UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Reference
1086

APR 28 1987

ADR: PDR
R. EMRIT

MEMORANDUM FOR: Bill M. Morris, Director
Division of Regulatory Applications, RES

FROM: Brian W. Sheron, Director
Division of Reactor and Plant Systems, RES

SUBJECT: LOCA CONCERN OF SCE EMPLOYEE

A Mr. Dan Johnson of Southern California Edison (SCE) has recently written two letters to a Mr. S. P. Kalra of EPRI expressing concern about steam generator tube ruptures in conjunction with a large LOCA. The concerns are described in the enclosure and focus on the possibility of a recriticality due to primary system dilution with unborated water from the secondary system, with subsequent containment overpressurization and failure. These concerns are stated to be a result of Mr. Johnson's review of the proposed Regulatory Guide on ECCS methods which accompanies the proposed ECCS rule change.

Although Mr. Johnson sent the enclosures to EPRI, he has also sent copies directly to a number of NRC employees. I have spoken with Mr. Ken Baskin, Vice President for Nuclear at SCE, to determine if Mr. Johnson had a safety concern but was unaware of how to contact the NRC, or to otherwise understand his motives for sending selected NRC staff copies of his letters to EPRI.

According to Mr. Baskin, Mr. Johnson is fully aware of his responsibility to write directly to NRC if he believes a safety deficiency exists. He has chosen not to. Therefore, I do not believe his letters constitute an allegation but rather should be considered as comments on the proposed ECCS rule.

However, in order to ensure the issue is fully evaluated, I believe it is prudent and in the best interests of the agency to classify the issue as a generic safety issue and prioritize it consistent with Office Letter #40 of the old NRR organization. If the issue is prioritized as "medium" or "high," I believe you should work in cooperation with the Reactor and Plant Systems Branch to develop a resolution in conjunction with the final ECCS rule.

1/1

for Brian W. Sheron, Director
Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research

DF03

cc: See next page.

NUREG Add. R. EmRiT, RES CF Subj RES

Bill M. Morris

APR 28 1987

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Dan Lewis Johnson
Engineer
SONGS Units 1, 2, and 3

March 26, 1987

Mr. S. P. Kalra
Project Manager
Safety Technology Department
Nuclear Power Division
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, California 94303

Subject: Proposed ECCS Regulatory Change

References:

1. Letter from S. P. Kalra, EPRI to D. L. Johnson, Southern California Edison Company, dated March 19, 1987 of Subject: ECCS Regulation Change
2. Emergency Core Cooling Systems; Revisions to Acceptance Criteria - Proposed Rules Federal Register, Vol. 52, No. 41, March 3, 1987
3. Secondary Side Transport and Retention of Radioactive Species - STARRS Computer Code - Retention of Radionuclides in U-Tube SG during SGTR events. RP 2453-4, October 20, 1986

Dear Mr. Kalra:

Thank you very much for your response (Reference 1) to my letter (Attachment 1) pertaining to LBLOCA with consequential SGTR in light of the proposed rules associated with Emergency Core Cooling Systems; Revision to Acceptance Criteria (Reference 2).

I have reviewed the BNL Report (Attachment 2) and would like to relay several comments to you which I have also previously discussed with Mr. L Neymotin of BNL.

Briefly, the heat levels (decay heat) used in the BNL experiment are not realistic in a large PWR core undergoing LBLOCA transient with SGTRs. During this transient, even with all rods in, large portions of the core become supercritical due to the affected secondary side - steam generator inventory discharging unborated steam/water to both the top and bottom of the core. Local power levels can be enormous. Furthermore, the accumulator inventory which reaches the core is not adequate in immediately mitigating this transient. As a result of this, large portions of the core are no longer amendable to cooling from either the high or low pressure safety injection systems. Furthermore, higher than designed containment pressures are reached resulting in increased backpressure during the reflood stage of the LOCA.

It is obvious that unresolved steam generator integrity safety issues should be fully resolved such that the margin of safety offered to the general public is not lessened in the hypothetical LBLOCA safety analysis.

Thank you for the material (Reference 3) which you have provided us.

Sincerely yours,

Dan Johnson 3-26-87

D. L. Johnson
Engineer
Southern California Edison Company

Attachment

cc: ECCS Rulemaking Distribution
Revised Regulatory Guide/SECY 86-318
L. Neymotin, Brookhaven National Labs

ATTACHMENTS

March 2, 1987

Mr. S. P. Kalra
Project Manager
Nuclear Power Division
Electric Power Research Institute
3412 Hillview Avenue
P. O. Box 10412
Palo Alto, California 94303

Dear Mr. Kalra:
After reviewing in detail the draft of the ECCS Rulemaking Regulatory Guide, I would like to make the following comments based on FIV problems and other steam generator integrity issues with CE System 80 and 3410 MW_t vertical U-tube steam generators.

1. Specifically, the models should be capable of predicting peak clad temperatures during a LBLOCA with concurrent steam generator tube ruptures. The steam generator tube ruptures are sequential failures in the primary to secondary pressure boundary as a result of the LOCA and SSE hydroloads. Such tube ruptures can occur due to large volume defects (associated with tube to tube support interaction and wear) and other defects which are located in regions of high rarefaction stress intensity as a result of the depressurization wave. Tests performed in the Semiscale Mod 1 System (attachment 1) identifies an extremely narrow band of tube rupture flows (that is, flow from between 12 and 20 tubes) resulting in significantly higher peak cladding temperatures. This was due primarily to the tube rupture flow, resulting in a retarded reflooding of the core which, in turn, caused later cladding temperature turnaround times and corresponding higher peak cladding temperatures.

2. During Engineering reviews of ECT data, for example (attachment 2), those tubes identified and characterized as having large volume fretting defects (where leak before break criteria does not apply) in excess of their degradation limits governed by LOCA hydroloads, should be assumed to shear during a LBLOCA safety analysis of the FIV problem. Peak clad temperatures should then be determined using models that take into account ECT uncertainties, i.e., copper plateout from feedwater heater trains, U-bend uncertainties, etc. ECT frequencies and sample size should be specified in a plant specific license amendment to assure that peak clad temperatures will be maintained less than 2200°F with a 95% confidence level. The reasons are obvious as radiological consequences can be severe as the clad rupture levels are higher than previously analyzed, and a direct release path is developed from the fuel pellet to the outside air via the secondary plant and multiple sequential SGTRs.

Attachment 3 describes a potential problem should large volume fretting defects develop in regions of high LOCA rarefaction stress intensities.

3. ECCS performance and rulemaking should also be assessed based on MSLB induced SGTRs in essentially the same fashion.

Should any further questions exist, feel free to contact me at (714) 368-9793.

Sincerely,

Dan L Johnson
3-2-87

D. L. Johnson
Engineer

DLJ:4011I/sas

cc: ECCS Rulemaking Distribution
Revised Regulatory Guide/SECY 86-318

L.Y. Neymotin and U.S. Rohatgi
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DRAFT

STATUS: WORKING VERSION

EFFECT OF STEAM GENERATOR TUBES RUPTURE ON PWR LOCA

1. Introduction.

The large break loss of coolant accident is a design based accident which has been analyzed for each of the plants operating in USA. Even with the conservative guidelines of the Appendix K of 10CFR50, the peak clad temperatures have been shown to be smaller than 2200 F. However, as the existing plants are aging, a new concern is created due to the possibility of steam generator tubes becoming weak and rupturing in a course of a hypothetical LOCA.

Steam generator tube ruptures had occurred in many existing plants but did not cause any serious abnormal conditions and plants have been safely operated. However, tube

rupture incident could happen during LOCA due to condensation-induced oscillations expected in Refil/Reflood phases of the accident. This could increase the severity of the large break LOCA. In fact, this type of behaviour was observed in some of the semiscale large break LOCA tests with steam generator tube rupture. The peak clad temperature in these tests had an extremum for a certain critical number of tube ruptures. However, these observation can not be directly extrapolated to a full scale plant due to many scaling compromises made in designing the semiscale facility. Nevertheless, it remains to be shown whether similar behavior will be observed in the full scale plant and what would be the degree of severity of this scenario.

The objective of this report is two-fold. The first objective is to briefly overview the main features of the accident scenario relating it to the semiscale steam generator tube rupture experiments. The analysis is done from the standpoint of the experiments representativeness of a typical PWR in terms of modeling the accident sequence as well as scaling the thermal hydraulic phenomena. Secondly, after the dominant phenomena controlling the accident are established, a conservative estimate of the effect of steam generator tubes rupture on the reflood progression and peak clad temperature will be performed.

Dominant thermal hydraulics phenomena are described in Chapter 2. A brief description of the semiscale test facility, transient scenario, and analysis of results and scaling

considerations can be found in Chapter 3. This chapter also discusses some questions of the typicality of the semiscale Mod-1 experiments vs. the real PWR system transient scenario. Finally, a symplified analysis predicting the critical number of tube ruptures and the reflood delay due to steam generator tube ruptures is presented in Chapter 4.

2. Dominant Phenomena.

Description of dominant phenomena can be started from the time just following the blowdown stage of a LOCA after the emergency core cooling (ECC) system had been initiated. Until the ECC water fills up the lower plenum, the steam from the secondary of the ruptured steam generator can follow two paths escaping the vessel: ~~a~~ 1) through the lower plenum, downcomer (CCFL and downcomer gap width are important) and through the cold leg break, and ~~b~~ 2) through the broken loop SG, and through the RCP. The distribution of the flows depends on the relative hydraulic resistances along the two paths. During this period the flow in the core is negative.

As soon as the lower plenum is filled, the first path can be blocked. The water level in the downcomer rises, and the reflood stage may begin. The reflood will start only if the upper plenum pressure increase due to the flow resistance of path 2 does not overcome the downcomer static head. However, depending on the rupture flow rate and the

resistances of the simulated SG and RCP in the broken loop, pressure can increase and the core flow reverse which will retard the onset of the reflood stage. As it will be seen later, resistances of these two simulated components have been larger than expected from the point of view of the intact vs. broken loop scaling considerations.

After the reflood begins, the vapor generation in the core will begin. The effect of this is equivalent to an increase in the rupture flow rate: both lead to increase in the upper plenum pressure. The increase in pressure will evantually decrease the reflood rate and increase PCT.

After the ECC injection is initiated, the cold water in all four cold legs (mostly in the intact loops) comes into contact with the steam flowing either from the core region through the downcomer or through the hot leg and the steam generator primary. The condensation has a balancing effect on the vessel upper head pressure by offsetting the upper plenum pressere increase due to the SG rupture steam flowing into the vessel.

The amount of steam leaving the upper plenum and so the upper plenum pressure depend both on the condensation rate in the cold legs and downcomer as well as on the pressure losses in the intact loop steam generator and RCP (scaling of pressure losses in these components). Distortions in modeling of the loop resistances (on the higher side, in particular) can lead to increased pressure losses in the intact loop or in the simulated SG and RCP resulting in the upper plenum pressure increase and reflood retardation.

The phenomena in the downcomer region can be important for the lower plenum and downcomer filling rate. All ECC water is injected into the intact loop cold leg. The scaling of the DC and modeling of injection only through one integrated intact loop will distort the real flow pattern which is expected to develop in the downcomer when the injection is provided through the three intact loops. Also, boiling in the downcomer (increased area-to-volume scaling) will be over-modelled thus potentially reducing the filling rate through the steam-liquid interaction phenomena (CCFL). The downcomer water level history becomes an important parameter together with the upper plenum pressure by controlling the reflood progression.

Vapor generation in the ECC and steam generator secondary water heated by the metal structures in the downcomer and upper plenum regions has also some impact on the downcomer and upper plenum pressure histories. Vaporization in the downcomer may have a favorable effect on reflood by increasing the reflooding driving pressure head. However, pressure increase due to vaporization in the upper plenum can retard the reflood phase.

3. Semiscale Tests.

3.1 Description and Main Results.

The series S-28 of the Semiscale Mod-1 experiments simulated the phenomena in a PWR during a large break LOCA followed by the steam generator tube rupture. The objective of the experiments was to evaluate the effects of the SG tube rupture on progression of the core reflood and, eventually, on the core peak clad temperature (PCT) history. The results of these experiments were published in 1977 and 1978 [1,2,3].

The simplified schematic of the test facility is shown on Fig. 1. The PWR core is simulated using 40 electrically heated rods with 36 rods actually powered with an average power of 1.4 MWt in all S-28 series tests. The heated length of the rods was about 168 cm. One of the facility's loops (intact loop) represents the plant's three intact loops combined, and the second loop represents the plant's broken loop. The intact loop contained an active steam generator and the RCP, while the steam generator and RCP were simulated as passive flow resistances in the broken loop.

Initially the system was filled with water, warmed up to about 542 F in the cold legs using the core heater rods, and pressurized to about 2300 psi (158 MPa). The double ended cold leg break transients were initiated by breaking two rupture discs separating the vessel inlet and the simulated RCP outlet of the broken loop (200% break) from the suppression pool. Pressure in the pressure suppression tank represented the containment back-pressure of about 32.6 psi (2.25 MPa). At approximately 5 seconds into the transient the heater power was tripped to the decay heat level.

The HPCI system was initiated immediately after blowdown, while LPIS was triggered by the system pressure dropping to a set point of 1034 psi (71.3 MPa). The ECC water temperature was 102.6 F (25 C). The HPCI injection flow rates in the intact loop were around 0.025 kg/s except for the test S-28-2 [2] (6 tubes rupture) where for an unspecified reason it was 0.17 kg/s (see, for example, Figs. 2 and 3 showing the HPCI volumetric flow rates for tests S-28-2 and S-28-3 [4]; LPIS flow rate was approximately 0.3 kg/s for all tests (a typical LPIS history is shown on Fig. 4). It could be expected that the increased HPCI injection rate in the 6 tube rupture case would result in faster reflood and lower PCT.

There were 12 experimants in the S-28 series which differed mainly by the number of ruptured tubes in the intact loops steam generators. Key parameters of these experiments are shown in Table 1.

The steam generator secondary-to-primary flow caused by the rupture was simulated by injecting pressurized water from a heated accumulator at a fixed metered rate into the hot le piping of the intact loop steam generator. The injection l ne was connected to the hot leg piping between the SG hot ple um and vessel.

The injection flow rates varied in different tests of the Series S-28 from 0.054 kg/s to 0.544 kg/s which, assuming that the HEM model was applicable for the choked flow calculations, corresponded to single-ended rupture of between 6 to 60 tubes in the three intact PWR loops [5]. Core area scaling was chosen for the above calculations of

correspondence between the injected flow rate and equivalent number of tube ruptures. Implications of this choice are discussed later in 3.2.

Beside the variations imposed on the SG rupture flow rate, sensitivity of the peak clad temperature to the time of the rupture was also investigated. Two sets of experiments have been performed with the tube rupture at 40 and 60 seconds after the blowdown; these times are supposed to correspond to the beginning of refill and reflood, respectively. Comments on these sensitivity tests can be found in 3.3.

The principal objective of the S-28 series of experments was to the obtain information on the peak clad temperature magnitude and history during a LOCA complicated by a SG tube rupture occuring after the blowdown stage. Since the main parameter changing in these tests was the secondary-to-primary flow rate presented through an equivalent number of the ruptured SG tubes, the plot showing the PCT as a function of the number of ruptured tubes (Fig. 5) [5].

The two conclusions from the results were: a) the cladding temperature does not reach "... the temperature necessary to impar the structural integrity of PWR fuel rod cladding" [5] and b) there is a fairly narrow band of the secondary-to-primary flow rates when PCT reaches a high value of 1258 K as seen in Fig.5. The range of the tube rupture flow rates where the PCT has a potential of reaching high values correspond to rupture of between 12 and 20 SG tubes.

The main purpose of this report is to review the results and conclusions and attempt to evaluate the

applicability of the latter to a full scale PWR system undergoing a LOCA with a similar scenario.

3.2. Review of the Mod-1 Scaling Considerations.

There have been basically two alternatives in scaling the Semiscale test facility - linear and volume scaling. The linear scaling has certain positive features; for example, it preserves the loop transit time which is important in representation of the acoustic wave propagation as well as fluid transit time. However, a number of drawbacks in this approach make it less appropriate when comparing with the volume scaling. Some of the major disadvantages in linear scaling is a need in scaling of acceleration; also, linear scaling of lengths, areas, and volumes leads to a very small-sized test's geometry and very high values of the core power density.

Among the advantages of the volume scaling principle followed in designing of the Mod-1 facility are preservation of length (correct elevation representation is important in the gravity driven transients), preservation of the core geometrical parameters (rod diameters and pitch; the heated length was not preserved), preservation of time, velocities, and acceleration.

Unavoidable compromises, however, had to be made since it was impossible to strictly follow the volume scaling

principle and at the same time, for instance, maintain pressure losses representation in the pipes: the lengths had to be reduced to avoid excessively large pressure drops in the loop components.

Two studies of the Semiscale Mod-1 and Mod-3 test facility scaling effects have been performed at INEL in 1979 and 1980 [6-8]. Main issues considered problematic by these studies are the following:

a) Surface area to fluid volume ratios as well as the pressure losses are larger in the model components than in the full scale system;

b) Structural heat transfer is not represented properly;

c) Progression of very slow transients will be influenced by the oversized SG secondary;

d) Loop pumps do not have properly scaled locked-rotor resistances; they also do not have properly scaled performance curves. Degradation was much earlier than in the full scale pumps;

e) Heated length for the rod bundle is not scaled correctly;

f) The steam generator and pump in the broken loop are modeled as passive pressure loss components;

g) The downcomer gap is not scaled properly and is too narrow.

A few selected scaling issues are discussed below.

3.2.1 Structural Heat Transfer.

It is recognized that a large distortion (on the higher side) in the heat transfer area to the liquid volume ratio is introduced as compared to the full-scale system when a volume scaling is adopted. This would lead to inaccuracies in modeling of the heat transfer associated with the test facility's metal structures. So that the course of the fast transients (blowdowns) overly large amounts of heat will be deposited into the system liquid inventory, while in the case of slow transients, losses to the environment would be unreasonably large.

Because of the surface area-to-volume ratio in Semiscale facility is larger than in a real reactor, the vapor generation and condensation phenomena will be distorted. The overscaled vapor generation (or condensation) in the downcomer region will influence the system pressure responce and distribution which are important factors in the reflood progression.

3.2.2 Critical Flow.

There are two basic issues related to modeling of the tube rupture (secondary-to-primary) flow rate: estimation of the critical flow rate and dependency of this flow rate on the break location. As follows from [5] the HEM for choked flow was used to find correspondence between the flow rate injected into the SG hot leg in the experiments and the number of tubes

ruptured in the real system which would provide the corresponding flow rate. It should be pointed out that the HEM model can significantly underpredict the critical flow rate which would lead to misinterpretation of the PCT vs. number of rupture tubes results.

The core area scaling was used for the secondary-to-primary flow scaling in these evaluations. The alternatives for scaling are the core power, system volume, and system hydraulic resistance ([5], p. 99); the corresponding flow rate ratios (assuming the ratio to be unit in the case of the core area scaling) have been predicted to equal 0.43, 0.66, and 1.57, respectively.

This means that depending on the scaling criterion chosen for the SG break flow rate, the injection rate in the experiments simulating the SG primary-to-secondary flow can correspond to a different number of tubes ruptured in the real PWR SG. For example, the 16 tubes rupture test (S-28-8) with the rupture flow rate of 0.145 kg/s can be interpreted as 7 or 25 tubes break depending on whether the reactor power or system hydraulic resistance is chosen for the break flow scaling, respectively.

The uncertainty in the tube rupture flow rate scaling and modeling would produce uncertainty in interpretation of the location of the maximum on the PCT vs. number of ruptures curve (Fig. 5): the PCT curve could shift along the axis either to the right or to the left. This would make the conclusion on the band of secondary-to-primary flow rate which produces maximum PCT (12 to 20 tubes ruptures in the tests)

not appropriate for the full scale extrapolation.

Location of the SG tube ruptures can also influence the primary-to-secondary flow rate history.

3.2.3 Downcomer Gap Width and Flow Modeling.

The downcomer region representation can significantly impact the core reflood development. Importance of the downcomer hydraulics has been emphasized in another series of the Semiscale experiments ***footnote***). The volume scaling approach together with preservation of the vertical size utilized in the Mod-1 test facility would have resulted in a very narrow downcomer. However, in the facility the downcomer gap area is approximately three times larger than would be needed to satisfy the volume scaling requirement.

It should be pointed out here that in case of the liquid flow in narrow gaps (like the downcomer annulus of a scaled-down experimental facility) the wall friction pressure losses and Helmholtz instability are the phenomena controlling the liquid propagation in the theta and vertical directions (as opposite to the cases when the CCFL effects play the major role). The two former phenomena will begin to control the flow at some point as the width of the annulus decreases.

The implication of the said above is that depending on scaling of the diameter of the cold leg pipe, ECC flow rate, and the downcomer gap width one can find, for example, a CCFL controlled phenomena in the full scale system, and the wall friction and Helmholtz instability controlled phenomena - in a

scaled-down test facility.

Although a CCFL analysis can be performed for the case in consideration, it has to be realized that the available Wallis-type correlations [9] were developed for the idealized flow conditions and geometries. Applicability of those correlations to the three dimensional flow which develops when a jet enters an annulus normally to the channel walls is questionable. Depending on the flow parameters and geometry, one can conjecture development of an oscillatory flow pattern phenomenologically different from those in the Wallis correlation data base tests.

Theoretically, however, if one were to assume that the stable counter current flow conditions do develop in the downcomer annulus, the following correlation had to be considered

(Wallis corr.)

It follows from the correlation that at the fixed steam upflow rates, the liquid downflow rate will increase with the downcomer gap diameter. This indicates that having a too narrow gap would delay the lower plenum and downcomer fill-up, retard the reflood, and potentially elevate the PCT.

FOOTNOTE)

The alternative ECC injection tests [1], p.13, Series

5 have shown that the lower plenum injection is considerably more effective than the downcomer injection in terms of the core cooling and eventual quenching. Note, that switching to the lower plenum ECC water injection essentially eliminats effects of the downcomer flow phenomena and, to a great extent, of the inter-phase mass transfer effects.

Although the latter tests have not been done with the simultaneous SG tube rupture, the results are representative in terms of the influence of the flow conditions and inter-phase mass transfer phenomena in the downcomer region on the core reflood.

END OF FOOTNOTE

Finally, it should be noted that although the bundle heated length was smaller then in the prototype (168 cm vs. 366 cm), the linear power density in the experiments was preserved. In the S-28 series, the value for linear power density was approximately 21 kW/m. It can be expected that shortening of the heating length will result in a quicker quench. This, together with the fact that the bundle power should have been 1.9 MWt in order to agree with the volume scaling (it was about 1.4 MWt in the tests), will bring another degree of uncertainty in the issue of extrapolation.

3.3. <u>Typicality of the Mod-1 S-28 Test Series.</u>

It has been emphasized in all reports on the Semiscale Mod-1 experiments that the tests have been conceived primarily as a data base for the nuclear safety code assessment applications. The authors caution the potential data users from direct extrapolations of the results contained in the reports to full scale [1,5,8]. With this in mind, we consider below a few issues related to typicality of the SG tube rupture series of experimants. The typicality in this context refers to accuracy of the real reactors' components representation as well as of the scenario and boundary conditions modeled in the experiments.

3.3.1 Pumps Running

As it has been mentioned earlier, the intact loop RCP is left running at 1470 rpm thus providing positive driving force for the steam or two-phase flow. At the same time, the broken loop RCP is modeled as a passive resistance component. There is no clear justification for this differentiation in representing the intact and broken loop RCP's behavior. The potential effect of this scenario selection is a loss of typicality in the plant representation. It also provides an additional driving head for reflood which can enhance core cooling and bring the core PCT down.

3.3.2 Broken Loop Pump and SG Flow Resistances.

One of the system's distributed variables most important for the reflood progression during the transient is the pressure distribution in the vessel; this pressure distribution controls the water flow from the downcomer region through the lower plenum into the core. The reflood driving pressure head (RDPH) is defined as the difference between the sum of the cold leg pressure and the downcomer water static head, and the sum of the upper plenum pressure and the core static head. It is this pressure head in the vessel what affects the core inlet flow rate.

There are many system modeling parameters which control the sign and magnitude of RDPH. However the two major contributors to RDPH are the hydraulic resistances of the main circulation pumps and the primary sides of the steam generators.

Following the facility design, it should be expected that the steam generator and pump simulators (which are passive resistance devices in the tests) should have had the hydraulic resistance nine times bigger (9:1) than the corresponding values for the explicitly modeled steam generator and pump in the intact loop (we refer here to the locked-rotor resistances of the RCPs). This resistance distribution follows from the fact that the intact loop in the experiments is actually modeling three intact loops of the real system which implies that the flow rate through the test's intact loop primary has to be equal to three quarters of the total core exit flow.

In reality, however, the ratios between the

corresponding resistances were 24 to 1 for the steam generators and 41.5 to 1 for the main recirculation pumps in the broken and intact loops, respectively [1]. This distortion, together with the fact that the intact loop pump was left on running throughout the transient may contribute significantly to the uncertainty in extrapolating the results to a real PWR. From the qualitative point of view, an increase in the broken loop resistance can lead to elevated values of the upper plenum pressure and, consequently, to a delay of the reflood phase.

3.3.3 Time of tube rupture.

Time of the tubes rupture occurence is another parameter which introduces uncertainity in the Mod-1 tests scenarios. The results of the rupture timing sensitivity study showed that the time of initiation of the nitrogen injection into the intact loop cold leg was an equally (if not more) important parameter as the time of the SG rupture initiation. Another differentiating parameter was the SG flow rate. Figures 6,7, and 8 show typical cladding temperature histories. Also, a large scatter in the measured PCTs was observed even within the same range of test parameters and timing of the boundary conditions events.

Another issue related to typicality of the experiments is the choice of location of the tube ruptures: intact vs. broken loops. Although the latter choice should be less imposing in terms of the steam binding, its consequences have

to be investigated.

4. Simple Calculation of Reflood Delay.

The semiscale tests have shown that when the steam tube rupture occurrs during a large break LOCA, the course of the transient is significantly influenced by the number, time and location of the tube ruptures. The semi-scale facility has been designed with many scaling compromises so that only qualitative conclusions about the course of the transient and the dominant phenomena can be made. However, evaluation of results of the semiscale tests along with the available best estimate code calculations for the large break LOCA can help to estimate limiting values of the important parameters such as critical number of the tube ruptures and the consequent peak clad temperature in case of a full-scale plant.

It is generally understood that if there exists any possibility of steam tube rupture during a large break LOCA, it is most likely to occur during the refill or reflood stage of the transient because of the mechanical vibrations induced by condensation. The number of the tubes ruptured at that time can significantly affect the transient progression. The critical number of the tube ruptures is a function of the quality and flow rate at the location of tube break, and the capacity of the heat structures in the upper head and upper

plenum. The heat stored in the structures will vaporize the liquid coming into the vessel from the SG-secondary through the tube breaks. For the current estimate, it is assumed that the tube breaks occur at the end of the refill stage; the plant conditions at that time are taken from a previous best estimate calculation (Rohatgi, 1986) for a Westinghouse four-loop PWR. The end of refill occurred at 39 seconds in that calculatuion.

4.1 The Critical Number Of the Tube Ruptures.

The flow rate from the steam generator secondary side to the primary side is estimated using the subcooled critical flow data. This approach is used because the secondary side has subcooled liquid; possible overestimation of the break flow rate is expected to make these calculations more conservative. The flow rate from one tube is

Wtube=A*Gcrit,

where Gcrit is given as

The flow quality of the two phase mixture coming from the break is esimated from an isentropic expansion from the secondary side liquid conditions to the primary side conditions.

x=(Hliq,sec-Hf,prim)/Hfg,prim

The amount of liquid reaching the upper part of the vessel from the steam genarator secondary per tube is given as,

Wtube, liq=Wtube*(1-x)

The critical number of the tubes will depend upon the heat capacity of the structures in the upper part of the vessel that provide heat for evaporization of the incoming liquid. The heat structure conditions such as the teperature, surface area, and the heat transfer coefficient are taken from the previous best estimate calculation.

Q=Ahtst*H*(Thtst-Tsat, prim)

The number of the tubes than can be esimated from Q and Wtube, liq as

N=Q/(wtube, liq*Hfg, prim)

Here N is the number of tubes whose liquid portion of the break flow can be vaporized by the heat structures. For the larger number of the tube breaks the excess liquid will just flow down in the core region and assist in cooling the rods; for smaller number of the tube breaks the heat stored in the structures will vaporize all the liquid and, possibly, superheat vapor.

This apprach was applied to a Westinghouse four-loop plant, where the total area of the heat structure surface in the upper part of the vessel is about 350 m2. The heat structures and the saturation temperature in the primary side were assumed to be equal to 500 K and 410 K, respectively. The heat transfer coefficient for film boiling was 3000 J/M2-K-S. For the calculation of the break flow from the tubes, the assumed secondary side conditions where as following:

Tsec=530. K

Psec=57.1 Bar

Atube=0.0001862 M2

The critical number of tubes predicted following this approach was estimated to be 6, which is close to the semiscale result. However, it is coincidental as the semi-scale facility is very atypical. Furthermore, the critical number of tubes will strongly depend upon the location of the breaks since the steam generator secondary side has a large variation of the fluid conditions and the amount of liquid per tube coming into the primary side will be very different. For breaks at the top of the U-tubes the liquid flow through the break will be smaller and so the critical number of tubes will be larger. It was also assumed in this calculation that only the fluid from one side of the break will enter the upper part of the vessel and that the fluid will not acquire any

heat from the hot leg walls.

4.2 Reflood Delay.

It has been shown in the previous discussion that the steam generated in the upper part of the vessel tends to increase the pressure which impeades the liquid flow into the core. However, if there is enough liquid in the downcomer so that the hydrostatic head can overcome the pressure in the core/upper plenum region, reflood will begin. From the best estimate calculations, the equivalent loss coefficients for the total fluid path from the vessel to the loop breaks were estimated and used to estimate the break flow. The upper plenum pressure was assumed to stabilize when the steam generation was equal to the break flow. This is conservative as the reduction in the steam generation due to the structures cool-down and also decrease in the flow from the secondary side to the primary side due to the decrease in the pressure drop has not been accounted here.

Two different times were computed. Firstly, the time needed to fill the downcomer was computed assuming that only three cold leg injection flows were coming into the downcomer and that 50% of this injection was leaving through the break as was predicted in the best estimate calculation at 39 seconds. Dimensions of the vessel, downcomer, and upper part of the vessel were corresponding to the Westinghouse plant:

Volume of Vessel: 123.5 m3

Volume of the Downcomer: 16.82 m3

Volume of the Core: 41.63 m3

Volume of the Lower Plenum: 29.2 m3

Downcomer Height: 5.64 m

Loss Coefficient from Upper Plenum to Break: 16.0

Total ECC Flow per Loop: 168.8kg/s

The larger of the two calculated times was selected as the delay in reflood due to the exra steam generation. In the present calculation the larger time period was 84 seconds. This can be assumed to be the time by which the clad temperature turn-around will be delayed. Figure ... shows the clad temperature for the hot rod as predicted by TRAC-PD2 (Rohatgi,1986). The peak clad temperature occurred in the blowdown phase and the temperature turn around took place around 30 seconds due the flow oscillations at the core inlet. In the present estimate, this temperature turn-around in the refill phase will be neglected, and it will be assumed that the cladding continues to heat up at the rate predicted prior to the turn-around. Figure shows the clad temperatures for middle locations as extrapolated up to 84 seconds beyond the end of refill phase at 39 seconds. The peak clad temperature in the reflood phase as estimated here will be 825 K which is still smaller than the upper limit of 1478 K (2200 F) set in the Appendix K of 10CFR50.

Based on the simple and conservative analysis it can

be concluded that within the assumptions applied in the model described above, the peak clad temperature for a large break LOCA coupled with steam generator tubes rupture will most likely be below the 1478 K limit suggested in Appendix K.

5. Conclusions.

The bottom line of the evaluations of the semiscale experiments was answering the question whether the rate of the steam leaving the upper plenum and the phenomena controlling this flow rate are represented (scaled) adequately. These phenomena control the upper plenum pressure, and thus the steam binding due to the rupture flow injection.

The experimental curve showing PCT as a function of the equivalent number of tube ruptures (Fig. 9) results from a superposition of effects of different modeling features: scaling, test facility geometry and details in component representation, and a number of scenario and boundary condition choices. Presence of the modeling atypicalities pointed out above makes it difficult to confidently extrapolate the conclusions based on the semiscale results (at least in a quantitative sense) to the real PWR systems. Also, it appears not attainable to quantify a large number of competing effects leading either to increased or decreased measured core PCT during the experiments.

It has to be pointed out, however, that the data obtained in the Mod-1 S-28 series of experiments still can be used for the code assessment purposes as it has been intended

by the test designers. Unfortunately, the tube rupture experiments similar to the S-28 series have not been performed at the more accurately scaled Mod-3 test facility (see for example [8, 10-11].

An alternative way of addressing the SG tube rupture scenario would be to first assess a PWR safety code (TRAC, for example) using the experimantal data from selected series of the Semiscale program tests and, provided that the code is found adequate, to run the assessed code for the LOCA and SG-tube-rupture scenarios.

The simplified conservative estimates of the reflood delay have shown that within the assumptions applied, the peak clad temperature for a large break LOCA coupled with steam generator tubes rupture will most likely be below the 1478 K limit suggested in Appendix K.

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