TO: ACRS Members

FROM: Ivan Catton

DATE: 7 September 1996

SUBJECT: Conditional Probability of a Steam Generator Tube Rupture Following a Core Damage Accident

The Issue: Given that a core damage accident has occurred and that the system pressure is high, determine what the probability that a steam generator tube will rupture is. The NRR view is that it should be less than 0.1 to be consistent with requirements placed on the containment as it plays a similar role. This is a defense in depth type statement and comes from the use of a 10^{-6} bypass frequency for an acceptance criterion. If it can be shown that this is the case under the new rule, the rule is viewed to be safety neutral. If it does not, then the mitigating effects of the steam generator internals need to be considered, a much more difficult chore. Here one would attempt to take credit for the actual flow rates out of the tube break, the plate out of radio nuclide materials on the separators and the resistance to transport afforded by distance from the tube break to the outside world. This is my conjecture on how one should approach the problem.

What is the issue before the ACRS? In our letter on the Severe Accident Research Program, we said that the code of choice for this problem, RELAP5/SCDAP, could not do the appropriate calculations. Such a statement is overly strong as most codes like RELAP5 (the SCDAP part plays little role in the issue other than establishing the source temperature, it has its own uncertainties) can be tuned to yield adequate results. At the time of our letter writing, we had not been made aware of the tuning that had been carried out so that the code could be used. In what follows, I will first comment on the tuning that has been done, the highlights of a peer review held at Fauske and Associates and then give my view how one could generate the results that are needed by NRR to resolve the issue posed above.

The issue of steam generator tube rupture following a core degradation event first arose in 1983 as the result of calculations done using RELAP5 in the Nederlands. It took a long time for the nuclear safety community to come to grips with the problem. The first calculations done at INEL used an off-the-shelf version of RELAP5 much like the Dutch did. It became clear that DCH type accidents in a PWR would be low probability and that what would fail first (the hot leg, the surge line or the steam generator tubes) was uncertain. It is a race determined primarily by our knowledge of heat transfer and fluid mixing. This led to a very exhaustive joint experimental study by NRC and EPRI. The results were clear but qualitative; buoyancy driven recirculation will effectively heat up the upper parts of the primary system. Another set of experiments were carried out, without the participation of the NRC, that were more quantitative. These experiments were then used to supply the needed data for code tuning. The experimental results were supplemented by use of the COMMIX code. Given this, where are we at? The experimental was well scaled except for the fuel rod and steam generator tube sizes. Things like flow area and pressure drop were matched to a Westinghouse PWR. It is not clear that appropriate convergence testing was

done for the COMMIX to qualify the code results. This leaves us with the feeling that the RELAP5/SCDAP code has been tuned well enough to give one a reasonable result. In my view this is not enough to make a conditional probability statement and assurance of safety neutrality.

Our statements in the Severe Accident Research letter led RES to meet with myself and Tom Kress and then to convene a peer review group made up of Professor Ishii, Griffith and Viskanta. At our meeting with RES, I gave RES my view on what our concerns were. I told them that it was my view that RELAP5/SCDAP was a far too complex a code for the problem at hand and that although it could be used, uncertainties would still have to be dealt with. I chose to attend the meeting with the peer review group and RES. The meeting was opened by Charlie Tinkler with some instructions to the peer review group. The mind set was clear. No mention of the need for uncertainties was made. The peer review group was asked only to comment on whether or not the code could do the job. I then asked the peer review group to consider the need to consider uncertainties in heat transfer and mixing in a way that would allow one to make a statement about the probability of tube failure with some level of confidence. NRR made a presentation and, although not as clearly stated, indicated a need for such a statement. The resistance by RES to doing so was extremely strong. I don't know if this is because they don't understand that all experiments and code results are uncertain to some degree and that in the world of risk informed regulation this must be addressed or that they have made a policy decision not to do so. There was a great deal of discussion as to how difficult it would be to do it. I suggested that it would be much like the CSAU exercise and the response was that we can't spend two or three years on this. Such a response is difficult to understand because the present problem is much easier to treat than the one dealt with during the CSAU exercise.

Now, what do I think should be done? A simple study using the RELAP5/SCDAP code should be carried out to generate a response surface for the time to fail a steam generator tube. The parameters one needs to consider are few and their ranges are fairly well known. I will list them and give some rational for some ranges and estimate the number of computations one might need.

PARAMETER RANGES AND DISTRIBUTIONS

Vessel upper plenum temperature:

It is well accepted that the upper plenum will be well mixed. It will be a boundary condition on the problem and should be well determined by past SCDAP output. One only needs to estimate what probability to associate with the resulting spread. I will leave this to those more familiar with core degradation calculations.

Hot leg heat transfer:

Outside surface: The outside of the hotleg is insulated and there is no indication of how the insulation will perform under severe accident conditions. Without insulation, the bare pipe heat transfer coefficient will be around 25 W/m^2K. As an estimate, one might assume a chance of ½ for the interval 0-5.0W/m^2K, a chance of 1/4 for the interval 5.0-15W/m^2K and 1/4 for the interval 15-30W/m^2K.

Inside surface: This is the region of most uncertainty. There is no available data or

7

theories that deal with heat transfer from countercurrent single phase flow in a thick walled steel pipe. It is a conjugate problem with the complexity of countercurrent flow. One could start with the results of Jackson's work for mixed convection in a horizontal pipe. A range of 50 to 200% of Jackson's result could be probable. I would assign a 1/4 chance to the 50 to 100% range, a ½ chance to the 100 to 150% interval and a 1/4 chance to the 150 to 200% interval.

Surge line: The surge line is a problem. The heat transfer will be a function of whether or not the PORVS operate and how the flow separates at the nozzle as well as its location around the hot leg. At this moment, I do not have a recommendation for its value(s).

Steam generator tube heat transfer:

Inside surface: The heat transfer coefficient on the inside of the tube will most likely be unimportant as it is not the controlling value. The tube temperature will track pretty close to the steam temperature as soon as one is away from the lower tube sheet. A good value can be obtained from the work of Jackson. He has studied mixed convection in vertical tubes and written many papers during the past twenty or so years. Results such as these are typically good to within +/- 20%. With this in mind, I would assign a ½ chance to the interval +/20%, a 2/10 chance to the interval from 0 to 80% of the Jackson correlation and 3/10 to the interval from 120% to 200% of the Jackson correlation.

Outside surface: The outside of the steam generator tube is exposed to a large volume of steam. An estimate of the heat transfer coefficient under such conditions would be around 12 W/m^2K. A ½ chance could be applied to the range 7-13 W/m^2K, a 1/4 chance to the ranges 0-7 and 13-20 W/m^2K.

Steam generator hot side plenum mixing:

The distribution needed for the mixing parameter, f, should be based on the Westinghouse experiments where it was found to be 0.7. It is reasonable to assume that 0.7 +/- 0.1 is a possible range. The flow rate could be higher in a full plant and the mixing parameter should increase with flow rate. On the other hand, the steam generator tubes are not geometrically scaled and this could lead to a reduction in mixing (or maybe an increase). Their are some indication from the data that penetration of the hot layer under the tube sheet took place allowing hotter than average gasses to enter the tubes. As a result, one might assign a ½ chance to the 0.6-0.8 range, a 1/10 chance to the 0.0-0.6 range and a 3/10 chance to the 0.8 to 1.0 range.

RESPONSE SURFACE GENERATION

To generate a response surface, one only needs to be sure the parameter ranges are covered well

enough to create a surface. There are a number of ways to do this. Westinghouse did their LOCA BE work by assuming a linear relationship, then checking and developing a correction method leading to their final response surface. There are certain characteristics of the present problem that might help. One can, for example, assume that the steam generator tube behavior has little effect on the hot leg. What is done with the surge line separated flow heat transfer effects little else. An open PORV affects everything. A series of such rationalizations will lead to a set of calculations. My guess is that the number will be well under 200. On a workstation, the time per run is about 20 minutes. With careful compounding of runs, my guess is that the entire exercise will take on the order of 100 hours of SUN workstation time.

CONCLUDING REMARKS

The steam generator rule is a good place to start implementing risk informed regulation. To do so, one needs to incorporate what is known or not known in a meaningful and scrutable way. This means one needs to begin incorporating deterministic results into the PRA matrix in a meaningful way. A prescription might be the following:

1) definition of the issue
2) description of a model for the phenomena in question
3) specification (and justification of) of uncertainty distributions for parameters used in the models
4) establish a calculational matrix for generation of a response surface
5) random sampling of the model parameters according to their distributions
6) estimation of the probability distribution from the results

It seems to me that this is a simple and straightforward way to deal with many of the problems where PRA and deterministic results need to be married into a statement of risk.

9

IN RESPONSE, PLEASE
REFER TO: M960826

September 10, 1996

MEMORANDUM TO: E. Thomas Boulette, Chairman
Nuclear Safety Research Review Committee

FROM: John C. Hoyle, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - MEETING WITH CHAIRMAN OF NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE (NSRRC), 2:00 P.M., MONDAY, AUGUST 26, 1996, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by Dr. E. Thomas Boulette, Chairman of the Nuclear Safety Research Review Committee (NSRRC), and Dr. David Morrison, Director of the Office of Research, on the results of the NSRRC's June 27-28, 1996 meeting.

The Commission requested that the NSRRC coordinate its activities with those of the ACRS in areas of joint interest to ensure that the activities are supportive and complimentary and not duplicative. The NSRRC should also continue to review the progress of human factors research. The Committee should identify those human factor aspects that can be treated adequately in PRA, as well as those human factor areas where progress for inclusion in PRA is likely. The Committee should also provide recommendations for integrating these human factors considerations into PRA methods.

(NSRRC) (SECY Suspense: 3/7/97)

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
EDO
OGC
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR - Advance
DCS - P1-24

10

9609110107 1p.