the steam generator has the potential to transfer energy, and in the event of a tube leakage also mass, to the primary system. Transfer of either mass or energy can add significantly to the pressure drop from the upper plenum to the downcomer annulus. Unless the secondary side has depressurized before reflooding is initiated, significant errors in reflooding rates would occur if energy transfer processes are not properly taken into account. Current predictions, which are based on the secondary side of the steam generator being at normal operating temperatures at reflooding initiation, are considered to account for all energy transfer processes. If only a few tubes leak, mass transfer processes can also significantly add to the error in the calculated reflooding rates. To provide a 17% reduction in reflooding rates (at 1.5 in./sec), a tube break area of about 0.003 ft² would be required. Present predictions are based on the assumption that none of the tubes fail.
- (3) Analysis Underpredicts Pump Resistance. During the reflooding phase of the accident the pump is subjected to a superheated steam flow at a rate which considerably exceeds the pump capacity. In this situation the pump is simply a resistance. It is, however, the component with the highest resistance in the operating loop and would cause unacceptable reflooding rates if actual values of pump resistance exceed predicted values by about 50%. Potential sources of errors in evaluating pump resistance include:
 - (1) Error in calculating pump speed
 - (2) Application of an inappropriate pump characteristic curve
 - (3) Compressibility effects not taken into account.

Current licensing criteria require the assumption of a locked rotor resistance for the pump when ECC performance is being evaluated. This assumption provides a resistance about twice as high as would be obtained on the basis of the predicted pump speed.

SUMMARY OF FAULT TREE APPLICATION

The following discussion summarizes the example tree developed for a PWR and shows how the fault tree maps to a BWR.

Example Fault Tree for PWR's

Figure 9 shows a composite of the tree developed in this section. Because of space limitations the houses which state assumptions and some of the boxes which were included for explanatory notes have been removed. The tree record of the logic processes used to determine systematically the sources of errors in current predictive analysis could lead to the conclusion that cladding temperature and embrittlement limits during an LOCA would not be exceeded, they may be exceeded. The diamonds on the composite tree which characterize events or potential faults considered to be significant in specifying ECC design requirements have been enclosed by double lines.

The double lines should not be considered as an indication that present ECC system designs are inadequate; they are intended to highlight variables or items within the analysis which exhibit relatively high influence. For the particular example considered in this example, faults on Branches 1 and 2, "analysis in error-limits violated prior to core reflooding", are to be unlikely and are not enclosed by double lines. Faults on Branches 1 and 2, involving blowdown heat transfer, are also shown on Branch 3. On Branch 3 acceptable errors in the two variables are considerably less than for faults on Branch 1. For this reason, factors contributing to core reflooding being delayed in time are not enclosed on Branch 2, but are enclosed on Branch 3. As noted in the preceding discussions the most demanding requirements for analysis errors on inputs related to core reflooding delayed in time would be identified by development of the diamond "analysis overpredicts time to empty accumulators" on Branch 5. On this branch, a few seconds error in blowdown time with ECC bypass could result in a fault condition.

opinion is that if the angle of the ECC injection line is such that the pipe the plugging problem would be less than for an horizontal piping. Although perhaps not significant, the assumption of steady-state operation, which is customarily assumed in safety analysis, may overestimate the reflooding rates provided the downcomer is filled by the

Transfer from Steam Generator Secondary System to Primary
The secondary side of the steam generator has the potential for tube leakage also mass, to the primary system. Transfer of mass from the secondary to the primary is dependent on the pressure drop from the upper plenum to the primary side has depressurized before reflooding is initiated. If energy transfer processes are not properly taken into account, the analysis based on the secondary side of the steam generator being reflooded initiation, are considered to account for all energy transfer processes can also significantly add to the error. A 17% reduction in reflooding rates (at 1.5 in./sec), a tube failure. Present predictions are based on the assumption that

During the reflooding phase of the accident the pump is operating at a rate which considerably exceeds the pump capacity. In this case it is, however, the component with the highest resistance in determining the reflooding rates if actual values of pump resistance are used. Potential sources of errors in evaluating pump resistance

pump characteristic curve

is taken into account.

Assumption of a locked rotor resistance for the pump when ECC injection provides a resistance about twice as high as would be the case at full speed.

The fault tree developed for a PWR and shows how the fault tree might be applied

in this section. Because of space limitations the houses which were used to group the events included for explanatory notes have been removed. The tree is primarily a qualitative analysis of the sources of errors in current predictive analysis methods which are used to determine the failure and embrittlement limits during an LOCA would not be exceeded when, in the composite tree which characterize events or potential analysis faults, the events which are considered to be important have been enclosed by double lines.

The indication that present ECC system designs are inadequate; they are simply not adequate for the analysis which exhibit relatively high influence. For the particular plant design under consideration, the "analysis in error-limits violated prior to core reflooding", are considered as faults on Branches 1 and 2, involving blowdown heat transfer and stored energy. The errors in the two variables are considerably less than for the same errors in the analysis of core reflooding being delayed in time are not enclosed by double lines in the preceding discussions the most demanding requirement for minimum delay in time would be identified by development of the tree below the "Faults" on Branch 5. On this branch, a few seconds error in the calculated reflooding condition.

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Fig. 9 Summary fault tree for analysis of PWR design basis accident.

On the composite fault tree all of Branch 4 has been reduced to a single double-lined diamond "analysis metal-water reaction or decay heat power generation". The double line has been added because power generation correlation with maximum temperature and time-at-temperature when heat transfer coefficients are small.

On Branch 5, which identifies events that can reduce energy removal rates during reflooding, five of the eight enclosed in double lines. As discussed in the preceding text, the PWR-FLECHT results are among the key factors in reflooding heat transfer rates and must be carefully used in evaluating ECC performance. Another important factor is the overprediction of the time for the accumulators to empty such that in reality the low pressure injection system downcomer. Because of the relatively small system pressure drops involved, all three of the diamonds under the "analysis underpredicts pressure drop from upper plenum to annulus" have been enclosed in double lines.

Discussion of Factors Affecting Selection of BWR-ECCS Design Requirements Using Example "Tree"

Although a fault tree has not been constructed for a BWR system, the events on the PWR tree would be expected to be in similar form on the BWR tree. Since the BWR system uses core spray and reflooding sequenced in time, the top event of the tree might be changed from "prior to reflooding" and "during reflooding" to "prior to ECC action" and "during ECC action". If this approach were taken, Branch 1 of the tree could remain essentially the same but the top would have "analysis underpredicts temperature-limit exceeded prior to predicted time to spray initiation". The top of Branch 2 might be changed to "analysis underpredicts time to spray initiation". The elements on Branch 2 would also have to be changed to reflect the following analysis faults:

- (1) Analysis underpredicts spray system delivery
- (2) System effects delay spray initiation
 - (a) Analysis underpredicts blowdown time
 - (b) Analysis overpredicts core spray rate--core steam flow distorts spray distribution until the end of blowdown.

The top of Branch 3 would probably be changed to "analysis underpredicts temperature at spray initiation" and the branch would include faults similar to those of Branches 1 and 2. Branch 4 would remain the same with the exception that "during reflooding" would be changed to "during ECC action".

The top of Branch 5 would require use of "ECC action" instead of "reflooding" and the branch would be moved below the top "or" gate on "BWR-FLECHT results" much like the one for "PWR-FLECHT results". The remainder of Branch 5 below the first "or" gate would change considerably. Perhaps the event "analysis underpredicts core spray rate" would be replaced with "analysis underpredicts core spray rate to hot bundles" and the tree would be developed further for this event. Another event would also be defined for input to the top "or" gate and this event would be "analysis underpredicts time to initiate core reflooding". In a BWR, a steam binding problem is not expected, but for high power density plants the problem of core reflooding could be critical because the spray system only slows the rate of the cladding temperature rise. The time to start core reflooding is shown by the performance studies in the next section.

III. PERFORMANCE OF LIGHT-WATER REACTOR ECC SYSTEMS

This section presents an evaluation of the sensitivity of the performance of ECC systems for BWR's and PWR's over a range of possible conditions existing at the time ECC action is initiated. Calculated cladding temperatures and embrittlement due to metal-water reaction are presented for various initial cladding temperatures and ECC availabilities. These calculations are utilized in conjunction with families of temperature-time curves characterizing LOCA behavior to establish the time to failure. It would be required to assure that temperature or embrittlement limits are not violated.

For a PWR with reflooding as the ECC action, performance maps are presented which relate the maximum cladding temperature and embrittlement to the temperature at reflooding initiation for a wide range of nominal core reflooding rates. The influence of other parameters which affect the maximum cladding temperature during ECC action are also presented. This discussion covers the sensitivity of temperature and embrittlement to initial power, containment backpressure, cladding thickness, metal-water reaction rate, and delay in reflooding time. To assist in relating the performance maps to actual plant characteristics that control reflooding rates in PWR's, calculational results are presented showing reflooding rates for a break in the four-loop plant considered previously in this paper. The sensitivity of the reflooding rate in the reference plant to water plugging in the inlet piping, pump resistance, and depletion of the accumulators before the downcomer is reflooded is shown.

A performance map for a BWR ECC system (spray and reflooding) is presented in essentially the same form as for characterizing PWR ECC systems (that is, the maximum cladding temperature and embrittlement are related to the time to spray initiation). The time-to-reflood-initiation is the primary variable for the BWR-ECC performance map, whereas the reflood rate is the primary variable for the PWR-ECC performance map.

Cladding embrittlement values are also shown on the PWR and BWR performance maps in terms of percent embrittlement for a given maximum cladding temperature. The percent embrittlement describes the percent of the original cladding that has been converted to ZrO_2 and α -zirconium. Embrittlement of zircaloy has been noted to be a function of the oxygen content of the material.⁷ Oxygen, in addition to forming a brittle layer of ZrO_2 on the outer surface of the cladding, diffuses through the oxide layer to form a layer of brittle oxygen-induced α -zirconium. Photomicrographs of reactor zircaloy samples⁷ were used to establish that a factor of approximately 2.2 is required to correlate ZrO_2 thickness to the total thickness of $\text{ZrO}_2 + \alpha$ -zirconium. Thus, the percent embrittlement is calculated directly from the predicted ZrO_2 thickness by assuming oxidation occurs on only the outside surface of the cladding. A generally accepted value for the percent of cladding that can be oxidized and still survive rod quenching is 17 to 18%.^{8,9} These percentages correspond to an embrittlement of approximately 40% ($2.2 \times 18\%$) of the original cladding thickness. At 40% embrittlement, a radial load of 80 pounds per square inch on a sample was found to cause brittle failure.⁷

PRESSURIZED WATER REACTOR ECCS PERFORMANCE

The selection process for PWR-ECC system design requirements discussed in previous sections involved use of temperature-time curves portraying system LOCA response for the complete range of break sizes and locations in conjunction with system response data after ECC action had been taken. This section presents calculational methods and results of the study using these methods to provide PWR system response after reflooding is initiated. The results are expected to be used in the overall process of selecting ECC system design requirements. The results presented can be used to evaluate the impact of the errors in items identified on the fault tree in the previous section as having significant effect on reflooding end rates.

Source of Heat Transfer Coefficients for Study

As discussed in previous sections, evaluation of PWR system performance during core reflooding involves application of heat transfer coefficients derived from the PWR-FLECHT test program.⁵ The PWR-FLECHT program provided steady state data for a wide range of reactor conditions. Local cladding surface temperatures were measured in PWR-FLECHT tests and heat transfer coefficients based on the saturation temperature as the reference temperature were calculated. The PWR-FLECHT experimental program included reflooding experiments on 12-ft electrically heated rods (both 49- and 100-rod bundle configurations). The rod diameter and spacing were typical of those of a PWR. A filament down the center of each rod was wound to provide a chopped-cosine axial power profile. The filament was surrounded by boron nitride which served as an electrical insulator. The results used in the following discussion were obtained from stainless steel-clad rods.

Figure 10 is a typical example of heat transfer coefficients obtained at five axial locations along a rod in a PWR-FLECHT test. The data on which these coefficients are based were obtained during an experiment which employed a temperature of 2000°F and which employed a reflooding rate of 1 in./sec. Even though the heat transfer coefficients are significantly lower at elevations higher than the 6-ft elevation (midplane), the maximum temperatures usually occur at the midplane because of the cosine power distribution. Heat transfer coefficients for the midplane were used in the study.

Application of Heat Transfer Data to Obtain Nuclear Rod Performance Using the HEAT-1 Code

The HEAT-1 computer code¹⁰ was used to solve the transient energy conduction equation for one axial location (the midplane) on a simulated nuclear rod. The code represents the rod with 33 radial nodes from the rod center line to the outer surface of the cladding. An implicit finite-difference solution is used. A gap conductance of 500 Btu/hr-ft²-°F was used in the study. Energy generated in the zircaloy cladding due to metal-water (zirconium-oxygen) reaction was calculated through the Baker-Just equation⁴ reduced by 25%.

Input to the code included the initial and decay power generation rates in the UO_2 , physical properties of zircaloy¹¹, ZrO_2 thickness at transient initiation, metal-water reaction equation multiplication factor, cladding temperature at time of flooding initiation, fluid saturation temperature, and the surface heat transfer coefficients. The appropriate boundary conditions at the time core reflooding was initiated were obtained by setting the rod materials temperatures to the selected initial cladding temperature to be investigated (or 1800°F if initial temperatures were greater than 2000°F), specifying initial power at time of reflooding and a constant surface heat transfer coefficient of 2 Btu/hr-ft²-°F, and then calculating cladding surface temperature to continue. When the calculated cladding temperature reached the selected temperature value, the programmed power decay was applied and the appropriate heat transfer coefficient from the PWR-FLECHT bundle midplane was substituted for the constant value.

system (spray and reflooding) is presented in essentially the same form used for maximum cladding temperature and embrittlement are related to the temperature at which the primary variable for the BWR-ECC performance map, whereas the reflood performance map.

shown on the PWR and BWR performance maps in terms of percent embrittlement. The percent embrittlement describes the percent of the original cladding thickness remaining. Embrittlement of zircaloy has been noted to be a function of the oxygen content, forming a brittle layer of ZrO_2 on the outer surface of the cladding, at high temperatures to form a layer of brittle oxygen-induced α -zirconium. Photomicrographs taken of zircaloy rods show that a factor of approximately 2.2 is required to correlate ZrO_2 thickness to the percent embrittlement is calculated directly from the predicted ZrO_2 thickness on the outside surface of the cladding. A generally accepted value for the percentage of rod quenching is 17 to 18%.^{8,9} These percentages correspond to an embrittlement of 17 to 18% of the original cladding thickness. At 40% embrittlement, a radial load of 80 pounds per inch of

PERFORMANCE

Design requirements discussed in previous sections involved use of a family of PWR rod bundle response for the complete range of break sizes and locations in conjunction with the PWR rod bundle design. This section presents calculational methods and results of a sensitivity study of rod bundle response after reflooding is initiated. The results are expected to be helpful in design requirements. The results presented can be used to evaluate the importance of the rod bundle design in the previous section as having significant effect on reflooding energy removal.

Simulation of PWR system performance during core reflooding involves application of heat transfer coefficients to the rod bundle. The PWR-FLECHT program provided steady state reflooding heat transfer coefficients. Local cladding surface temperatures were measured in PWR-FLECHT tests, and local rod bundle temperature as the reference temperature were calculated. The PWR-FLECHT experiments on 12-ft electrically heated rods (both 49- and 100-rod bundles were used). The rods were of a PWR. A filament down the center of each rod was wound to produce a rod bundle surrounded by boron nitride which served as an electrical insulator. The test was obtained from stainless steel-clad rods.

Heat transfer coefficients obtained at five axial locations along a rod during a reflooding experiment are based on the heat transfer coefficients obtained during an experiment which started at a reflooding rate of 1 in./sec. Even though the heat transfer coefficient is based on the 6-ft elevation (midplane), the maximum temperatures usually occurred at the top of the rod bundle. Heat transfer coefficients for the midplane were used in the study discussed.

Nuclear Rod Performance Using the HEAT-1 Code

The HEAT-1 code was used to solve the transient energy conduction equation for one axial location (core midplane). The code represents the rod with 33 radial nodes from the rod center line to the outer surface. A gap conductance of 500 Btu/hr-ft²-°F was used in the code. The gap due to metal-water (zirconium-oxygen) reaction was calculated through use of the

decay power generation rates in the UO_2 , physical properties of UO_2 and zirconium, metal-water reaction equation multiplication factor, cladding temperature at the top of the rod bundle, and the surface heat transfer coefficients. The appropriate initial conditions were obtained by setting the rod materials temperatures 200°F below the design temperature (or 1800°F if initial temperatures were greater than 2000°F), applying the rod bundle a constant surface heat transfer coefficient of 2 Btu/hr-ft²-°F, and allowing the rod bundle to continue. When the calculated cladding temperature reached the selected reflooding temperature, decay was applied and the appropriate heat transfer coefficients from the rod bundle were used for the constant value.

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Fig. 10 PWR-FLECHT nominal heat transfer coefficients.

The code output was the radial temperature distribution across the rod and the percent of cladding embrittlement functions of time after initiation of reflooding.

Figure 11 presents examples of the results calculated by the HEAT-1 code. These particular examples of the cladding temperature-time behavior and the percent embrittlement of the cladding at various times after reflooding of calculations based on an initial cladding temperature at time of reflooding of 2000°F and various core flooding examples, the cladding is subjected to high temperatures for a relatively long time.

To provide a summary of core performance during reflooding, a map, shown in Figure 12, was constructed maximum cladding temperature to the temperature at reflooding initiation. Each point on the map is taken from a transient (Figure 11 shows typical curves) characterized by the flooding rate, initial cladding temperature, maximum temperature reached during the transient, and the percent of embrittled cladding ($\text{ZrO}_2 + \alpha\text{Zr}$) which will occur during cooldown. The initial cladding temperature refers to the cladding temperature at the core hot spot at the time the loop is filled and flooding of the core begins.

Results shown on the map are for a peak initial power of 1.22 kW/ft at reflood initiation, the PWR-FLECHT rate, an upper plenum pressure of 60 psia , an inlet water subcooling of 142°F (150°F water temperature), a multiplier of 0.75 in the Baker-Just metal-water energy equation, and an initial ZrO_2 thickness of 0.1 mil .

To characterize the effects of reflooding rate, six different reflooding rates, including both constant and variable, are shown in Figure 12. For example, this figure shows that if the ECC system delivers coolant to the core at a reflooding rate of 1 in./sec and if the core midplane temperature at the time of reflooding is 1500°F , the maximum temperature will be about 19% embrittled. Thus, a temperature margin of safety of 100°F and an embrittlement margin of 21% exist for this example.

Average Heat Transfer
Coefficients on Central Rod

Test 6553

$V = 1 \text{ in./sec}$

$T_i = 2000^\circ\text{F}$

$P = 60 \text{ psia}$

$\Delta T_s = 148^\circ\text{F}$

$Q_i = 1.22 \text{ kW/ft}$

4 Foot

6 Foot

8 Foot

10 Foot

80 100 120 140 160 180 200 220
Time (sec)

ANC-B-864-H

10 PWR-FLECHT nominal heat transfer coefficients.

temperature distribution across the rod and the percent of cladding embrittlement both as a function of time and position along the rod.

Results calculated by the HEAT-1 code. These particular examples of the core midplane temperature distribution and the percent embrittlement of the cladding at various times after reflooding are for a series of initial temperatures at time of reflooding of 2000°F and various core flooding rates. For these conditions, the results are shown for a relatively long time.

Performance during reflooding, a map, shown in Figure 12, was constructed to relate the initial temperature at reflooding initiation. Each point on the map is taken from a thermal analysis of a rod element characterized by the flooding rate, initial cladding temperature, maximum cladding temperature, and the percent of embrittled cladding ($\text{ZrO}_2 + \alpha\text{Zr}$) which will exist following reflooding. The map refers to the cladding temperature at the core hot spot at the time the lower plenum of the core is reflooded.

For a peak initial power of 1.22 kW/ft at reflood initiation, the PWR-FLECHT power decay curve, an inlet water subcooling of 142°F (150°F water temperature), a multiplication factor of 1.0, and an initial ZrO_2 thickness of 0.1 mil .

For a flooding rate, six different reflooding rates, including both constant and variable rates, are shown. It shows that if the ECC system delivers coolant to the core at a reflooding rate of 1 in./sec and the initial temperature at the time of reflooding is 1500°F , the maximum temperature will be 2200°F and the cladding will be embrittled. Thus, a temperature margin of safety of 100°F and an embrittlement margin of

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Fig. 11 Calculated PWR temperature-time behavior for various flooding rates.

An interesting characteristic revealed by Figure 12 is that the calculated lines of constant percent ϵ are approximately horizontal. For the type of temperature transients investigated, embrittlement was almost wholly a function of maximum temperature. Although the time at temperature is important, variations in the time to reach the maximum temperature were generally overshadowed by the dependence of the exponential temperature rise on the reaction rate coefficient.

As an aid in estimating the effect of several other parameters, results from other sensitivity studies are presented in Table I. This table shows the percent change in the cladding temperature rise and embrittlement for two points on Figure 11. The points are for 2000 and 1600°F initial temperatures for a flooding rate of 6 in./sec for 4 seconds followed by a flooding rate of 1 in./sec. By utilizing the information in Table I and the curves of Figure 12, new performance maps could be constructed.

The largest changes occur in the 2000°F column because the metal-water reaction energy is more significant at this temperature than for the 1600°F temperature at reflood initiation. The parameter which caused the greatest effect was the reaction energy multiplying factor. Next, in the order of importance, are: (a) metal-water reaction energy multiplying factor; (b) the time at temperature; and (c) ZrO_2 thickness.

Another important parameter affecting the relationship between the temperature at the time of reflooding and the maximum temperature for a given reflooding rate is containment pressure. The containment pressure affects the rate of temperature rise as well as the heat transfer for a specific flooding rate. As an example of this effect, if the core pressure is 25 psia

ed PWR temperature-time behavior for various flooding rates.

by Figure 12 is that the calculated lines of constant percent embrittlement are temperature transients investigated, embrittlement was almost wholly a function of the temperature. At the temperature where the maximum temperature is important, variations in the time to reach the maximum temperature have a significant effect on the reaction rate coefficient.

Several other parameters, results from other sensitivity studies are presented in Table I. The effect of varying the initial temperature rise and embrittlement for two points on Figure 12. These points are for a flooding rate of 6 in./sec for 4 seconds followed by a flooding rate of 1 in./sec. The curves of Figure 12, new performance maps could be constructed.

100°F column because the metal-water reaction energy is more significant at this temperature. The parameter which caused the greatest effect was the initial temperature rise. (a) metal-water reaction energy multiplying factor; (b) the time at which reflooding

the relationship between the temperature at the time of reflooding initiation and the flooding rate is containment pressure. The containment pressure affects the flooding rate as well as the reaction rate. As an example of this effect, if the core pressure is 25 psia instead of 60 psia

Fig. 12 PWR-ECC performance map obtained through use of PWR-FLECHT data.

and the reflooding rate is 1 in./sec, a 1500°F temperature at reflood initiation would result in a maximum temperature above the value obtained from Figure 12.

Calculation of Reflooding Rates

A reflood computer program, FLOOD1, recently developed by Aerojet Nuclear Company for the AEC, calculates average reflooding rates. This program uses a relatively straightforward approach in which the loop pressure drop is accounted for in the continuity of mass equation. The primary system is modeled as a resistance network and, therefore, includes system flow oscillations. The analysis starts at the time the water level reaches the bottom of the core. The water is assumed to flow into the downcomer region with no interaction with the steam flow. The liquid carryover fraction, the ratio of the mass leaving the core to the mass entering the core, is an input variable normally set to 5% until the liquid height in the core reaches 6 inches. When the liquid height in the core reaches 6 inches the fraction is assumed to increase until it equals 0.80 at a height of 12 inches. For heights above 12 inches, a constant fraction equal to 0.80 is used. The fraction, which is based on data obtained from the PWR-FLECHT experiments, has been corrected to account for the fact that actual fuel rod temperatures at the core entrance may be higher than those in the PWR-FLECHT experiment. The program can be performed on the carryover fraction by variation of input parameters. In the FLOOD1 analytical model,

ECC performance map obtained through use of PWR-FLECHT data.

500°F temperature at reflood initiation would result in a maximum temperature 5000°F

FLOOD1, recently developed by Aerojet Nuclear Company for the AEC has been used to program uses a relatively straightforward approach in which the loop pressure drops are equation. The primary system is modeled as a resistance network and, therefore, does not starts at the time the water level reaches the bottom of the core. The ECC injected is on with no interaction with the steam flow. The liquid carryover fraction, expressed as mass entering the core, is an input variable normally set to 5% until the liquid height in height in the core reaches 6 inches the fraction is assumed to increase exponentially. For heights above 12 inches, a constant fraction equal to 0.80 is used. This carryover from the PWR-FLECHT experiments, has been corrected to account for the fact that entrance may be higher than those in the PWR-FLECHT experiment. Parameter studies by variation of input parameters. In the FLOOD1 analytical model, the annulus and

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TABLE I

SENSITIVITY STUDY SHOWING EFFECT OF PARAMETER CHANGES ON RESULTS IN FIGURE 1

Parameter Value	Temperature of 2000°F at time of reflooding		Temperature of at time of re
	Temperature Rise (% Change)	Cladding Embrittlement (% Change)	Temperature Rise (% Change)
Baker-Just multiplication factor			
= 0.5	-21	-21	- 6
= 1.0	55	75	9
Initial power (kW/ft) = 1.0	-56	-50	-45
= 1.4	106	170	53
ZrO ₂ thickness (in.) = 0.001	-13	- 9	- 9
= 0.00001	1	2	2
7-sec delay in time to initiate reflooding[a]	-16	-16	-10

[a] Delays in flooding initiation result in a reduced temperature rise at any given temperature at the time of flooding initiation because the decay power is decreased.

core bypass regions are viewed conceptually as one leg of a manometer in which water levels are computed, the other leg. Pressure drops due to flow through the unbroken loop to the inlet annulus are balanced by the difference in water levels in the manometer legs.

Calculations of reflooding rates using FLOOD1 oversimplify the complex events which occur during the reflooding of a BWR core following an LOCA. Use of this code in calculations of average reflooding rates is possible even though the dynamic effects (e.g., pressure oscillations) are not correctly predicted because of the generalized treatment of the controlling phenomena.

Figure 13 shows the reflooding rates derived from the application of this FLOOD1 code to a four-loop PWR. The results are presented to show the effect of several different assumptions which could characterize system resistances and flow behavior. Assumptions common to all of the results shown on Figure 13 are a system pressure of 50 psia, a pressure injection system operational, and a fluid temperature of 500°F in the steam generator secondary side. Curve 1 shows the reflooding rate as a function of time based on the assumption that the accumulators empty 14 seconds after reflooding initiation, no plugging effect of ECC and with the pump resistance corresponding to a pump operating with a flow rate between free wheeling and locked rotor incompressible flow values. The maximum value of the reflooding rate from three seconds is 7 in./sec. Curve 2 is based on the same input parameter as Curve 1 except the pump resistance is the locked rotor value. Curve 3 shows the effect of assuming that while the accumulators are discharging, the cold leg loop is plugged with water. Curve 4 shows the effect of accumulators emptying prior to reflood initiation. In the pressure injection system is the only source of fluid in the downcomer.

To use the reflooding curves on Figure 13 to determine approximate relationships between initial temperatures on Figure 12, the longer term rates are treated as steady state values. Curve 1 (1/2 stopped pump) would correspond to a constant reflooding rate of 1.75 in./sec. Curve 2 (fully stopped pump) would correspond to a rate of about 1.40 in./sec and Curve 3 (pipe plugging) would correspond to a rate of about 1.3 in./sec. The most conservative (accumulators depleted prior to reflooding) would be equivalent to a constant reflood rate of about 0.7 in./sec.

BOILING WATER REACTOR ECCS PERFORMANCE

This section provides performance information for BWR cores following ECC system actuation. Information is provided for use in selecting ECC system design requirements and identifying important parameters affecting ECC system performance.

TABLE I

WING EFFECT OF PARAMETER CHANGES ON RESULTS IN FIGURE 12

Temperature of 2000°F at time of reflooding		Temperature of 1600°F at time of reflooding	
Temperature Rise (% Change)	Cladding Embrittlement (% Change)	Temperature Rise (% Change)	Cladding Embrittlement (% Change)
-21	-21	- 6	- 9
55	75	9	12
-56	-50	-45	-61
106	170	53	101
-13	- 9	- 9	- 8
1	2	2	2
-16	-16	-10	-15

result in a reduced temperature rise at any given temperature at re-
decay power is decreased.

as one leg of a manometer in which water levels are computed, the core being the
h the unbroken loop to the inlet annulus are balanced by the differential height of

FLOOD1 oversimplify the complex events which occur during the reflooding phase of
f average reflooding rates is possible even though the dynamic effects (for example,
ase of the generalized treatment of the controlling phenomena.

derived from the application of this FLOOD1 code to a four-loop PWR system. Results
different assumptions which could characterize system resistances and accumulator
he results shown on Figure 13 are a system pressure of 50 psia, one low pressure
temperature of 500°F in the steam generator secondary side. Curve 1 shows the
the assumption that the accumulators empty 14 seconds after reflood initiation with
ump resistance corresponding to a pump operating with a flow resistance midway
mpressible flow values. The maximum value of the reflooding rate from about one to
on the same input parameter as Curve 1 except the pump resistance is increased to the
of assuming that while the accumulators are discharging, the cold leg of the operating
he effect of accumulators emptying prior to reflood initiation. In this case, the low
f fluid in the downcomer.

Figure 13 to determine approximate relationships between initial and maximum
na rates are treated as steady state values. Curve 1 (1/2 stopped pump) would
ooding rate of 1.75 in./sec. Curve 2 (fully stopped pump) would correspond to a rate
gging) would correspond to a rate of about 1.3 in./sec. The most significant effect
ould be equivalent to a constant reflood rate of about 0.7 in./sec.

PERFORMANCE

Information for BWR cores following ECC system actuation. Information is provided for
ts and identifying important parameters affecting ECC system performance.

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Fig. 13 PWR reflooding rates-parameter study using FLOOD1.

Source of Heat Transfer Coefficient for Study

The primary source of experimental information on BWR-ECC systems (spray and reflooding) has been the BWR-FLECHT⁶ program. The basic test assembly for this program employed 49-rod bundles of stainless steel heated fuel rod simulators. The rods were 12 feet long, located on typical BWR spacing, and enclosed by a shroud. Results were obtained during both bottom flooding and top spray injection for various bundle powers, flow rates, and initial temperatures. The bottom flooding test data were applied in a manner similar to that discussed in the previous section. The PWR-FLECHT data. Because of the relatively small convective heat transfer coefficients associated with spray, large radial power and resulting temperature gradients within a BWR bundle, the heat transfer coefficients for spray were calculated differently than those for the reflooding experiments. The data reduction process included a radiative analysis to determine rod-to-rod and rod-to-canister radiant heat fluxes. These heat fluxes were then used in conjunction with the total heat fluxes to obtain local convection heat transfer coefficients. Correlations of convection coefficients that are commonly used as input to multirod codes to calculate the performance of nuclear rod bundles during spray cooling.

Application of BWR-FLECHT Results to Simulate BWR Core Performance

The MOXY code¹³ was used for conducting the ECC performance studies in this section. This code employs a bundle description that describes heat transfer by thermal radiation, conduction, and convection, and heat generation by fission and by fission product decay. For the evaluation performed in conjunction with this paper, the MOXY code was run under typical BWR operating conditions, and blowdown heat transfer coefficients and power decay were provided. For the calculations, the spray system was actuated 40 seconds after blowdown was initiated. At this time the BWR-FLECHT

R reflowing rates--parameter study using FLOOD1.

Information on BWR-ECC systems (spray and reflooding) has been obtained from the assembly for this program employed 49-rod bundles of stainless steel-clad, electrically heated, 12 feet long, located on typical BWR spacing, and enclosed by a shroud. Experimental data were applied in a manner similar to that discussed in the previous section for spray cooling. The data reduction process included a radiation calculation to determine heat fluxes. These heat fluxes were then used in conjunction with individual rod heat transfer coefficients. Correlations of convection coefficients thus derived^{1,2} are used to calculate the performance of nuclear rod bundles during spray cooling.

Rate BWR Core Performance

During the ECC performance studies in this section. This code employs models of a rod bundle, radiation, conduction, and convection, and heat generation by metal-water reaction. In conjunction with this paper, the MOXY code was initialized at shutdown. Down heat transfer coefficients and power decay were provided as input. For all times after shutdown was initiated. At this time the BWR-FLECHT heat transfer

convection correlations were used to provide the thermal response for spray-on conditions. A representative transfer behavior for the accident is shown in Figure 14 in which a base case and one conservative variation, contributed by flashing of the fluid in the lower plenum, are presented. The heat transfer characteristics, steady and the decay heat function are the only inputs required in a typical MOXY calculation. The physical geometry, metal-water reaction contribution, and other pertinent features are incorporated into the code.

Fig. 14 Two case assumptions of BWR-LOCA heat transfer coefficient response.

The calculated transient bundle responses, as represented by the temperature-time responses of three rods presented in Figure 15. Only top spray cooling has been used in these calculations because the temperature transfer is to be terminated when the core becomes flooded by liquid accumulated in the lower plenum. Because of different designs, pump sizes, and assumptions of liquid inventory remaining at the end of blowdown, the time to be terminated is considerably different, and has been left as an independent variable. The curves in Figure 15 represent typical output. The sensitivity of the calculated bundle temperature response has been investigated separately on a parametric basis. Power levels, decay power curves, metal-water reaction rates, and other variables all are important in the calculations.

An array of curves, such as shown in Figure 15, was generated by varying blowdown heat transfer rates in a BWR performance map. This map, Figure 16, presents the loci of maximum bundle temperatures as a function of time delay to flooding initiation. The map has been completed by inclusion of percentage embrittlement lines also obtained from the MOXY calculations.

Figure 16 shows that if the cladding temperature is 1600°F at the time of spray initiation and flooding occurs later, a maximum temperature of 2030°F and an embrittlement of about 5% would occur. Thus, this particular case has a 270°F temperature margin and approximately a 35% embrittlement margin.

A noticeable feature of the BWR performance map is that 40% embrittlement occurs at a temperature of 2470°F compared to 2470°F for a PWR. One reason for the 40% embrittlement occurring at a higher temperature is the greater cladding thickness used for the BWR study compared to 0.025 in. cladding thickness used for the PWR. Another feature is the canister quench phenomenon of a BWR allows the maximum temperature to shift from the rods near the top of the core to the central rods later in the transient (Figure 15); the PWR analysis considered only one rod through the core.

the thermal response for spray-on conditions. A representative plot of the heat transfer characteristics, steady state power levels, and time required in a typical MOXY calculation. The physical geometry of the system, the pertinent features are incorporated into the code.

assumptions of BWR-LOCA heat transfer coefficient response.

As represented by the temperature-time responses of three rods in the bundle, the temperature response has been used in these calculations because the temperature transient was assumed to be dominated by liquid accumulated in the lower plenum. Because of differences in plant liquid inventory remaining at the end of blowdown, the time of flooding varies from plant to plant. The curves in Figure 15 represent typical output of the MOXY code. The temperature response has been investigated separately on a parametric basis by Evans.¹⁴ The reaction rates, and other variables all are important in the calculation.

Figure 15, was generated by varying blowdown heat transfer rates in order to construct Figure 16, presents the loci of maximum bundle temperatures as a function of initial bundle temperature (at spray initiation) for given time delays to flooding initiation. The resulting performance embrittlement lines also obtained from the MOXY calculations.

The temperature is 1600°F at the time of spray initiation and flooding occurs 100 seconds and an embrittlement of about 5% would occur. Thus, this particular transient would result in a 35% embrittlement margin.

The performance map is that 40% embrittlement occurs at a temperature of about 2870°F and for the 0% embrittlement occurring at a higher temperature is the 0.0315 in. compared to 0.025 in. cladding thickness used for the PWR. Another reason is that the maximum temperature to shift from the rods near the canister early in the transient (Figure 15); the PWR analysis considered only one rod throughout the transient.

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Fig. 15 Calculated BWR transient temperature response for two case assumptions of LOCA heat transfer.

IV. RESEARCH AND DEVELOPMENT RELATED TO EMERGENCY CORE COOLING

Ultimately, safety assessment of large power reactors must be derived from:

- (1) The development and application of a system of analyses which can describe the complex transient, highly interacting array of events in the loss-of-coolant accident with sufficient precision to quantify the adequacy of emergency safety systems for terminating the fuel cladding temperature rise and to quantify the margins to cladding failure
- (2) The design and conduct of experiments from which data can be applied toward a meaningful assessment of the capability, completeness, and adequacy of the analytical techniques used to predict the performance of emergency safety systems in contemporary and future nuclear power plants.

The adequacy of the current analysis methods is uncertain in some important areas as a result of: (a) developing the analysis for certain heat transfer processes from first principles; (b) the simplifying assumptions and representations necessary to maintain the digital computer techniques manageable and economical; (c) the lack of empirical data for correlations which must substitute for basic analytical relationships; and (d) the lack of verification of the analysis through tests of representative systems which reproduce all important, controlling effects.

In order for the analysis to approach rapidly the desired level of precision and confidence, the development and the acquisition of experimental data must proceed together -- each supplying the other with direction and cor-

temperature response for two case assumptions of LOCA heat transfer.

DEVELOPMENT RELATED TO EMERGENCY CORE COOLING

For reactors must be derived from:

1. A system of analyses which can describe the complex transient, the loss-of-coolant accident with sufficient precision to quantify systems for terminating the fuel cladding temperature rise and to prevent core damage.

2. Elements from which data can be applied toward a meaningful assessment, and adequacy of the analytical techniques used to predict systems in contemporary and future nuclear power plants.

3. The methods is uncertain in some important areas as a result of: (a) the difficulty in processes from first principles; (b) the simplifying assumptions and lumped-system models; (c) computer techniques manageable and economical; (d) the lack of applicable substitute for basic analytical relationships; and (e) the lack of experimental representative systems which reproduce all important, controlling, and interacting

4. Only the desired level of precision and confidence, the development of the analysis need together -- each supplying the other with direction and confirmation.

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Fig. 16 BWR Performance Map.

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Fig. 1- BWR Performance Map.

ANC-A-863-H

A summary of the current analysis methods under development, refinement, and verification is included and is followed by a listing summarizing the various related experimental programs dealing with the loss-of-coolant

ANALYSIS DEVELOPMENTS

Current analyses use digital computer codes together with simplifying assumptions to solve the balance of conservation of mass, momentum, and energy of the system coolant. The reactor coolant circuits are modeled by fluid "control volumes" to which the conservation equations are applied. The computer models (codes) may be blowdown codes, core codes, and reflood codes. Reference 15 is a comprehensive treatise on many of the current

Blowdown codes are of two general types -- those with only a few fluid "control volumes" used for slow transients initiated by small breaks and those with up to 70 "control volumes" used for large breaks. A partial blowdown codes includes the Babcock & Wilcox Company CRAFT code,¹⁶ the Combustion Engineering, Inc., the Westinghouse Electric Corporation SATAN¹⁷ and BLOWDN codes,¹⁸ the German BRUCH code,¹⁹ the Italian RELAP code²¹ developed by Aerojet Nuclear Company for the USAEC.

Core codes primarily use thermal-hydraulic information generated by the blowdown and reflood codes to calculate the core flow history and resulting fuel cladding temperature. Single-flow-channel core codes include the Westinghouse Electric Corporation LOCTA-R2 code,²² the Danish DINO code,¹⁹ and the Aerojet Nuclear Company code.²³ Multichannel core codes have been written to allow interchannel flow caused in part by such things as ballooning of the fuel cladding. Currently used multichannel codes include the Westinghouse Electric Corporation CHARM codes, the Battelle Northwest Laboratories COBRA code,²⁴ and the English HAMBO code.²⁵ The latter two Company codes, MOXY,¹³ RADHT, and STAB, calculate in two-dimensions the rod-to-rod radiant energy in the bundle at one axial elevation. RADHT is a steady state version of MOXY, whereas STAB is the RADHT code with radiation interchange between the rods and steam in the channels.

The blowdown and core codes have the difficult task of predicting the all important magnitude of fuel cladding temperature at the time of reflooding initiation. As discussed in the previous section and developed in some detail in Appendix A, the temperature removed from the core during blowdown, particularly for the large inlet break in a PWR, depends heavily upon the initial conditions which influence the core flow and vessel fluid state throughout the entire blowdown process. The calculations become more complicated by the injection of emergency core coolant which is at a temperature and velocity quite different from that existing in the primary system at the ECC injection port. At present the conventional codes are often in error for some of the subtleties because they include simplifying assumptions such as:

- (1) Momentum flux terms in the momentum equation can be neglected
- (2) Core heat transfer has negligible effect on the core flow
- (3) No energy exchange between the fluid and the structural material
- (4) Water and steam velocities are identical and water and steam temperatures are identical
- (5) Flow divisions can be represented by one-dimensional equations
- (6) Performance of the primary system pumps can be predicted from pump data taken in liquid-only loops.

The current system codes are able to predict the trends of such data as that obtained in the semiscale blowdown and heated-core tests²⁶ but, generally, whereas one parameter such as pressure may follow close to the experimental parameter such as core inlet density may deviate considerably.

Reflood code development has lagged blowdown code development. In general, the current reflood codes do not adequately solve the conservation equations or they use simple approximations with weak coupling between the equations. The Aerojet Nuclear Company FLOODI, UKAEA CLADFLOOD,²⁷ and the Westinghouse Electric Corporation reflood code use flow approximations from the PWR-FLECHT⁵ experimental data. All three codes represent the primary loop thermal resistances without accounting for fluid inertia and system volumes. The Babcock & Wilcox Company REFLO, the Combustion Engineering, Inc., PERC code include the primary system volumes and fluid inertia as well as include the energy transfer processes in the core. The PERC code is potentially able to predict system flow oscillations during reflooding. Although oscillations have been shown to exist in the FLECHT-SET tests,²⁹ the oscillation frequencies predicted by the PERC code have not, as yet, been verified. Reflood code development is being pursued by researchers and manufacturers in many countries.

is under development, refinement, and verification is included for completeness in related experimental programs dealing with the loss-of-coolant accident.

codes together with simplifying assumptions to solve the basic equations for the system coolant. The reactor coolant circuits are modeled by interconnecting the equations. The computer models (codes) may be classified as follows. Reference 15 is a comprehensive treatise on many of the current codes.

1. -- those with only a few fluid "control volumes" used for slow decompression with up to 70 "control volumes" used for large breaks. A partial listing of the Aerojet Nuclear Company CRAFT code,¹⁶ the Combustion Engineering, Inc., CEFLASH code,¹⁷ and BLOWN codes,¹⁸ the German BRUCH code,¹⁹ the Italian TILT code,²⁰ and the Aerojet Nuclear Company for the USAEC.

2. -- those which use the information generated by the blowdown and reflood codes to provide a detailed calculation of the fuel cladding temperature. Single-flow-channel core codes include the code,²² the Danish DINO code,¹⁹ and the Aerojet Nuclear Company THETA code, written to allow interchannel flow caused in part by such nonuniformities as axial power distribution. Multichannel codes include the Westinghouse Electric Corporation THINC and the Aerojet Nuclear Company COBRA code,²⁴ and the English HAMBO code.²⁵ Three Aerojet Nuclear codes calculate in two-dimensions the rod-to-rod radiant energy interchange for a rod bundle version of MOXY, whereas STAB is the RADHT code modified to include axial conduction in the channels.

3. -- the difficult task of predicting the all important magnitude of fuel cladding temperature in the previous section and developed in some detail in Appendix B, the energy release rate for the large inlet break in a PWR, depends heavily upon subtle processes that take place throughout the entire blowdown process. The calculations of these processes require the use of an emergency core coolant which is at a temperature and velocity markedly different from the main coolant at the ECC injection port. At present the conventional codes are deficient in treating these processes. Simplifying assumptions such as:

1. The axial conduction equation can be neglected

2. The effect of the core flow

3. The effect of the structural material

4. The liquid and water and steam temperatures are identical

5. The use of one-dimensional equations

6. The flow regime can be predicted from pump data taken in liquid-only flow

7. The trends of such data as that obtained in the semiscale blowdown single-loop tests. If the parameter such as pressure may follow close to the experimental data, another parameter may deviate considerably.

8. Reflood code development. In general, the current reflood codes do not consider the simple approximations with weak coupling between the equations. The Aerojet Nuclear Company CRAFT code,²⁷ and the Westinghouse Electric Corporation reflood code obtain the steam flow from experimental data. All three codes represent the primary loops as a series of flow and system volumes. The Babcock & Wilcox Company REFLOOD code²⁸ and the Aerojet Nuclear Company THINC code represent the primary system volumes and fluid inertia as well as incorporate estimates of the effect of the primary system flow oscillations during core blowdown. The FLECHT-SET code is potentially able to predict system flow oscillations during core blowdown. To exist in the FLECHT-SET tests,²⁹ the oscillation frequency and amplitude must be known. Reflood code development is being pursued by nuclear organizations

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As indicated by the fault tree discussed previously, the important variables which the codes must predict with certainty are: (a) the coolant flow and enthalpy available to the core region and which control the energy removal, (b) the coolant efflux rate at the break which controls the duration of the blowdown, (c) the ECC intermediate depressurizing fluid, and (d) the cladding temperature resulting from the influence of the foregoing variables. Although the codes have not demonstrated the capability to predict adequately these variables, recent refinements and more comprehensive system experiments are expected to provide for critical comparisons in the near future.

EXPERIMENTAL PROGRAMS

The experimental investigations range from fundamental studies of heat transfer phenomena and bubble studies of nearly complete systems incorporating all important components and geometric features of a large power reactor. II. Ongoing and Planned Research and Development Related to LOCA-ECC in Water Cooled Reactors, provides a current and ongoing research and development related to emergency core cooling and is arranged in categories as follows:

Fundamental Studies in Blowdown and Heat Transfer

Emergency Core Coolant Delivery

Reflooding Heat Transfer

Scaled System Effects Experiments

Large-System Effects Experiments with Nuclear Heat Source.

The table shows the programs and their sponsors, the purpose of the investigations, experimental system descriptions and remarks relating to the emphasis of the experimental activities. An identifying code name is provided for each program so that Table II may be cross-referenced to a second table, Table III.

TABLE II

ONGOING AND PLANNED RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PROGRAM	CODE	PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.
<u>Fundamental Studies in Blowdown and Heat Transfer</u>					
1. PWR blowdown heat transfer (USAEC, ORNL)	ORNL/BDHT	Obtain data on blowdown heat transfer prior to ECC injection and on heat transfer related to operational upsets.	7 x 7 electrically heated rod array 12-ft-long, profiled power distribution, 1975 PWR pin diameter and pitch.	Continuing to July 1975	Emphasis on effects of blowdown on emergency core cooling array initialization and examination of the system
2. Investigation of Pressure Drops and Heat Transfer Coefficients (USAEC, University of Cincinnati)	U. Cin/AEC	Investigate pressure drops and heat transfer phenomena of importance during LOCA.	Two laboratory scale test systems: (1) small piping test section using freon to obtain two-phase pressure drops and flow patterns, and (2) a single tube with mercury on the inside to obtain basic transition boiling and reflooding heat transfer data.	Continuing	Fundamental to information on nuclear reactor

viously, the important variables which the codes must predict with reasonable accuracy are: (a) the amount of energy available to the core region and which control the energy removal from the core, (b) the amount of energy which controls the duration of the blowdown, (c) the ECC interaction with the core, and (d) the temperature resulting from the influence of the foregoing variables or processes. The codes must be able to predict adequately these variables, recent refinements in the codes are being made, and it is expected to provide for critical comparisons in the near future.

Transfer

the purpose of the investigations, experimental system descriptions, references, experimental activities. An identifying code name is provided for each program in the second table, Table III.

DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

TABLE 11 (Contd.)

ONGOING AND PLANNED RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PROGRAM	CODE	PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	
3. Deficient Cooling (US-AEC, General Electric Company, San Jose, Calif.)	GE/DC/AEC	Determine CHF and temperature regimes occurring due to power, flow, or pressure transient upset; evaluate temperature regimes in LOCA; evaluate consequences of fuel rod geometry changes.	Single-rod and 9(or 15) rod array transient CHF tests; single-rod simulated swelling effects on steady-state CHF; 9-rod array with geometry changes on spacer components and simulated bowing and swelling to investigate steady-state CHF.	Continuing	30	Transi includ flow, pressu study combin bundl power and s
4. Blowdown of Long Straight Pipes (Risley, UKAEA)	UKAEA	Acquire basic decompression data for model development.	Long, straight pipes up to 8 in. in diameter, 12 ft in length, heavily instrumented.	Continuing	31	Studi minin growt which trol
5. Research on Safety Assessment (ROSA) (Japan Atomic Energy Research Institute, Tokai Laboratory)	ROSA	Measure pressure and leak flow transients during blowdown, obtain scoping data on DNB during blowdown, evaluate pressure oscillations.	ROSA I: 56-cm-ID and 708-cm-long pressure vessel, rod array typical of BWR fuel assembly with up to 13 electrically heated rods, rupture disc unit.	Continuing	32	BWR v blowd trans part gram. facil fabri descr scale fects
6. Heat Transfer During Blowdown (AEG-Te Telefunken, German Federal Ministry)	AEG/GFM	Acquire data base for heat transfer coefficients and thermal-hydraulic behavior during blowdown.	Parametric tests with an internally cooled tube 3 m in length. Test section 4-rod bundle tests using BWR and PWR power profiles and 36-rod bundle tests with BWR and PWR power profiles.	Continuing	33	Paramet LOCA c develop transf
7. Depressurization Dynamics and Heating Transient, Fundamental Two-Phase Flow Studies (EURATOM, Ispra)	Ispra	Obtain data on heat transfer during system decompression.	Test loop consisted of an electrically-heated round-tube test section (15-mm OD, 13.8-mm ID, and 3-m length), an upper plenum, a lower plenum, a recirculation line, a pressurizer, and a quick opening valve.	---	34	One su mulation
8. Investigation of the Phenomena Involved in the Depressurization of Water-Cooled Reactors (Battelle-Institute in Frankfurt, Germany)	BMI/GFM	Determine loads on vessel internals during depressurization, investigate phenomena in the initial phase of depressurization (in particular the discharge rate).	Tests utilize a vessel of 0.8-m ID and 11.2-m length with simplified internals representation. Exit pipe diameter of 150 mm.	---	35	Both B initial dition vestig

TABLE 11 (Contd.)

RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	REMARKS
me CHF and ture re- occurring power, flow, sure tran- pset; eval- mperature in LOCA; e conse- of fuel metry	Single-rod and 9(or 15) rod array transient CHF tests; single-rod simu- lated swelling effects on steady-state CHF; 9- rod array with geometry changes on spacer com- ponents and simulated bowing and swelling to investigate steady- state CHF.	Continuing	30	Transient CHF tests include transient flow, power, and pressure testing and study of effects of combinations of intra- bundle clearance, power distribution, and spacer components.
basic de- sion data el develop-	Long straight pipes up to 8 in. in diameter, 12 ft in length, heavily instrumented.	Continuing	31	Studies for deter- mining basic bubble growth phenomena which cause and con- trol critical flow.
pressure k flow tran- during blow- btain scoping DNB during m, evaluate- e oscillations.	ROSA I: 56-cm-ID and 708- cm-long pressure vessel, rod array typical of BWR fuel assembly with up to 13 electrically heated rods, rupture disc unit.	Continuing	32	BWR vessel system blowdown and heat transfer data as part of this pro- gram. A ROSA II facility is being fabricated and is described under scaled system ef- fects experiments.
data base transfer ents and hydraulic during	Parametric tests with an internally cooled tube 3 m in length. Test section 4-rod bundle tests using BWR and PWR power profiles and 36-rod bundle tests with BWR and PWR power profiles.	Continuing	33	Parametric tests at LOCA conditions for development of heat transfer correlations.
data on heat during ecompression.	Test loop consisted of an electrically-heated round- tube test section (15-mm OD, 13.8-mm ID, and 3-m length), an upper plenum, a lower plenum, a recir- culation line, a pres- surizer, and a quick opening valve.	---	34	One subchannel si- mulation.
e loads on internals epressuri- investigate in the phase of fization (in r the dis- te).	Tests utilize a vessel of 0.8-m ID and 11.2-m length with simplified internals representation. Exit pipe diameter of 150 mm.	---	35	Both BWR and PWR initial fluid con- ditions to be in- vestigated.

TABLE II (Contd.)

ONGOING AND PLANNED RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PROGRAM	CODE	PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	R
9. Bottom Flood- ing Experi- ments (CISE, Milan, Italy)	ECC/CISE	Preliminary experi- mental investigation of coolant channel thermal transient following bottom ECC injection.	Test facility of directly heated tubular and annu- lar test elements 4 m in length.	---	36	Param were rate, entha power wall Data deter ence meter turna quenc
10. Dryout Ex- periments (CISE, Milan, Italy)	BDHT/CISE	Develop a model of local and instan- taneous values of flow and quality along fuel chan- nel, compare transient dryout conditions with steady-state con- ditions.	Externally heated annulus, 1.35-mm ID, 2.1-cm OD and 400-cm length, with initial pressures to 50 bars.	---	37	Exper ditic state stop sure power
11. Cooling of High- Temperature Surfaces (Central Electricity Generating Board, United Kingdom)	CEGB	Obtain data on the direct injection of coolant onto the heated surface for development of physical models.	Pin cluster tests with a concentric ring of heat- ers with a central sparge tube, single rod and tube experiments with water directed at top of heated length, and flat plates experiments with water directed at surface of plate.	Continuing	38	Studi distr hot c cool ness are
<u>EMERGENCY CORE COOLANT DELIVERY</u>						
1. Semiscale - Plexiglass Vessel (USAEC, Aerojet Nuclear Company, NRTS)	P-SSBD	Provide an under- standing of ECC delivery phenomena, investigate appli- cation of air-water countercurrent flow correlations to downcomer annulus geometries.	Plexiglass test vessel with geometry similar to the 1-1/2-loop semiscale vessel annulus and down- comer.	Continuing		Work othe test stan and fect leng plen stud
2. Semiscale Vessel - Steady- State Tests (USAEC, Aerojet Nuclear Com- pany, NRTS)	SS-SSBD	Same as for Semiscale-Plexiglass Vessel with steam- water countercurrent flow and in addition, effect of heat trans- fer from fluid bound- aries at elevated temperatures.	12-ft long, 8.5 in.-ID pressure vessel, 1/2-or 1-in. annulus width, outlet phase operator.	Last quar- ter of 1972		Stea water water curri to s of w ture ener gap lowe metr
3. 1-1/2-loop Semiscale - Isothermal Tests (USAEC, Aerojet Nuc- lear Company, NRTS)	I-SSBD	Investigate ECC de- livery with emphasis on the applicability of steady-state low pressure results to transient blowdown conditions.	Semiscale vessel; vessel downcomer; operating loop with pump, steam generator, and pressurizer; blowdown loop with simulated steam generator and pump; ac- cumulators.	Mid 1973		Test clud and ject in pres tion

TABLE II (Contd.)

RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	REMARKS
Primary experi- ment investigation of plant channel and transient injection bottom injection.	Test facility of directly heated tubular and annu- lar test elements 4 m in length.	---	36	Parameters varied were pressure, flow rate, coolant inlet enthalpy, heating power, and initial wall temperature. Data analyzed to determine influ- ence of test para- meters on temperature turnaround and quenching times.
Develop a model of and instan- taneous values of and quality of fuel chan- nel compare transient dryout conditions with steady-state con- ditions.	Externally heated annulus, 1.35-mm ID, 2.1-cm OD and 400-cm length, with initial pressures to 50 bars.	---	37	Experimental con- ditions: steady- state, inlet flow stoppage with pres- sure transient or power surge.
Obtain data on the injection of plant onto the surface for development of analytical models.	Pin cluster tests with a concentric ring of heat- ers with a central sparge tube, single rod and tube experiments with water directed at top of heated length, and flat plates experiments with water directed at surface of plate.	Continuing	38	Studies on water distribution in a hot cluster and cooling effective- ness of aerosols are in progress.
Develop an under- standing of ECC delivery phenomena, investigate appli- cation of air-water countercurrent flow analogs to downer annulus experiments.	Plexiglass test vessel with geometry similar to the 1-1/2-loop semiscale vessel annulus and down- comer.	Continuing		Work related to other semiscale testing to under- stand ECC delivery and bypass. Ef- fects of downcomer length, and lower plenum geometry studied.
Develop as for semiscale-Plexiglass test with steam- water countercurrent flow and in addition, study of heat trans- fer from fluid bound- ary at elevated temperatures.	12-ft long, 8.5 in.-ID pressure vessel, 1/2-or 1-in. annulus width, outlet phase operator.	Last quar- ter of 1972		Steady-state air- water and steam- water counter- current flow tests to study effects of water tempera- ture, wall stored energy, downcomer gap width, and lower plenum geo- metry.
Investigate ECC de- livery with emphasis on applicability of steady-state low pressure results to transient blowdown conditions.	Semiscale vessel; vessel downcomer; operating loop with pump, steam generator, and pressurizer; blowdown loop with simulated steam generator and pump; ac- cumulators.	Mid 1973		Test conditions in- clude blowdown with and without ECC in- jection; variation in ECC injection pressure and loca- tion, lower plenum geometry, pump shut- down times, and steam

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PROJECT	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	REMARKS
Primary experimental investigation of channel transient bottom section.	Test facility of directly heated tubular and annular test elements 4 m in length.	---	36	Parameters varied were pressure, flow rate, coolant inlet enthalpy, heating power, and initial wall temperature. Data analyzed to determine influence of test parameters on temperature turnaround and quenching times.
A model of instantaneous values of quality of channel compare dryout tests with state conditions.	Externally heated annulus, 1.35-mm ID, 2.1-cm OD and 400-cm length, with initial pressures to 50 bars.	---	37	Experimental conditions: steady-state, inlet flow stoppage with pressure transient or power surge.
Data on the injection of water onto the surface for test of models.	Pin cluster tests with a concentric ring of heaters with a central sparge tube, single rod and tube experiments with water directed at top of heated length, and flat plates experiments with water directed at surface of plate.	Continuing	38	Studies on water distribution in a hot cluster and cooling effectiveness of aerosols are in progress.
Work under way on ECC phenomena, late application of air-water current flow tests to annulus systems.	Plexiglass test vessel with geometry similar to the 1-1/2-loop semiscale vessel annulus and downcomer.	Continuing		Work related to other semiscale testing to understand ECC delivery and bypass. Effects of downcomer length, and lower plenum geometry studied.
For 1-1/2-loop semiscale vessel with steam-intercurrent in addition, heat transfer fluid boundary conditions.	12-ft long, 8.5 in.-ID pressure vessel, 1/2-or 1-in. annulus width, outlet phase operator.	Last quarter of 1972		Steady-state air-water and steam-water counter-current flow tests to study effects of water temperature, wall stored energy, downcomer gap width, and lower plenum geometry.
Test ECC design with emphasis on applicability of state low pressure results to blowdown tests.	Semiscale vessel; vessel downcomer; operating loop with pump, steam generator, and pressurizer; blowdown loop with simulated steam generator and pump; accumulators.	Mid 1973		Test conditions include blowdown with and without ECC injection; variation in ECC injection pressure and location, lower plenum geometry, pump shutdown times, and steam generator secondary conditions.

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TABLE II (Contd.)

ONGOING AND PLANNED RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PROGRAM	CODE	PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	RE
4. Steam-Water Mixing Test Program (USAEC and Combustion Engineering, Inc. Windsor, Connecticut)	CE/USAEC	Investigate interaction of the ECC fluid with steam in the primary system.	Steam generator; cold leg with loop seal, simulated pump, and ECC injection nozzle; and reactor vessel. Test section is a geometrically scaled model of the piping from the steam generator to the reactor vessel inlet.	Continuing	39	Investig hydraul injectio clination section librium various to obtai water re influenc vessel water re
<u>Reflooding Heat Transfer</u>						
1. Low Rate ECC Reflooding Heat Transfer (FIAT Nuclear Energy Section, Italy)	FIAT	Obtain rod bundle heat transfer data during reflooding.	Twenty-one heater rod assembly with upper and lower plenum. Heater OD is 9.8 mm, length is 1184 mm, and pitch is 12.9 mm. Uniform power distribution.		40	Flooding the rang to 0.6 power va 0.45 to kW/ft. ship bet flooding mum acc power wa
2. 1-1/2-Loop Mod-1 (USAEC, Aerojet Nuclear Company, NRTS)	Mod-1	Includes investigation of blowdown and ECC injection variables and their influence on cladding temperature.	A simulated PWR system with generic component elevation, electrically heated core, operating loop, blowdown loop, accumulator, HPIS and LPIS.[a] Capability for 5.5- and 12-foot heated rod lengths.			Emphasiz and loop effects livery to region d down.
3. ECC Reflooding Experiments (Siemens, Germany)	Siemens/GFM	Investigate the performance of low pressure ECCS and obtain fluid flow and heat transfer data during reflooding.	Preliminary reflooding tests with a single internally cooled tube. Also an electrically heated 340-rod array, 10.75-mm rod diameter, 3 m in length with chopped cosine power distribution. Bundle has three regions which can be independently heated with maximum power input of 1.4 MW. A 16-bar pressure vessel contains the bundle. A bypass simulates the annular downcomer.	Continuing	41	Flooding bottom c simultan and bott
4. Loss-of-Fluid Test (USAEC, Aerojet Nuclear Company, NRTS)	LOFT	Obtain data on thermal-hydraulic behavior associated with core and on core thermal response during ECC injection.	55 MW(t) PWR with an operating loop, a blowdown loop, accumulators, LPIS, HPIS[a], supporting systems, and initially a 5.5-ft-long core of typical PWR fuel dimensions with about 1300 fuel pins. Capability for a 12-ft-long core provided.	Continuing		Refloodi transfer nuclear Wide ran injectio and injec tations selected

TABLE II (Contd.)

AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

USE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	REMARKS
the inter- the ECC steam primary	Steam generator; cold leg with loop seal, simulated pump, and ECC injection nozzle; and reactor vessel. Test section is a geometrically scaled model of the piping from the steam generator to the reactor vessel inlet.	Continuing	39	Investigate piping hydraulic resistance, injection nozzle inclination, test section size. Equilibrium tests at various pressures to obtain data on water remaining, and influence of reactor vessel geometry on water remaining.
bundle transfer data flooding.	Twenty-one heater rod assembly with upper and lower plenum. Heater OD is 9.8 mm, length is 1184 mm, and pitch is 12.9 mm. Uniform power distribution.		40	Flooding rates in the range of 0.2 to 0.6 in./sec, power varied from 0.45 to 0.65 kW/ft. A relationship between bottom flooding and maximum acceptable power was obtained.
investi- blowdown injection and their on clad- temperature.	A simulated PWR system with generic component elevation, electrically heated core, operating loop, blowdown loop, accumulator, HPIS and LPIS.[a] Capability for 5.5- and 12-foot heated rod lengths.			Emphasizes system and loop component effects on ECC delivery to core region during blowdown.
the the ce of ure ECCS n fluid heat data dur- oding.	Preliminary reflooding tests with a single internally cooled tube. Also an electrically heated 340-rod array, 10.75-mm rod diameter, 3 m in length with chopped cosine power distribution. Bundle has three regions which can be independently heated with maximum power input of 1.4 MW. A 16-bar pressure vessel contains the bundle. A bypass simulates the annular downcomer.	Continuing	41	Flooding from the bottom only and simultaneous top and bottom flooding.
ata on hydraulic associated and on mal re- ring ECC	55 MW(t) PWR with an operating loop, a blowdown loop, accumulators, LPIS, HPIS[a], supporting systems, and initially a 5.5-ft-long core of typical PWR fuel dimensions with about 1300 fuel pins. Capability for a 12-ft-long core provided.	Continuing		Reflooding heat transfer data on nuclear fueled core. Wide range of ECC injection parameters and injection locations can be selected.

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TABLE II (Contd.)

ONGOING AND PLANNED RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

PROGRAM	CODE	PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.
5. Power Burst Facility (USAEC, Aerojet Nuclear Company, NRTS)	PBF	Determine maximum fuel cladding temperature permitted at the time of coolant delivery without loss of cladding integrity, the effect of ballooning on maximum permissible temperature, and the effect of degraded coolant performance on the maximum permissible cladding temperature.	Open tank reactor vessel housing 3-ft-long driver core, a central flux trap region containing an in-pile tube in which the test fuel is located and a loop coolant system for providing required system conditions in the test section.	Continuing	
<u>Scaled System Effects Experiments</u>					
1. 1-1/2-Loop Mod-1 (USAEC, Aerojet Nuclear Company, NRTS)	Mod-1	Provide LOCA-ECC data; for analytical model evaluation of total system codes, on effects of physical scale in relation to LOFT scale, and for guidance of LOFT test program.	A simulated PWR system with an electrically heated core, an operating loop, a blowdown loop, accumulator, HPIS, and LPIS[a]. Capability for 5.5- and 12-ft-long cores and other system geometry changes.	---	
2. BWR Blowdown Heat Transfer (USAEC and General Electric Company, San Jose, Calif.)	GE/BDHT/AEC	Provide information on the transient heat transfer following a rupture of a steam line or recirculation line in a BWR.	Scaled BWR system consisting of a pressure vessel with two external drive pump recirculation loops. Vessel contains a full-size 49-rod electrically heated bundle, two jet pumps, and a steam separator.	Continuing	42
3. Research of Safety Assessment (ROSA), (Japan Atomic Energy Research Institute Tokai Laboratory)	ROSA II	Obtain information on the effects of system parameters such as piping resistance, power, pressure, break location and size, and ECCS variations, for both BWR and PWR conditions.	ROSA II facility consists of a 265-mm-ID and 5900-mm-long pressure vessel containing 109 electrically heated rods 1500 mm in length, two loops with pump and steam generators, one of these loops has a rupture unit, pressurizer, and ECCS.	Continuing	32
4. Full Scale Safety Experiments of FUGEN	FUGEN	Obtain cladding temperature information during blowdown and ECC injection, discharge reaction forces, and transient characteristics of the primary system.	The facility consists of a steam drum, circulating pump, downcomer, lower header, 25 pressure tube assemblies 3 of which contain electrically heated full-scale fuel assemblies, and ECCS. Fuel assemblies contain 28 heaters 3.7 m in length.	Continuing	43

PROPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.	REMARKS
<p>the maxi- clad- perature d at the</p> <p>coolant de- without loss ling inte- the effect of ing on maxi- missible tem- , and the of degraded performance maximum per- cladding ture.</p>	<p>Open tank reactor vessel housing 3- ft-long driver core, a central flux trap</p> <p>region containing an in- pile tube in which the test fuel is located and a loop coolant system for providing required sys- tem conditions in the test section.</p>	Continuing		<p>Tests with both PWR and BWR fuel cluster; up to 24-pin array. Top spray or bottom flooding capability. Tests include use of irradiated fuel.</p>
<p>LOCA-ECC for analy- model eval- of total codes, on of phy- scale in n to LOFT and for e of LOFT ogram.</p>	<p>A simulated PWR system with an electrically heated core, an operat- ing loop, a blowdown loop, accumulator, HPIS, and LPIS[a]. Capability for 5.5- and 12-ft-long cores and other system geometry changes.</p>	---		<p>Extensive systems effects and para- meter variations on system component in- fluence on blowdown, blowdown heat trans- fer, ECC delivery, and core reflooding cooling.</p>
<p>informa- the tran- heat transfer ng a rupture eam line or lation line R.</p>	<p>Scaled BWR system con- sisting of a pressure vessel with two ex- ternal drive pump re- circulation loops. Vessel contains a full-size 49-rod elec- trically heated bundle, two jet pumps, and a steam separator.</p>	Continuing	42	<p>Testing conditions to include: variation of initial system con- ditions, break size and locations, and power decay transient.</p>
<p>information effects of parameters piping re- se, power, re, break on and size, CS variations, th BWR and ditions.</p>	<p>ROSA II facility con- sists of a 265-mm-ID and 5900-mm-long pres- sure vessel containing 109 electrically heated rods 1500 mm in length, two loops with pump and steam generators, one of these loops has a rup- ture unit, pressurizer, and ECCS.</p>	Continuing	32	<p>ROSA II facility is being fabricated. Testing is scheduled to begin in October 1973.</p>
<p>cladding tem- re information blowdown and ection, dis- reaction and transi- racteristics primary sys-</p>	<p>The facility consists of a steam drum, circulating pump, downcomer, lower header, 25 pressure tube assemblies 3 of which contain electrically heated full-scale fuel assemblies, and ECCS. Fuel assemblies contain 28 heaters 3.7 m in length.</p>	Continuing	43	<p>Experimental facility is a mockup of the prototype, heavy- water-moderated, boiling-light-water- cooled reactor, FUGEN.</p>

TABLE II (Contd.)

ONGOING AND PLANNED RESEARCH AND DEVELOPMENT RELATED TO LOCA-ECC IN WATER COOLED REACTOR

PROGRAM	CODE	PURPOSE	EXPERIMENTAL SYSTEM	SCHEDULE	REF.
5. Full Length Emergency Cooling Heat Transfer-System Effects Tests (USAEC and Westinghouse Electric Corporation, Pittsburgh, Pa.)	FLECHT-SET	Provide experimental data on the influence of system effects on ECC behavior during reflooding.	FLECHT flow housing and a power profiled 10 x 10 heater rod test assembly, coolant supply system, downcomer, pressure control system, upper and lower test section plenums, and single loop consisting of piping, valves, and orifices. In later tests the single loop will be replaced by two loops each having a full length steam generator, pump simulator, and piping arranged to represent elevations and flow characteristics in the loops.	End 1973	29 Test influ chara flood relea ding carry sure syste testi consi
<u>Large-System Effects Experiments with Nuclear Heat Source</u>					
1. Loss-of-Fluid Test (USAEC, Aerojet Nuclear Company, NRTS)	LOFT	Provide PWR integral test data on all principal aspects of an LOCA with ECC injection including transient thermal, mechanical, and nuclear response of the system, capability of ECCS, margins of safety, containment system effects, and fission product behavior.	55 MW(t) PWR with an operating loop, a blow-down loop, accumulators, LPIS, HPIS[a], supporting systems, and initially a 5.5-ft-long core otherwise of typical PWR fuel dimensions with about 1300 fuel pins. Capability for a 12-ft-long core provided.	Continuing	Test vide menta at la more sion react gener of nu inter cladd chang ditio coola metry temper condi minin emerg ing p

[a] HPIS -- high pressure injection system; LPIS -- low pressure injection system.

Table III is intended to show the relation of the various research programs throughout the world to those previously presented fault tree which are enclosed by diamond symbols. The table is arranged starting with Branch tree in which, for example, information needs in the blowdown heat transfer area are exhibited. For each event diamond, a list of "subevent effects" is developed which indicates specific technical problems which must be resolved. These problems are in turn related to a code symbol of the various research programs identified in Table I. This table is intended more to be illustrative of the manner in which the total research and development is applied to the problem rather than to be a comprehensive or exhaustive treatment identifying deficiencies of areas not addressed by current development efforts.

II (Contd.)

EVENT RELATED TO LOCA-ECC IN WATER COOLED REACTORS

EXPERIMENTAL SYSTEM	SCHEDULE	REF.	REMARKS
<p>CHT flow housing and a er profiled 10 x 10 ater rod test assembly, plant supply system, ncomer, pressure con- l system, upper and er test section ple- s, and single loop sisting of piping, rves, and orifices. later tests the single p will be replaced two loops each having ull length steam gen- ator, pump simulator, piping arranged to resent elevations and w characteristics in loops.</p>	End 1973	29	<p>Test emphasize the influence of system characteristics on flooding rate, heat release from clad- ding and flow housing, carryover, and pres- sure drop. Additional system features and testing are being considered.</p>
<p>MW(t) PWR with an erating loop, a blow- n loop, accumulators, S, HPIS[a], supporting tems, and initially .5-ft-long core other- e of typical PWR fuel ensions with about 0 fuel pins. Capability a 12-ft-long core vided.</p>	Continuing		<p>Test data will pro- vide unique experi- mental information; at large scale, in more than one dimen- sion within core and reactor vessel, with generic time constants of nuclear fuel, with interacting effects of cladding ballooning, changing fuel gap con- ditions and changing coolant channel geo- metry, and at high temperature fuel conditions for deter- mining margins of emergency core cool- ing performance.</p>
<p>w pressure injection system.</p>			

ous research programs throughout the world to those events in the
 d symbols. The table is arranged starting with Branch 1 of the fault
 own heat transfer area are exhibited. For each event identified as a
 ndicates specific technical problems which must be considered or
 of the various research programs identified in Table II. This set of
 n the total research and development is applied to the problem areas
 identifying deficiencies of areas not addressed by research and

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TABLE III

RESEARCH AND DEVELOPMENT RELATED TO FAULT TREE EVENTS

BRANCH 1 -- Analysis Overpredicts Energy Removal Rates Or Underpredicts Stored Energy

BRANCH EVENTS	SUBEVENT EFFECTS TO BE CONSIDERED	POTENTIALLY R AND D PR
<ul style="list-style-type: none"> - Gap conductance during blowdown with nuclear fuel - Conditions to initiate CHF - Conditions to rewet and restore pre-CHF cooling - Conditions for transition boiling heat transfer - Conditions for film boiling heat transfer - Radial coolant flow in core 	<p>PBF, LOFT</p> <p>In one dimension:</p> <p>UCIN/AEC ORN/BDHT ROSA GE/BDHT/AE Mod-1 GE/DC/AEC AEG/GFM ISPRA BDHT/CISE CEGB</p> <p>In more than one dimension:</p> <p>FUGEN LOFT</p>
<ul style="list-style-type: none"> - Operating gap conductance - Axial location of peak power - Magnitude of peak power - Power decline rate - Decay heat rate - Metal-water reaction energy - Fuel densification 	<p>PBF LOFT</p>

BRANCH 2 -- Analysis Underpredicts Time To Initiate Reflooding.

<ul style="list-style-type: none"> - Broken pipes - Stuck check valves - Line restrictions - Low gas pressure - Low water inventory 	<p>Design - HHS Quality Assurance Technical Specification, and Control</p>
<ul style="list-style-type: none"> - ECC entrainment to break - Steam condensation and pressure drop in cold leg piping 	<p>CE/AEC, SSBD Mod-1, LOFT</p>
	<ul style="list-style-type: none"> - Downcomer penetration of ECC with counter-current blowdown flow 	<p>SSBD, ROSA LOFT</p>

TABLE III

DEVELOPMENT RELATED TO FAULT TREE EVENTS

Removal Rates Or Underpredicts Stored Energy

SUBEVENT EFFECTS TO BE CONSIDERED	POTENTIALLY APPLICABLE R AND D PROGRAMS
Gap conductance during blowdown with nuclear fuel	PBF, LOFT
Conditions to initiate CHF	In one dimension:
Conditions to rewet and restore pre-CHF cooling	
Conditions for transition boiling heat transfer	
Conditions for film boiling heat transfer	
Radial coolant flow in core	
	UCIN/AEC ORNL/BDHT ROSA GE/BDHT/AEC Mod-1 GE/DC/AEC AEG/GFM ISPRA BDHT/CISE CEGB
	In more than one dimension:
	FUGEN LOFT
Operating gap conductance	PBF LOFT
Axial location of peak power	
Magnitude of peak power	
Power decline rate	
Decay heat rate	
Metal-water reaction energy	
Fuel densification	
Time To Initiate Reflooding.	
Broken pipes	Design - HHST ^[a]
Stuck check valves	
Line restrictions	
Low gas pressure	
Low water inventory	
	Quality Assurance, Technical Specification, and Compliance ^[b]
ECC entrainment to break	CE/AEC, SSBD, ROSA II, Mod-1, LOFT
Steam condensation and pressure drop in cold leg piping	
Downcomer penetration of ECC with counter-current blowdown flow	SSBD, ROSA II, Mod-1, LOFT

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TABLE III (Contd.)

RESEARCH AND DEVELOPMENT RELATED TO FAULT TREE EVENTS

BRANCH EVENTS

SUBEVENT EFFECTS TO BE CONSIDERED

POTENTIALLY
R AND D PR

Analysis Underpredicts
Water Loss from
Lower Plenum

- Blowdown backflow from core
- Geometry sensitivity for lower plenum water entrainment

SSBD, ROSA,
LOFT

Analysis Underpredicts
Blowdown Time

- Leak flow rate model for double ended break or for split break
- Influence of pipe size on leak flow rate
- Influence of stored heat from piping and system internals
- Influence of secondary system energy and leaking steam
- Influence of ECC injection

SSBD, Mod-1

SSBD, UKAEA,
BMI/GFM, Mod

UKAEA, FUGEN

SSBD, Mod-1

SSBD, Mod-1

SSBD, ROSA,

BRANCH 3 -- Analysis Underpredicts Core Temperature At Time Of Reflooding Initiation.

Analysis Overpredicts Energy
Removal Rates During
Blowdown or Under-
predicts Stored Energy

This event is in essence low blowdown heat transfer which is developed in Branch 1.

Analysis Underpredicts
Time to Initiate Reflooding

This event and its subevents are identical to Branch 2.

BRANCH 4 -- Analysis Underpredicts Power Generation During Core Reflooding.

Analysis Underpredicts
Metal-Water or Decay
Heat Power Generation

This event is caused by subevents identical to those of the second event developed for Branch 1.

BRANCH 5 -- Analysis Overpredicts Energy Removal Rates During Core Reflooding

PWR-FLECHT
Results not Typical
or Misapplied

- One dimensionality
- Axial power profile noncosine
- Zircaloy versus stainless steel
- Experimental errors
- Limited system representation

FLECHT-SET
ROSA II
FIAT
Mod-1
Siemens/GFM
LOFT

D DEVELOPMENT RELATED TO FAULT TREE EVENTS

SUBEVENT EFFECTS TO BE CONSIDERED	POTENTIALLY APPLICABLE R AND D PROGRAMS
- Blowdown backflow from core	} SSBD, ROSA, Mod-1, LOFT
- Geometry sensitivity for lower plenum water entrainment	
- Leak flow rate model for double ended break or for split break	} SSBD, Mod-1
- Influence of pipe size on leak flow rate	
- Influence of stored heat from piping and system internals	} SSBD, UKAEA, ROSA, BMI/GFM, Mod-1, LOFT
- Influence of secondary system energy and leaking steam	
- Influence of ECC injection	} UKAEA, FUGEN, LOFT
	} SSBD, Mod-1, LOFT
	} SSBD, ROSA, Mod-1, LOFT

Core Temperature At Time Of Reflooding Initiation.

This event is in essence low blowdown heat transfer which is developed in Branch 1.

This event and its subevents are identical to Branch 2.

Power Generation During Core Reflooding.

This event is caused by subevents identical to those of the second event developed for Branch 1.

Energy Removal Rates During Core Reflooding

- One dimensionality	} FLECHT-SET ROSA II FIAT Mod-1 Siemens/GFB LOFT
- Axial power profile noncosine	
- Zircaloy versus stainless steel	
- Experimental errors	
- Limited system representation	

TABLE III (Contd.)

RESEARCH AND DEVELOPMENT RELATED TO FAULT TREE EVENTS

BRANCH EVENTS	SUBEVENT EFFECTS TO BE CONSIDERED	POTENTIAL R AND D
Analysis Underpredicts Effect of Plugging by ECC Water in Pipes	<ul style="list-style-type: none"> - Momentum exchange between system steam and injected ECC. - Geometry of injection configuration - Flow regime downstream of injection location 	CE/USAEC Mod-1
Analysis Underpredicts Energy or Mass Transfer from Steam Generator Secondary System to Primary Loop	<ul style="list-style-type: none"> - Secondary system fluid conditions - Heat transfer characterization across steam generator tubes - Acceleration-pressure-drop due to compressible flow - Steam leakage from secondary system 	Mod-1 ROSA II LOFT
Analysis Underpredicts Pump Resistance	<ul style="list-style-type: none"> - Two-phase pressure drop for driven, coasting and stalled large scale pumps - Compressibility effects - Density of steam leaving the steam generator 	Small Scale Mod-1 ROSA II Large Scale LOFT
Analysis Underpredicts Mass Flow Rate from Vessel	<ul style="list-style-type: none"> - Liquid mass fraction carried from upper plenum into steam generator - Division of coolant mass leaving core and entering unbroken loops and broken loops - Steam flow distribution and entrainment of ECC in inlet pipes and in upper core barrel annulus 	SSBD Mod-1 ROSA II CE/USAEC FLECHT-LOFT

[a] Heavy Section Steel Technology Program, ORNL/USAEC.
[b] Not R and D -- added for completeness.

The description of the experiments referenced in the two tables, when viewed in light of the current state of the art techniques, reveals the following:

- (1) A great deal of research and development has been and is continuing to be addressed to bench and laboratory scale experiments which are improving the statistical data base for analytical model development in areas of leak flow rate, rod bundle heat transfer, difference between steady state and transient heat transfer, and two-phase pressure drops. The various experiments are being conducted in several countries.
- (2) A considerable number of system effects experiments are to become operational within the next year. The difficulty in applying data from any system effects test at less than full size is the understanding of the scaling methods used. Although significant differences exist in the scaling of similar experiments,

H AND DEVELOPMENT RELATED TO FAULT TREE EVENTS

POTENTIALLY APPLICABLE
R AND D PROGRAMS

SUBEVENT EFFECTS TO BE CONSIDERED

- Momentum exchange between system steam and injected ECC
- Geometry of injection configuration
- Flow regime downstream of injection location

CE/USAEC
Mod-1

- Secondary system fluid conditions
- Heat transfer characterization across steam generator tubes
- Acceleration-pressure-drop due to compressible flow
- Steam leakage from secondary system

Mod-1
ROSA II
LOFT

- Two-phase pressure drop for driven, coasting and stalled large scale pumps
- Compressibility effects
- Density of steam leaving the steam generator

Small Scale:
Mod-1
ROSA II
Large Scale:
LOFT

- Liquid mass fraction carried from upper plenum into steam generator
- Division of coolant mass leaving core and entering unbroken loops and broken loops
- Steam flow distribution and entrainment of ECC in inlet pipes and in upper core barrel annulus

SSBD
Mod-1
ROSA II
CE/USAEC
FLECHT-SET
LOFT

ology Program, ORNL/USAEC.
completeness.

nts referenced in the two tables, when viewed in light of the current state of the analysis

nd development has been and is continuing to be addressed to bench and
nts which are improving the statistical data base for analytical model
k flow rate, rod bundle heat transfer, difference between steady state and
Two-phase pressure drops. The various experiments are being conducted in

system effects experiments are to become operational within the next year.
ata from any system effects test at less than full size is the understanding of
Although significant differences exist in the scaling of similar experiments,

through adequate measurements, the data from systems effects experiments could still provide significant increase in the data base for testing the capability of system codes. Unfortunately, with exception of the LOFT tests, and possibly the FUGEN tests⁴³ in Japan, the various system effects configurations are small in scale compared to large power reactors and dominantly one-dimensional the reactor core and vessel representation.

- (3) The various research and development programs promise to provide a wealth of information on blowdown fluid behavior, blowdown heat transfer, and reflooding heat transfer. Little attention outside the U.S. and Japan, however, is being directed to the question of ECC delivery during the blowdown process. Scoping information is needed in this area in order to guide the analysis and development and determine whether the effects of interaction between the ECC delivery and blowdown processes can contribute to both loss of the ECC inventory as it is delivered and extension of the blowdown time.

V. CONCLUSIONS

The "fault tree" method has been used to identify and evaluate the sensitivity of potential analytical faults of performance requirements for a PWR emergency core cooling system. From the example case studied the following: the predicted energy removal rates prior to ECC action are significantly lower than actual values, the majority of faults identified on the tree will be unimportant. However, if predicted energy removal rates are higher than actual, cladding temperature limits are approached before the lower plenum is refilled, then the subtle errors in the analysis, duration, "steam binding", friction pressure drop, application of flooding heat transfer data, and decay heat rates could have significant influence on ECC performance requirements.

The performance maps and discussion presented have shown for the PWR case that the cladding temperature initiation of flooding is primarily influenced by the variables of flooding rate, initial decay heat power level, initial metal-water reaction, rate, and initial zirconium oxide thickness. For the BWR case, the influence of the thickness change in location of the hot spot from a corner position to a central position in the fuel assembly during the accident permits cladding temperatures considerably in excess of 2300° before a 40% embrittlement is reached.

The research and development programs throughout the world are addressing many aspects of the all important blowdown heat transfer, and ECC heat transfer question. System effects tests addressing the ECC delivery and lower plenum refill are limited to a few programs in the U.S. and Japan.

Improved precision in analysis and verification of the calculational techniques used for predicting blowdown duration are expected to reveal conservatism in quantifying the margins of ECC performance. Care is required in the analysis and verification of emergency core cooling delivery and lower plenum refill in order to determine potential margins of ECC system design.

data from systems effects experiments could still provide a basis for testing the capability of system codes. Unfortunately, with the possibly the FUGEN tests⁴³ in Japan, the various system effects compared to large power reactors and dominantly one-dimensional in nature.

Current programs promise to provide a wealth of information on natural convection heat transfer, and reflooding heat transfer. Little attention is being directed to the question of ECC delivery during the blowdown. More information is needed in this area in order to guide the analysis of the effects of interaction between the ECC delivery and the blowdown, to both loss of the ECC inventory as it is delivered and extension of the blowdown.

V. CONCLUSIONS

To identify and evaluate the sensitivity of potential analytical faults in the selection of the core cooling system. From the example case studied the following are cited. If the input data are significantly lower than actual values, the majority of the fault states are conservative. However, if predicted energy removal rates are higher than actual values so that the lower plenum is refilled, then the subtle errors in the analysis of blowdown, application of flooding heat transfer data, and decay heat power generation performance requirements.

The results presented have shown for the PWR case that the cladding temperature rise after blowdown is sensitive to the variables of flooding rate, initial decay heat power level, initial temperature, and cladding oxide thickness. For the BWR case, the influence of the thicker cladding and the fuel assembly position to a central position in the fuel assembly during the course of the blowdown is probably in excess of 2300° before a 40% embrittlement is reached.

Throughout the world are addressing many aspects of the all important blowdown, a major question. System effects tests addressing the ECC delivery and potential delay in ECC delivery are in the U.S. and Japan.

Further development of the calculational techniques used for predicting blowdown heat transfer and the conservatism in quantifying the margins of ECC performance. Completeness is yet to be achieved in emergency core cooling delivery and lower plenum refill in order to quantify the margins of ECC performance.

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APPENDIX A

EMERGENCY CORE COOLING SYSTEM DESIGN CONCEPTS FOR CONTEMPORARY PWR's AND B

The treatment of the design of emergency coolant supply systems is described for those plants of current designs. The description for PWR designs characterizes the generic features which typify the plants of several manu

PWR ECC DESIGNS

A schematic representation of two loops of a multiple loop U.S. PWR is shown in rough scale in Figure A-1. This configuration has all primary system components and all inlet and outlet pipes to the reactor vessel above the top of arrangement is intended to assure an enclosure around the reactor core that has no penetrations and thus has a high auxiliary refill coolant to cover the core for all possible pipe breaks in the primary coolant circuit. The PWR ECC with injection into the inlet pipes or the upper regions of the inlet annulus, depending on the manufacturer, and flooding the reactor vessel from the core bottom.

Figure A-1. Typical U.S. Multiple Loop PWR System.

SYSTEM DESIGN CONCEPTS FOR CONTEMPORARY PWR's AND BWR's

Emergency coolant supply systems is described for those plants of current and near future. Characterizes the generic features which typify the plants of several manufacturers.

loops of a multiple loop U.S. PWR is shown in rough scale in Figure A-1. As shown, this components and all inlet and outlet pipes to the reactor vessel above the top of the core. This closure around the reactor core that has no penetrations and thus has a high potential for for all possible pipe breaks in the primary coolant circuit. The PWR ECC supply systems the upper regions of the inlet annulus, depending on the manufacturer, are intended for bottom.

n Flow

2

-1. Typical U.S. Multiple Loop PWR System.

Passively activated (gas dome) accumulators (storage tanks) and low capacity pumping systems de reactor primary system. Activation of the various ECC subsystems depends on the size of the primary pressure injection system (HPIS) pumps, although utilized for all break sizes, are intended primarily for small injection system (LPIS) pumps are used when breaks are sufficiently large that the primary system approximately 170 psia and are intended for long term core reflooding. The accumulator systems, although are intended primarily for large break sizes for which the primary system depressurization falls below the pre in the accumulators. The accumulator pressure varies among reactor manufacturers and ranges from 750 psia psia.

Both reactor systems and ECC plumbing configurations vary considerably among manufacturers with design features pertinent to the efficacy of emergency core cooling system performance. Particularly, the core barrel annulus, the geometry of the lower plenum of the reactor vessel, the method of ECC injection into and the quantity of available accumulator water are important features that vary. The varying configuration Figure A-2. The ECC systems in plants for two manufacturers inject emergency coolant into each inlet pipe however, varies between 60 and 90 degrees for the two different plant designs. The ECC system of a third injects the emergency coolant directly into the upper region of the inlet annulus.

Figure A-3 is a typical bar chart illustrating, for a range of break sizes, the various system component reflooding of the core for all primary system pipe break sizes. The typical unimpeded delivery capability for illustrated in Table A-I.

PWR configurations of other than U.S. designs are very similar in the design of the primary coolant circuit. The German version (Siemens) injects accumulator coolant into both the outlet and inlet pipes.

TABLE A-I

TYPICAL DELIVERY CAPABILITY FOR THE VARIOUS PWR-ECC SYSTEMS

<u>ECC System</u>	<u>Initiating Pressure (psid)</u>	<u>Total Number of Components</u>	<u>Number Assumed Available</u>	<u>Core</u>
Charging pumps	1000 to 2750	2	1	
HPIS pumps	1500	2	1	
Accumulators	200 to 750	2	2	
		or 2	1	
		or 3	2	
		or 4	3	
LPIS pumps	120 to 600	2	1	

[a] Designed reflooding rate for shutdown plant calibration of ECC capacity.

[b] Minimum guaranteed rate for systems assumed available.

accumulators (storage tanks) and low capacity pumping systems deliver the ECC to the various ECC subsystems depends on the size of the primary system rupture. High pressure systems, although utilized for all break sizes, are intended primarily for small breaks. Low pressure systems are used when breaks are sufficiently large that the primary system pressure falls below the pressure required for long term core reflooding. The accumulator systems, although utilized for all break sizes, are intended for which the primary system depressurization falls below the pressure of the gas dome. The pressure varies among reactor manufacturers and ranges from 750 psia to approximately 200 psia.

Plumbing configurations vary considerably among manufacturers with respect to certain features of emergency core cooling system performance. Particularly, the geometry of the upper and lower plenum of the reactor vessel, the method of ECC injection into the primary system, and the location of the water are important features that vary. The varying configurations are illustrated in Figure 1. For two manufacturers inject emergency coolant into each inlet pipe, the injection angle, and the injection rate for the two different plant designs. The ECC system of a third PWR manufacturer injects the coolant into the upper region of the inlet annulus.

Figure 2 illustrates, for a range of break sizes, the various system components available to assure adequate ECC delivery for pipe break sizes. The typical unimpeded delivery capability for each of the systems is shown in Figure 3.

U.S. designs are very similar in the design of the primary coolant circuit and reactor vessel. The ECC systems are designed to deliver coolant into both the outlet and inlet pipes.

TABLE A-1

DELIVERY CAPABILITY FOR THE VARIOUS PWR-ECC SYSTEMS

<u>Total Number of Components</u>	<u>Number Assumed Available</u>	<u>Designed Cold Vessel Core Flooding Rate [a] (in./sec)</u>
2	1	~ 0.1
2	1	~ 0.1
2	2	~ 8 [b]
or 2	1	
or 3	2	
or 4	3	
2	1	~ 1

* shutdown plant calibration of ECC capacity.

† systems assumed available.

2

Figure A-2. PWR Configurations for Various Manufacturers' Designs.

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Figure A-2. PWR Configurations for Various Manufacturers' Designs.

ANC-B-858-H

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Figure A-3. Typical Range of Assumed Core Protection Provided by Action of Combination of Systems.

BWR ECC DESIGNS

A schematic representation of a contemporary U.S. BWR system is shown in rough scale in Figure A-4. The reactor core is enclosed within an inner vessel in an arrangement similar to that of the PWR in that the system can be isolated (without coolant penetration) to a level about two-thirds the core height. The actual level is dictated by the elevation of the jet pumps.

The 1969 product line BWR emergency cooling network consists of one high pressure core spray (HPCS), one low pressure core spray (LPCS) system, a low pressure core spray (LPCS) system, and an automatic depressurization system (ADS). The two core spray systems are designed to provide both top spray cooling and bottom flooding, the accumulation in the lower plenum of the coolant not evaporated during spraying. The LPCS system refloods the core from the core top. All emergency coolant injection subsystems utilize pumps for coolant injection in contrast to emergency coolant storage systems. Because of the various reactor vessel configurations possible in a BWR, pump system performance characteristics vary from plant to plant. Presented in Table A-II for reference purposes is a description of the emergency cooling systems in a standard 2400 MW(t) BWR plant.

The emergency cooling systems are designed to operate over a complete spectrum of break sizes including a full range of accident. A performance capability chart is shown in Figure A-5 for both a recirculation line break and a steam line break.

e Injection Systems
ems

f 2 High Pressure Injection Systems
f 2 Makeup Systems

1 of 2 Low Pressure Injection Systems
3 of 4 Accumulators

1 of 2 High Pressure Injection Systems
1 of 2 Makeup Systems
3 of 4 Accumulators

Systems 1 of 2 Low Pressure Injection Systems
3 of 4 Accumulators

ssumed Core Protection Provided by Action of Combination of PWR-ECC

contemporary U.S. BWR system is shown in rough scale in Figure A-4. As shown, the vessel in an arrangement similar to that of the PWR in that the system can be liquid filled about two-thirds the core height. The actual level is dictated by the elevation of the top of

gency cooling network consists of one high pressure core spray (HPCS) system, a low m, a low pressure core spray (LPCS) system, and an automatic depressurization system e designed to provide both top spray cooling and bottom flooding, the latter th. ough e coolant not evaporated during spraying. The LPCI system refloods the core by deluge nt injection subsystems utilize pumps for coolant injection in contrast to gas driven PWR cause of the various reactor vessel configurations possible in a BWR, pump capacity and rom plant to plant. Presented in Table A-II for reference purposes is a data summary for ard 2400 MW(t) BWR plant.

e designed to operate over a complete spectrum of break sizes including the design basis is shown in Figure A-5 for both a recirculation line break and a steam line break.

Figure A-4. Typical U.S. BWR System.

The current German (AEG) version of the BWR incorporates an integral mechanical pump in place of the recirculating lines and thus also eliminating a recirculating line break. Multiple coolant makeup powered from off-line sources act as the ECC supply in a flooding mode similar to that of the PWR. A pipe makeup water line or rupture of a steam line creates a DBA condition which for this BWR concept results in a depressurization time of several hundred seconds as compared to a calculated 30-second depressurization time for the PWR.

The current Swedish (ASES-ATOM) version of the BWR also incorporates internal mechanical pumps. Multiple coolant injection systems act as the ECC supply to the vessel. The DBA condition for this BWR concept results in an extended depressurization time.

Figure A-4. Typical U.S. BWR System.

on of the BWR incorporates an integral mechanical pump in place of the jet pump thereby
 us also eliminating a recirculating line break. Multiple coolant makeup systems normally
 the ECC supply in a flooding mode similar to that of the PWR. A pipe break of a single
 eam line creates a DBA condition which for this BWR concept results in a calculated
 seconds as compared to a calculated 30-second depressurization time for a U.S. BWR.

M) version of the BWR also incorporates internal mechanical pumps. A combination of
 the ECC supply to the vessel. The DBA condition for this BWR concept also results in an

TABLE A-II

STANDARD PWR PLANT CORE STANDBY COOLING SYSTEM DELIVERY CAPABILITY

Function	Number Installed	Capacity (%)	Design Flow for Each System	
			Flow	Pressure ^[a] (psid)
HPCS	1	100	1300 gpm	1130
			4625 gpm	200
Automatic depressurizing valves	6	20	800,000 lb/hr	1125
LPCS	1	100	4625 gpm	119
LPCI	3	33-1/3	4970 gpm	20

[a] Differential pressure between the reactor vessel and the primary containment, in square inch.

Figure A-5. Typical Range of Assumed Core Protection Provided by Action of Combination Systems.

TABLE A-II

PLANT CORE STANDBY COOLING SYSTEM DELIVERY CAPABILITY

Capacity (%)	Design Flow for Each System		Pressure Range (psid)
	Flow	Pressure ^[a] (psid)	
100	1300 gpm	1130	1130 to 0
	4625 gpm	200	
20	800,000 lb/hr	1125	1130 to 0
100	4625 gpm	119	265 to 0
33-1/3	4970 gpm	20	225 to 0

between the reactor vessel and the primary containment, in pounds per

STEAM LINE BREAK

Auto-Relief

Maximum Steam Line
Break Area

Pressure Core Spray (HPCS)

Pressure Coolant Injection (LPCI)

Pressure Core Spray (LPCS)

0.8 1.0 1.2 1.4 1.6 1.8 2.0 2.5 3.0 3.5 4.0

Scale Changes

LIQUID LINE RELIEF

Auto-Relief

Maximum Recirculation
Line Break Area

Pressure Core Spray (HPCS)

Pressure Coolant Injection (LPCI)

Pressure Core Spray (LPCS)

0.8 1.0 1.2 1.4 1.6 1.8 2.0 2.2 2.4

Line Break Area (ft²)

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Assumed Core Protection Provided by Action of Combinations of BWR-ECC

GENERALIZED DESCRIPTION OF LOSS-OF-COOLANT ACCIDENT FOR PWR'S AND BWR'S

The events and processes of an LOCA are developed to illustrate the system behavior and phenomena accounted for by the calculational methods in order to prescribe the design and performance requirements for PWR-LOCA BEHAVIOR

PWR-LOCA BEHAVIOR

Figure B-1 depicts the generalized LOCA behavior for a postulated large break in one loop of a multiloop coolant circuit. This characterization of the accident is derived from many calculations carried out for various break locations for current PWR designs. The numbers on the figure are indexed to the following description of the loss-of-coolant process.

Figure B-1. Generalized Loss-of-Coolant Behavior for Large Pipe Breaks in PWR

ANALYSIS OF LOSS-OF-COOLANT ACCIDENT FOR PWR's AND BWR's

A are developed to illustrate the system behavior and phenomena which must be in order to prescribe the design and performance requirements for ECC systems.

CA behavior for a postulated large break in one loop of a multiloop PWR primary
e accident is derived from many calculations carried out for different pipe break
umbers on the figure are indexed to the following description of the course of the

Loss-of-Coolant Behavior for Large Pipe Breaks in a PWR.

Immediately following the pipe break, as the primary coolant is expelled from the rupture, the system subcooled depressurization (1) causing the flow within the reactor core to accelerate for an outlet break (2) inlet break (3). As the system depressurization continues (4) the local fluid saturation pressure is reached and an attendant fluid density decrease, occurs in the core as steam bubble growth is initiated. Within the core fluid density (moderator loss) causes the core power generation to decline within a few hundred milliseconds to decay heat power level (approximately 6% of the operating power).

For the inlet break conditions the reduced core flow commensurate with coolant voiding in the core can decrease in heat transfer from the fuel to the coolant and initiate the critical heat flux (CHF), or departure from nucleate boiling (DNB) (5). For the outlet break condition the core flow increase (2) tends to offset the density decrease and is preserved for an extended period until the local fluid conditions within the core are degraded sufficiently that CHF is reached. For either break, the abrupt decrease in heat transfer (5) (6) allows the large amount of stored energy to redistribute with a resultant rapid increase in fuel cladding temperatures (7) (8).

For the inlet break condition, at several seconds into the depressurization process the core mass flow rate is reduced because of nearly balanced fluid resistance paths to the break. For the outlet break condition, the flow from the core region is markedly lower and results in a continued significant upward core flow (10). The core flow histories (9) (10) respectively influence the cladding temperature histories for the inlet and outlet break conditions.

As the stored thermal energy within the fuel becomes redistributed the cladding temperature rise is limited. The cladding temperature may decline slightly (11) (12) as the competitive effects of continued fission product decay heat and amount of heat transfer exist for a few seconds. As the coolant conditions within the core continue to degrade the cladding temperature rises (13) (14) commensurate with adiabatic conditions dictated by the local fission product decay heat.

When the cladding temperature exceeds $\sim 1200^{\circ}\text{F}$ for either break condition, structural distortion, such as ballooning, may develop. Ballooning is postulated to result from a combination of the decreased strength of the cladding (due to temperature increases) and the increasing differential pressure between the internal fuel rod pressure and the system pressure.

As the coolant is expelled into the containment structure surrounding the reactor, the primary system depressurizes with an accompanying decrease of liquid level within the reactor vessel (15). When the system pressure falls below the gas dome pressure within the ECC accumulators (or core flooding tanks), relatively cold auxiliary coolant is injected into the appropriate inlet piping (or upper core barrel region) in an attempt to replenish the liquid inventory within the reactor vessel.

For an outlet break condition, soon after accumulator injection begins, the liquid inventory in the core is replenished to the bottom of the core (16). Core flooding is maintained by the low pressure coolant injected from the accumulator inventory is spent.

For an inlet break, some backflow from the core and continued boiloff of the liquid in the lower plenum plug up the downcomer which tends to inhibit the entry of auxiliary coolant to the lower plenum. In addition, the inlet pipes of the unbroken loops tends to entrain some of the injected coolant and this entrained coolant is carried into the downcomer annulus to the break. These conditions lead to the postulated "accumulator ECC injection" process. As decompression continues and the system steam flow rates decrease, the influence of gravity overcomes the effects of the lower plenum plug and the lower plenum begins to fill (17).

As the lower plenum fills and coolant reaches the bottom of the core, steam begins to be generated. Some liquid, rises in the core and cools the cladding. For the inlet break, the steam must escape from the core through the steam generators and pumps in order to reach the system vent, or pipe break (Figure A-1 of Appendix A) in passing through the various system components and particularly the steam generator where additional energy is transferred to the secondary system, is impeded by friction. The frictional pressure drop can reach a value of several pounds per foot of pipe, creating a backpressure on the reflooding process which competes against the head of water in the downcomer attempting to enter the core. The downcomer head in most current reactor designs can develop to a maximum head of ~ 70 feet. The process is relatively steady-state. If oscillatory effects occur as a result of the coupling between the inertia of gravity on the liquid in the downcomer and the coupling between the inertance of the liquid in the core and the compliance of the compressible steam volume above the flooding front, the average driving head could be less, thus lowering still further the time-averaged flooding rate within the core. The oscillatory effects, however, provide improved heat transfer in the early part of the flooding process.

plant is expelled from the rupture, the system experiences a rapid
for core to accelerate for an outlet break (2) or decelerate for an
the local fluid saturation pressure is reached and fluid flashing, with
a bubble growth is initiated. Within the core region the decreasing
to decline within a few hundred milliseconds to the fission product
er).

mensurate with coolant voiding in the core can cause a large abrupt
ate the critical heat flux (CHF), or departure from nucleate boiling
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the break. For the outlet break condition, the fluid resistance to the
continued significant upward core flow (10). These differences in the
temperature histories for the inlet and outlet break conditions.

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effects of continued fission product decay heating and some limited
at conditions within the core continue to deteriorate the cladding
ons dictated by the local fission product decay heat rate.

her break condition, structural distortion, such as ballooning of the
a combination of the decreased strength of the cladding (as the
between the internal fuel rod pressure and the decreasing external

ture surrounding the reactor, the primary system continues to
within the reactor vessel (15). When the system pressure decreases
core flooding tanks), relatively cold auxiliary coolant is injected
an attempt to replenish the liquid inventory in the bottom plenum

injection begins, the liquid inventory in the bottom plenum is
maintained by the low pressure coolant injection systems when the

continued boiloff of the liquid in the lower plenum cause steam flow
ry coolant to the lower plenum. In addition, the steam flow in the
e injected coolant and this entrained coolant is then carried around
ad to the postulated "accumulator ECC bypass" situation. As
case, the influence of gravity overcomes the entrainment forces and

n of the core, steam begins to be generated. The steam, entraining
e inlet break, the steam must escape from the system by passing
system vent, or pipe break (Figure A-1 of Appendix A). The steam,
arly the steam generator where additional energy is added from the
e drop can reach a value of several pounds per square inch causing a
the head of water in the downcomer attempting to drive coolant
signs can develop to a maximum head of $\sim 7\frac{1}{2}$ psi if the flooding
is a result of the coupling between the inertance and the force of
between the inertance of the liquid in the downcomer and the
g front, the average driving head could be less than the steady-state
dding rate within the core. The oscillatory flooding front could,
flooding process.

The competing effects of the limited driving head for flooding and the backpressure from the exiting postulated steam binding problem. The limited flooding rate for the inlet break, resulting from steam binding heat transfer in the core relative to that which would exist for higher flooding rates for an outlet break. With effects are involved for the inlet break, additional subtleties, such as the effects of containment backpressure on heat transfer and such as the compressible-flow acceleration-pressure drops due to energy being transferred from the side of the steam generator to escaping steam from the primary system, become important.

The temperature that the fuel cladding can attain without loss of structural integrity is determined, for example, by the amount of oxygen taken up by the cladding during metal-water reactions which become significant above 1800°F. At 1800°F the reaction rate is low but as the temperature increases to 2000°F and above, the rate increases rapidly. At 2300°F, for example, the oxygen uptake is such that reaction durations exceeding several minutes result in sufficient embrittlement that upon quenching of the fuel cladding by ECC, the structural integrity of the cladding cannot assure a definable heat transfer geometry within the hotter regions of the core.

The foregoing, intended to depict the general system behavior during an LOCA for a PWR, has been based on conditions which are expected to establish the ultimate requirements for ECC system design. The magnitude of effect of break size and location on the DBA is shown in Figure B-2. The calculations for developing the figure were performed by the reactor manufacturers and those performed independently by Aerojet Nuclear Company in conjunction with the program planning for the Loss-of-Fluid Test (LOFT). Apparent from the figure is the dominant influence of break size in determining the requirements of ECC designs for PWR's. However, the very largest break should not be considered as demanding on ECC design for all PWR's.

BWR LOCA BEHAVIOR

Figure B-3 depicts the generalized loss-of-coolant accident behavior for a postulated break in either the primary or the steam outlet lines of a contemporary boiling water reactor system. The numbers on the figure correspond to the following description of the course of the loss-of-coolant process.

Immediately following a steam line or recirculation line break of a BWR the system experiences a rapid depressurization because a significant amount of the fluid in the system during operation is at saturation, the remainder being slightly subcooled. The loss of one recirculating loop causes the core mass flow to drop rapidly from the initial value (1) as the other systems continue to provide coolant supply to the lower plenum, since a large volume of steam, at the outset, the depressurization process is relatively slow (2), and at several seconds into the accident the isolation valves in the outlet line close requiring that all system coolant exit from the pipe break region. Since the version of the BWR incorporates the internal jet pump design, all pipe breaks, including recirculation and primary line breaks, produce the effect of an outlet break in a PWR; that is, the depressurizing coolant flows in the normal direction through the core as illustrated in the figure.

At the approximate time the liquid level within the reactor drops to an elevation at which the jet pumps are no longer effective, the mechanical pumps in the recirculation line are coasting down and shortly cavitate dropping the core mass flow to zero. These conditions promote coolant starvation within the reactor core and initiate CHF in the hotter regions of the core. The liquid level in the outer annulus around the core barrel drops to the elevation of the recirculation line outlet and the break becomes steam and the depressurization rate is increased (4). Simultaneously, the saturation pressure in the lower plenum of the reactor vessel is reached and a process called lower plenum flashing is initiated (5). During this process, steam tends to flash violently and surges into the core region. The potential for significant cooling exists since the temperature rise may be terminated (6) and the cladding temperature may be restored to the fluid saturation temperature. As coolant inventory in the lower plenum is spent from flashing, the system pressure continues to decline and the temperature again rises in the hotter zones of the core and experiences DNB a second time (7). The cladding temperature rapidly until the energy redistribution within the fuel pin is complete at which time decay heat limits the rate of temperature rise (8). Shown in the figure for completeness is the continued temperature rise from the early event of CHF to the temperature limit (9) assuming no cooling due to lower plenum flashing. As the system pressure continues to decline, a spray system above the reactor is initiated and top spray flow is developed at about 260 psia. The spray system provides some convective heat removal from the fuel pins. The resulting steam from the spray system provides some convective heat removal from the cladding surface. This cooling process tends to slow the temperature rise. The lower reactor vessel plenum is filled by the accumulated spray and LPCI system coolant inventories and a condition similar to that for the PWR is initiated.

lead for flooding and the backpressure from the exiting steam give rise to the flooding rate for the inlet break, resulting from steam binding, causes decreased flooding rates for higher flooding rates for an outlet break. Where these competing subtleties, such as the effects of containment backpressure on entrainment and acceleration-pressure drops due to energy being transferred from the secondary to the primary system, become important.

Retention of structural integrity is determined, for zirconium-clad fuels, during metal-water reactions which become significant at temperatures above 2000°F and above, the reaction rate increases such that reaction durations exceeding several tens of seconds cause failure of fuel cladding by ECC, the structural integrity of the cladding is insufficient to the hotter regions of the core.

System behavior during an LOCA for a PWR, has emphasized the DBA estimate requirements for ECC system design. The magnitude of the calculated core exit temperature in Figure B-2. The calculations for developing the figure include those by Aerojet Nuclear Company in conjunction with the design and analysis. Apparent from the figure is the dominant influence of the large inlet break in PWR's. However, the very largest break should not be concluded to be the most

Plant accident behavior for a postulated break in either the liquid recirculation or boiling water reactor system. The numbers on the figures are indexed to the accident process.

In the case of a BWR the system experiences a very limited subcooled liquid flow in the system during operation is at saturation conditions, with the recirculating loop causes the core mass flow to drop rapidly to about one-half to provide coolant supply to the lower plenum, since a large volume of the vessel is relatively slow (2), and at several seconds into the transient, the steam flow at all system coolant exit from the pipe break region. Since the contemporary pump design, all pipe breaks, including recirculation and steam line breaks, in a PWR; that is, the depressurizing coolant flows in the normal upward direction

When the reactor drops to an elevation at which the jet pumps become uncovered, the system is coasting down and shortly cavitate dropping the core mass flow to nearly zero. This causes the reactor core to initiate CHF in the hotter regions of the core (3). As the core drops to the elevation of the recirculation line outlet, the flow out the core is increased (4). Simultaneously, the saturation pressure of fluid in the lower plenum called lower plenum flashing is initiated (5). During this process the fluid in the lower plenum region. The potential for significant cooling exists such that the cladding temperature may be restored to the fluid saturation temperature. As the system pressure continues to decline and the cladding temperature rises, the system pressure continues to decline and the cladding temperature rises until the fuel pin is complete at which time decay heat limits the rate of the temperature rise from the early event of CHF (3) on through to the lower plenum flashing. As the system pressure continues to drop, a high pressure spray flow is developed at about 260 psia. The spray tends to wet the fuel pins. The resulting steam from canister wetting also cools the cladding surface. This cooling process tends to slow the heatup rate until the spray and LPCI system coolant inventories and a core reflooding process

Figure G-2. Generalized Comparison of Maximum Cladding Temperature for Various Primary Break Conditions in a PWR.

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Figure B-3. Generalized Loss-of-Coolant Behavior for Large Pipe Breaks in

For the lower cladding temperature history (8), the cladding temperature turnaround (9) results in flooding at the bottom of the core. For the upper cladding temperature history (8'), the effects of metal-water reaction are seen to cause a significantly increased rate of temperature rise prior to the event of flooding (9').

For the steam line break, the various events are depicted by dashed lines in Figure B-3. The pressure drops (10) considerably more rapidly than for the recirculation line break. Since steam venting is taking place at the reactor vessel the liquid fraction in the system remains high and all recirculation line systems continue to operate. The flow is thus seen to continue (11); however, the flow eventually decreases as the pressure decay causes the recirculation pumps to cavitate. The flashing process continues to provide reasonable core flow and at least sufficient steam. The attendant cladding temperature indicates that nearly all the stored energy within the fuel is removed under the coolant conditions can no longer support the heat transfer required to keep the cladding temperature at saturation temperature (12). At this point the cladding temperature begins to rise as a result of the small stored energy and decay heat energy. Up to this time most of the fluid lost from the system as a result of the break has been steam and some two-phase mixture; that fraction of liquid having insufficient enthalpy to flash remains. The additional inventory necessary to fill the lower plenum to the bottom of the core and effect early temperature rise (13) is, therefore, considerably less than for the recirculation line break. The general behavior, effect of metal-water reaction, and embrittlement are sufficiently similar to those of a PWR that are unwarranted.

Cladding Temperature Behavior for Large Pipe Breaks in a BWR.

For curve (8), the cladding temperature turnaround (9) results from the initiation of natural circulation. For curve (8'), the effects of metal-water reaction energy are included in the temperature rise prior to the event of flooding (9').

Curves (12) and (13) are depicted by dashed lines in Figure B-3. The pressure is seen to decrease after the recirculation line break. Since steam venting is taking place at a higher region of the core, the pressure remains high and all recirculation line systems continue to operate. Significant core flow eventually decreases as the pressure decay causes the recirculating mechanical pumps to provide reasonable core flow and at least sufficient steam cooling to the core. At nearly all the stored energy within the fuel is removed until, at the worst case, the heat transfer required to keep the cladding temperatures near the coolant saturation temperature begins to rise as a result of the small amount of remaining liquid. At this time most of the fluid lost from the system as a result of a steam line break has been removed. A portion of liquid having insufficient enthalpy to flash remains in the lower plenum. This liquid flows from the bottom of the core and effect early turnover of the cladding temperature. The general behavior of the fuel cladding, however, is sufficiently similar to those of a PWR that additional discussion is

2

Unlike the process for the PWR, the effect of steam binding does not appear to inhibit the rate of flood need pass only through relatively small frictional pressure drop paths (Figure A-4 of Appendix A) on its way

Figure B-4 presents the calculated peak cladding temperature as a function of break area for steam line breaks for two separate single failure conditions in a contemporary BWRB-1. One case considers failure of considers failure of a diesel generator. These graphs are considered representative of a single-failure criterion a cladding temperature and should not be considered to be restrictive in defining the capability of a system systems. As would be expected, a general trend toward higher peak cladding temperatures occurs as break smallest breaks, no core heatup occurs. The exact shapes and magnitudes of the temperature curves for this depend to a large extent on such factors as the analytical techniques used in the calculations, assumptions correlations, and the particular single failure condition considered.

REFERENCE

B-1 Core Standby Cooling Systems for the General Electric 1969 BWR Standard Plants, NEDO-10183 (

Figure B-4. Generalized Comparison of Maximum Cladding Temperature for Various Pipe Breaks in a BWR.

on binding does not appear to inhibit the rate of flooding because the steam
the drop paths (Figure A-4 of Appendix A) on its way to the break.

temperature as a function of break area for steam line and recirculation line
contemporary BWRB-1. One case considers failure of the HPCS; the other
considered representative of a single-failure criterion approach to maximum
to be restrictive in defining the capability of a system or combination of
higher peak cladding temperatures occurs as break areas increase. For the
es and magnitudes of the temperature curves for this type of representation
typical techniques used in the calculations, assumptions on heat transfer
considered.

Electric 1969 BWR Standard Plants, NEDO-10183 (May 1970).

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Maximum Cladding Temperature for Various Pipe Break Conditions