Duke Power
A Duke Energy Company

D.M. JAMIL
Vice President

Duke Power
Catawba Nuclear Station
4800 Concord Rd. / CN01VP
York, SC 29745-9635

803 831 4251
803 831 3221 fax

August 18, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Response to Request for Additional Information

Reference: Catawba Proposed Amendment to the Facility Operating Licenses Concerning Steam Generator Tube Rupture Licensing Basis dated May 9, 2002

By letter dated May 9, 2002, Duke Energy Corporation submitted a license amendment request for the Catawba Steam Generator Tube Rupture Licensing Basis. During telecons on July 28 and August 5, 2004, the staff requested additional information associated with the submittal. The responses to the staff's questions are provided in the enclosed attachment.

The previous conclusions of the No Significant Hazards Consideration and Environmental Analysis as stated in the May 9, 2002 submittal are not affected by this response.

There are no NRC commitments contained in this letter or its attachment.

Pursuant to 10 CFR 50.91, a copy of this letter is being sent to the appropriate state official.

Inquiries on this matter should be directed to G.K. Strickland at (803) 831-3585.

Very truly yours,

D. M. Jamil

Attachment

A001

D. M. Jamil affirms that he the person who subscribed his name to the foregoing statement, and that all matters and facts set forth herein are true and correct to the best of his knowledge.

D. M. Jamil, Site Vice President

Subscribed and sworn to me: 8-18-2004
Date

Notary Public

My commission expires: 7-10-2012
Date

SEAL

xc (with attachment):

W. D. Travers
U.S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E. F. Guthrie
Catawba NRC Senior Resident
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

S. E. Peters (addressee only)
NRR Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O8, G9
Washington, D.C. 20555-0001

H. J. Porter
SC DHEC, Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

1) *How long will it take to depressurize to about one atmosphere pressure given the failure?*

Response: A combined response to Questions 1 and 2 is provided in the response to Question 2.

2) *What ensures this?*

Response: This is a combined response to Questions 1 and 2.

The analysis used the following times for cooldown and depressurization of the primary system:

t = 0 - Initial Steam Generator Tube Rupture
t = 20 minutes - reactor trip
t = 43 minutes - steam generator overfill
t = 20 minutes to t = 2 hours - plant stabilized at 550 degrees
t = 2 hours - plant cooldown started with an average cooldown rate of 50 degrees / hour
t = 9 hours - plant cooldown to 200 F and depressurization to atmosphere pressure completed

The following discussion provides additional details for the analysis inputs and assumptions, emergency procedure guidance for controlling the cooldown and depressurization, and results of the Unit 1 plant response from the simulator computer model.

A Steam Generator (S/G) Tube Rupture (SGTR) with overfill of the ruptured S/G may be followed by consequential failure of a relief valve for the ruptured (S/G), either its power operated relief valve (PORV) or one of its main steam code safety valves (MSSVs). Break flow now can be stopped only by cooling the affected nuclear unit to the ambient boiling point and lowering the pressure in the Reactor Coolant System (RCS) to ambient pressure. The extended break flow and the shrinkage of the reactor coolant with cooling place additional demands for make-up to the RCS.

Make-up to the RCS for inventory lost through the break in a SGTR would be provided by the Emergency Core Cooling System (ECCS). The water source for the ECCS for the SGTR is the refueling water storage tank (RWST) as there is no recirculation for the SGTR. The important characteristics of the SGTR with S/G overfill with regard to the adequacy of the RWST are the integrated break flow and the shrinkage of the reactor coolant. The inventory in the RWST should be enough to make up for the reactor coolant lost through the break and to ensure that the RCS is filled given the shrinkage of the reactor coolant as it is cooled.

Upon discovery of the effect of the EDE/EDF failure on the SGTR, administrative controls were put into place at Catawba in conformance to procedures within Duke for degraded conditions at one of its nuclear power plants. The administrative controls are discussed in the response to Question 4. These administrative controls were based on an assessment of radiation doses of a design basis SGTR with EDE/EDF failure. The failure was postulated to lead to overfill of the ruptured S/G and consequential failure of one of its MSSVs. Break flow was assumed to continue until the operators were assumed to cool the affected unit to 200 °F and lower the reactor coolant pressure to ambient pressure. Assumptions were made concerning the cooldown time and the break flow rates for this transient.

The assumed time line for the transient was as follows: Reactor trip was assumed to occur at 20 minutes after the initiating event. S/G overfill was projected to occur approximately 23 minutes after trip (43 minutes after the initiating event). Consequential failure of one of the MSSVs for the ruptured S/G was postulated to occur at this time. It was assumed that the operators would make no attempt to cool the affected unit until 2 hours after the initiating event (1 hour 40 minutes after unit trip). In addition, no credit was taken over this time span for cooldown of the reactor coolant associated with the consequential failure of the MSSV (small steam line break). Then the operators were assumed to cool the affected unit at a rate of 50 °F/hr. The average reactor coolant temperature was set to 550 °F at the initiation of the cooldown at 2 hours after the initiating event. Break flow was assumed to be terminated when the RCS temperature was projected to reach 200 °F seven hours into the cooldown and 9 hours after the initiating event.

These assumptions are considered to be reasonable bounds for such a transient at Catawba based on evaluations of a "desk-top" simulation of the SGTR scenario.

A desk-top simulation of the SGTR with EDE/EDF failure and overfill of the ruptured S/G was completed. In preparing the simulation, the following conservative assumptions were made:

2.01) Both trains of the ECCS were assumed to be in operation. This is conservative given that failure of Distribution Center EDE or EDF leads at least initially to the Minimum Safeguards Scenario. This assumption brings SGTR break flow rate and ECCS flow rate for this accident to their maximums and therefore is limiting for an evaluation of the adequacy of the inventory in the RWST.

2.02) The motor driven Auxiliary Feedwater System (AFWS) pump on the same Class 1E train as the failed EDE/EDF was simulated to function. This yields a more rapid S/G overfill, and higher initial ECCS flow rates. Again, this assumption is conservative in that the assumed EDE/EDF failure leads at least initially to a Minimum Safeguards scenario.

2.03) The design basis SGTR scenario includes loss of offsite power at trip. This may cause loss of flow of instrument air. The simulation of the SGTR with S/G overfill did not model the restoration of instrument air flow. Refer to the Response to Question 5, Bullet 5.05 for a discussion of restoration of instrument air.

2.04) The simulation did not include restoration of offsite power.

Like the main simulator, the desktop simulator models nominal expected plant system performance parameters. The initial conditions (S/G level, reactor power, pressure, etc.) and boundary conditions (AFWS and ECCS flow rates, etc.) as modeled in the simulator for the evaluation of the SGTR with S/G overfill take nominal values. Also, variables such as operator response times may result in some variations in results. The simulator was developed as a tool for training reactor operators. It was not developed to simulate limiting design basis accidents as are computer codes like RETRAN. Therefore, the results of the desk-top simulator are provided as a separate and detailed evaluation for comparison with the assumptions of the dose calculation and available analyses of SGTR with S/G overfill. The results of the simulator exercise should not be taken to be limiting for this scenario.

The scenario presented is a 440 GPM SGTR in "B" S/G of Unit 1. (The main and desk-top simulators replicate one Catawba unit: Unit 1. Based on experience with analyses of DB SGTRs at Units 1 and 2, it is judged that posing a SGTR on Unit 2 would not significantly affect the time line presented below. A quantitative estimate of the effect of a simulated SGTR on demand for RWST inventory is presented below.) No effort was made to reduce AFWS flow to this S/G until it was simulated to fill. At this point, consequential failure of one MSSV was simulated. In responding to this scenario after reactor trip, the control room operators follow the directions of the emergency operating procedures:

1) E-0 (Reactor Trip or Safety Injection),
2) E-3 (S/G Tube rupture - Ref. 29),
3) E-2 (Isolation of Faulted S/G - Ref. 30),
4) ECA-3.1 (SGTR With Loss of Reactor Coolant, Subcooled Recovery Desired - Ref. 2) and
5) ECA-3.2 (SGTR with Loss of Reactor Coolant, Saturated Recovery Desired - Ref. 3).

The following is a synopsis of the simulation.

E-0 (Reactor Trip or Safety Injection - Ref. 31) is the entry/ diagnostic procedure. E-3 is the first event-specific procedure. In this postulated scenario, E-3 is in effect before the fault occurs. At transition from E-0 to E-3, the total ECCS flow rate is ~375 gpm, all of which comes from the high pressure injection pumps. Based on desktop simulation, S/G overfill and resulting MSSV failure will result at some time after cooldown of the RCS is initiated per E-3 to establish a subcooled margin in the ruptured S/G. (E-3 directs the operators to take this action to provide a subcooled margin in the reactor coolant of at least 20 ^{o}F at the pressure in the ruptured S/G.) The ruptured S/G likely may overfill after this action is complete. The normal E-3 cooldown combined with the inadvertent cooldown created by the failed MSSV produce a significant RCS temperature drop, with attendant decreases in the pressures of both the RCS and ruptured/faulted S/G. (The pressure differentials across the S/G tubes drops somewhat during this period, but only by ~200 psi). The more important effect is the cooldown brings the plant much closer to the final desired temperature. Of course, there are several cooldown contributors just after reactor trip and Safety Injection (SI) initiation. By the time the operators are ready for intentional cooldown of the RCS per E-3, the hot leg temperatures have decreased to about 540 °F (due primarily to reactor trip, automatic S/G PORV actuation and ECCS flow). The ECCS flow rate has decreased slightly to ~360 gpm. The pressurizer level is just barely on scale at ~8% "Cold Cal" level. ("Hot Cal" is offscale low.) RWST level is ~94%. This data point is about 20 minutes after ECCS actuation.

The operators transition briefly from E-3 to E-2 in an attempt to isolate the faulted S/G (ruptured S/G with consequential failure of a MSSV). With this failure, neither the feed source (tube rupture) nor the fault (failed MSSV) can be addressed. The control room operators eventually transition back to E-3 with no significant effect on the scenario.

The operators then transition from E-3 to ECA-3.1 based on continued depressurization of the ruptured S/G. At the exit from E-3, the additional cooldown due to intentional cooldown and rupture/fault cooldown is about 100 °F, with a T-hot of about 452 °F. At the transition from E-3 to ECA-3.1, RWST outflow rate (ECCS flow rate) has increased to almost 700 gpm. This increase is caused by the cooldown, which causes the reactor coolant to "shrink" (increase in density and therefore take less space). This, in turn, forces primary system depressurization which produces the increased output of the ECCS pumps. Flow is now coming from the intermediate head Safety Injection pumps (~300 gpm) as well as the High Pressure Injection pumps. The pressurizer level is about the same, at about 7% Cold Cal. RWST level is ~91%. This data point is taken at about 35 minutes after ECCS actuation.

Initially after the consequential failure of the MSSV and in the early stages of ECA-3.1, the cooldown effect is being created entirely by the rupture/fault. Cooldown rates were about 60 °F/hr. Operator response per ECA-3.1 contributes significantly to lowering the break flow by depressurizing the RCS in an attempt to recover pressurizer level. In this scenario, depressurization is accomplished by using one pressurizer PORV. The PORV motive force is normally instrument air, which is not available in this scenario. A reserve supply of nitrogen from a separate Cold Leg Accumulator is aligned to two of the three pressurizer PORVs. Electric power to the controls of these two pressurizer PORVs is delivered from a separate Class 1E power train. Depressurization was not a problem.

The operator response as directed by ECA-3.1 reduces outflow from the RWST by sequential reduction in ECCS flow rates based on subcooling and pressurizer level. The result was a reduction in RWST outflow rate to about 360 GPM at the end of the first pass through ECA-3.1. RWST level was ~87%. This data point is taken about 55 minutes after ECCS actuation.

As simulated, the operators remain in ECA-3.1 for quite some time. The procedure calls for use of the pressurizer PORVs, if required, to keep pressurizer level above 25%. The RCS pressure remained relatively constant at about 1100 to 1200 PSI, and pressurizer level increased slowly to ~80% due to constant charging rate and decreasing break flow. Shrinkage due to cooldown was not enough to absorb all of the excess charging. Initially following the consequential fault, the cooldown is provided by the rupture/fault, and is later maintained by using S/G PORVs on intact S/Gs. (Two of which have to be operated locally, due to the postulated failure). ECA-3.1 was in effect for about an hour. During this hour, the hot leg temperatures dropped from ~452 °F to ~400 °F. More cooldown was available, but was not used due to the administrative limit of 50 °F /hr.

There are two exits from ECA-3.1 to ECA-3.2. One is based on RWST inventory. For this scenario, the operators would transition from ECA-3.1 to ECA-3.2 if the level in the RWST decreased below 70% with no water level in the sump. In this simulation, the RWST was still ample, and did not trigger the transition. The other transition point is level in the ruptured/faulted S/G greater than 92%. During the initial stage of the S/G blowdown and "B" S/G level dropped to about 50%. The level in "B" S/G eventually increased to the transition setpoint, and a transition was made to ECA-3.2. At transition to ECA-3.2, ECCS flow rate is ~ 345 gpm. RWST level is ~82% Cooldown is being provided by continued fault flow, one intact S/G PORV full open (from control room) and one additional intact S/G ~ 20% open (operated locally). This data point is taken at about 1 hour 50 minutes after ECCS actuation.

ECA-3.2 provided several benefits to the evolution. The first was a substantial reduction in RWST outflow. The lower subcooling requirements in ECA-3.2 allow normal charging to be established, and the S/I injection valves to be closed. Using normal charging reduces total RWST outflow to about 215 gpm some 8 minutes after entering ECA-3.2 (roughly 2 hours after ECCS actuation). ECA-3.2 also depressurizes the RCS to saturation, which resulted in primary system pressure of about 300 psig, greatly reduced primary/secondary break flow. The third major benefit of ECA-3.2 is that it overrides the 50 ^{o}F/hr administrative limit and allows cooldown rates up to 100 ^{o}F/hr. The cooldown rate was then increased to about 90 ^{o}F/hr. Later, the cooldown rate decreased with RCS temperature. Just after depressurization RWST total outflow is about 200 gpm and RWST level is ~80%. The hot leg temperatures are about 390 ^{o}F. This data point is about 2 hours 15 minutes after ECCS actuation.

Cycling through ECA-3.2 continues until the conditions for aligning the Residual Heat Removal System (RHRS) are met. The cooldown rate decayed as cooldown continued. The last degree of cooldown took ~4 minutes (15 ^{o}F/hr). This cooldown rate could have been temporarily increased by increasing S/G level, but it did not seem necessary. The RWST level is ~78% and RWST total outflow rate is about 85 gpm (35 gpm charging and 45 gpm mini-flow for the high head charging pumps.) The hot leg temperatures are all below 350 ^{o}F. This data point is taken about 3 hours 15 minutes after ECCS actuation.

The cooldown via one train of the RHRS took another 3 hours and 15 minutes. In the simulation of this phase, it was assumed that the control room operators began operation of the RHRS train with low flow rates, then increased flow to achieve the desired cooldown rate. The RWST outflow during this time was mostly minimal, with occasional adjustments to maintain pressurizer level reasonably constant. The pressurizer PORVs were used to maintain RCS pressure at saturation during the cooldown, which probably decreased primary to secondary break flow, but the break flow was too small to be really evident by this point. When the last hot leg temperature was below 200 ^{o}F, the simulation was stopped. The RWST level was ~70% and RWST total outflow rate was 95 GPM, 45 gpm of which was miniflow. This data point is taken at about 6 hours 30 minutes after ECCS actuation. During the simulated scenario, the rupture disk on the Pressurizer Relief Tank (PRT) remained intact. The final pressure and level in the PRT were, respectively, 73 psig and 80%.

In the simulation of the operation of the RHRS, the temperature of the ultimate heat sink was set to 68.5 °F. The temperature of the water in the ultimate heat sink may be as high as 91.5 °F (Ref. 32 TS 3.7.9). The results of the cooldown simulation from 350 F with the temperature of the ultimate heat sink set to 91.1 °F is provided below: (It is judged that setting the ultimate heat sink temperature to 91.5 °F would not have significantly affected the results at least with respect to inventory in the RWST.)

Core Exit Temperature	Time to Cooldown	RCS Pressure	RWST Level
350 °F to 250 °F	~1 hour 53 minutes from 350 °F to 250 °F	~181 psig at 350°F to ~21 psig at 250 °F	decrease from ~78% to ~71%
250 °F to 230 °F	~2 hours 1 minute from 250 °F to 230 °F	~21 psig at 250 °F to ~5.6 psig at 230 °F	decrease from ~71% to ~70%
230 °F to 200 °F	~7 hours 9 minutes from 230 °F to 200 °F	~5.6 psig to 0 psig	no significant change from ~70%, assume ~69% as a lower bound

The total time for cooldown from 550 °F to 200 °F with the ultimate heat sink temperature set to 91.1 °F was simulated to take approximately 14 hours 24 minutes. From this repeated simulator exercise, it is concluded that increasing the temperature of the ultimate heat sink could significantly extend the cooldown time - especially from 250 °F to 200 °F. However, the additional break flow, break flow rate, and RWST inventory loss was projected to be very low.

These evaluations conducted with the desk-top simulator provide "data points" to demonstrate the feasibility of cooldown of the RCS and bringing its pressure to ambient within the time span assumed in the dose calculation for the DB SGTR with S/G overfill.

As noted above, the starting point of the cooldown assumed in the dose calculation was an RCS average temperature of 550 °F. From a review of transient thermal hydraulic calculations in place at the time, this assumption was seen to be acceptable.

Assumed break flow rates also are important. Combined with the cooldown time line, they yield integrated break flow, one of the two factors in assessing the adequacy of the inventory in the RWST for this scenario, the other factor being shrinkage of the reactor coolant. The dose calculation for the design basis SGTR with S/G overfill incorporated assumptions concerning the SGTR break profile. The assumed break flow profile is as follows: 61 lbm/sec for 0-20 min, 50 lbm/sec for 20-120 min, and 40 lbm/sec for 120-540 min. This yields an integrated break flow of ~1,381,000 lbm. Westinghouse conducted a generic analysis of SGTR with S/G overfill in which they evaluated two cases (Ref. 1). The integrated break flows for the two cases evaluated by Westinghouse were 1,039,000 lbm and 1,159,000 lbm. The integrated break flow following from the break flow and cooldown profiles assumed in the dose calculation are significant upper bounds to the integrated break flow for the SGTR with S/G overfill evaluated by Westinghouse.

The upper bound for shrinkage of the reactor coolant, taken from hot full power to standard conditions, has been calculated to be 228,000 lbm. The break flow loss and shrinkage make-up add to 1,610,000 lbm. This equates to a final RWST level of 44.5%. The projected RWST level at the end of the simulation and the temperature of the ultimate heat sink set to 68.5 °F was 70%. As noted above, the simulation of the RHRS phase of the cooldown was repeated with the ultimate heat sink temperature set to 91.1 °F. This extended the total cooldown from 6 hr 30 min to 14 hr 24 min. However, there was no significant increase in the demand for water in the RWST. At the end of the repeated simulation, the RWST level was projected to essentially remain at 70%. It should be noted that the greatest effect in increasing the ultimate heat sink temperature was to increase the time span to cool the affected unit from 250 °F to 200 °F. Over this time span, the SGTR break flow rate was very small.

From the initial simulator exercise, the amount of water projected to be drawn from the RWST was 870,000 lbm. Although the repeated simulator exercise showed essentially no change in the RWST level with the increase in ultimate heat sink temperature, it is assumed that the RWST level associated with this exercise is 69%. This equates to a demand for 899,000 lbm of water from the RWST.

The above figures correspond to a SGTR at Unit 1. A SGTR occurring at Unit 2 would have an initial break flow rate of 550 gpm. As noted above, the time line would not be affected significantly by posing a SGTR on Unit 2. Given that, it is estimated that the total demand on the RWST for a SGTR simulated on Unit 2 and a RWST temperature of 91.5 °F would be 1,124,000 lbm, equating to a RWST level of 61%. This demonstrates the conservatism of the combined assumptions of break flow rate and cooldown times for the in-house dose calculation compared to the desk-top simulation.

The minimum inventory of water allowed in the RWST (350,000 gallons at 100 °F) is 2,901,000 lbm. Therefore, given the assumptions in the dose calculation, the design basis SGTR with EDE/EDF failure and S/G overfill would leave a minimum of 1,291,000 lbm of water. This equates to 156,000 gal at 100 °F and (as noted above) an RWST level of ~45%. This and the desk-top simulation provide the assurance that the inventory in the RWST is more than adequate to provide make-up and cooling water to the RCS of the affected unit, ensuring compliance with 10 CFR 50.46.

3) *What are the RWST makeup capabilities? Can the RWST of the other unit be used in this capacity?*

Response: The dose calculation included no assumption pertaining to make-up to the RWST. The RWST contains sufficient inventory to provide makeup to the primary system for this event. Additional borated water could be transferred to the RWST should this become necessary in the long-term response to an accident. Water could be transferred from the Reactor Makeup Water Storage Tank (RMWST) to the RWST at the rate of 120 gpm. The RMWST has a capacity of 112,000 gallons. In addition, water could be transferred from the Demineralized Water Storage Tank to the RMWST for transfer to the RWST. Boric acid also could be added from the Boric Acid Storage Tank with the use of the Boric Acid Transfer Pumps.

Overfill of the ruptured S/G is assumed to be followed by consequential failure of one of its relief valves, either an MSSV or a S/G PORV. With this failure, the pressure in the ruptured S/G would decrease. Eventually, the pressure in the ruptured S/G will be below the pressure in the intact S/Gs by the time the operators complete the steps in the SGTR emergency procedure E-3 for cooldown of the RCS to establish a subcooled margin in the RCS relative to the pressure in the ruptured S/G. This is a point for transition from E-3 to ECA-3.1, the emergency procedure for SGTR with loss of reactor coolant and subcooled recovery desired (Ref. 2). Once the RWST level fell below 70% or level in the ruptured S/G exceeds 92%, the operators would enter the emergency procedure for SGTR with loss of reactor coolant and saturated recovery desired (Ref. 3). In following the first two steps of this procedure, the operators monitor level in the RWST and begin makeup to it (Ref. 4). This is a step which is not credited in the estimation of radiation dose for a SGTR with S/G overfill.

The reactor make-up water pumps are blackout but non Class 1E loads. Loading them on the diesel generators (D/Gs) requires that Safety Injection (SI) be reset. However, the operators would have reset SI (Ref. 2 Step 4) before they begin makeup to the RWST (Ref. 3 Step 15). Failure of power from EDE / EDF will not affect the ability of the operators to reset SI on the opposite class 1E train. The SGTR is a SI event. The design basis SGTR also includes loss of offsite power (a.k.a. blackout) at reactor trip. For combined SI-blackout events, the operators normally would not load the blackout switchgear onto the associated 4160 volt Class 1E switchgear. The concern is loading the switchgear beyond its design capacity when it is supplying power to SI related loads (even with SI reset). Once offsite power is restored (cf. response to Question 5), the operators would begin make-up to the RWST with the RMW pumps. The operators also would initiate make-up to the RWST once it becomes evident that without this action, the RWST level eventually will fall to the threshold for stopping the ECCS pumps aligned to it.

The operators must open two Class 1E isolation valves (in series) in order to allow the flow of make-up water to the RWST. These valves are normally closed and also are closed on the SI signal. Power to one of the valves would be lost with loss of bus EDE/EDF. The operators could either open the valve with its handwheel or restore 600 volt power to the valves by closing the feeder breakers to the affected Class 1E 600 volt load centers (4160/600 volt transformers).

There is a connection between the RWSTs of the two units. If it were open, the RWST of the unaffected unit could be used as a source of water for the RWST of the affected unit. Flow would be driven by the hydrostatic head in the RWST of the unaffected unit. The operators would have to open three handwheel isolation valves (1FW22, 1NB291, and 2FW22).

For a scenario in which the required flow rate to the RCS is sufficiently low (i.e., 26 GPM or less), the operators have an additional option. They could use the Standby Makeup Water pump to transfer water from the spent fuel pool directly to the RCS. The operators could draw up to 38,000 gallons from the spent fuel pool and maintain its level above the limits of TS 3.7.14.

4) *What are the assumptions regarding the dose calculation in the April 14, 2004 e-mail? Are they realistic or are they consistent with the license basis of Catawba Nuclear Station?*

Response: The e-mail of April 14, 2004, is based on a calculation of radiation doses following a postulated design basis SGTR with failure of EDE or EDF and consequential failure of an S/G relief valve. Neither any assumption made in the calculation nor any input taken for it was "best estimate." The dose calculation was based on several assumptions that while not best estimate are not in the current license basis of Catawba. Specifically, these assumptions were based on what were draft regulatory positions but now are cited in R.G. 1.183 and R.G. 1.195. These assumptions are listed below. Note: The information in the e-mail of April 14, 2004 is contained in the response to this question. The information on the administrative controls in place at Catawba is presented in Bullets 4.06, 4.08, and 4.09. The results of the dose calculation are discussed in the last two paragraphs of this response.

4.01) In separate scenarios, a pre-accident iodine spike and an accident initiated or concurrent iodine spike were assumed. The concurrent iodine spike is based on an assumed increase in the rate of appearance of iodine in the reactor coolant to some multiple of the equilibrium appearance rate. For this scenario, the multiplier was set to 335 (Ref. 5-8). In the calculations of radiation doses following the design basis SGTR in the current license basis of Catawba, this multiplier is set to 500. However, the Staff has approved setting the multiplier for the concurrent iodine spike to 335 for the design basis SGTR (Ref. 6, 8). In addition, an informal analysis completed by Duke has confirmed the value of 335 as a 2-sigma value for the multiplier for the concurrent iodine spike for accidents such as the SGTR.

4.02) Dose coefficients were taken from Federal Guidance Reports 11 and 12 (Ref. 5, 9, 10). These dose coefficients have been used in the analyses of radiological consequences of the design basis Fuel Handling Accidents (FHAs), Weir Gate Drop, and Loss of Coolant Accident (LOCA) completed with the method of Alternative Source Terms (AST, cf. Ref. 11 & 12). The Staff has approved partial scope implementation of AST at Catawba based on the calculation of radiation doses for the design basis FHA and Weir Gate Drop (Ref. 13). The NRC Staff currently is reviewing the application for full scope implementation of AST at Catawba based on the analysis of radiological consequences of the design basis LOCA.

4.03) The iodine source term in the reactor coolant was computed from the administrative limits on Dose Equivalent Iodine-131 (DEI) based on coefficients for thyroid Committed Dose Equivalents (CDEs) taken from Federal Guidance Report 11.

4.04) New values were taken for atmospheric dispersion factors (χ/Qs) for transport of radioactivity to the outside air intakes of the Control Room Area Ventilation System (CRAVS). Baseline values of these control room χ/Qs were calculated with the computer code ARCON96 (Ref. 14) based on transport of radioactivity with dispersion from one release point to one CRAVS outside air intake. The release points for the design basis SGTR are the unit vent stack before reactor trip and the S/G relief valve vents (and the exhaust vent for the steam driven AFWS pump after unit trip and postulated loss of offsite power). Values for composite control room χ/Qs were calculated based on an assumed imbalance of 60/40 of airflow into the CRAVS outside air intakes. This information and additional details have been presented to the Staff for review as part of the application for full scope implementation of AST at Catawba (Ref. 11, 15). We note here that the values of the control room χ/Q for release from the S/G relief valve vents are higher than the values in the current Catawba licensing basis.

Assumptions concerning the cooldown time and SGTR break flow following postulated failure of an S/G relief valve have been noted in the response to Question 2. Additional methodology and assumptions were employed as follows:

4.05) The Bechtel proprietary computer code LOCADOSE (Ref. 16-18) was used to complete the calculation of radiation doses for the design basis SGTR with the EDE/EDF failure. The activity transport model in this code conforms to the germane regulatory positions that the Staff has published (Ref. 7). The code calculates activity in the control room based on the time dependent Murphy-Campe Equation in place of an equilibrium iodine protection factor model.

4.06) Two scenarios were postulated for the design basis SGTR with concurrent iodine spike as follows: For the first scenario, one letdown (L/D) line with a flow rate of 80 gpm was assumed to be in service and the equilibrium DEI specific activity in the reactor coolant was set to 0.099 μCi/gm. In the second scenario, the reactor coolant DEI specific activity was set to 0.064 μCi/gm and two L/D lines were taken to be on-line for a flow rate of 125 gpm. These are part of the administrative controls in place at Catawba pending resolution of the issue in this submittal.

4.07) The following assumptions were made for the calculation of the equilibrium iodine appearance rates for the design basis SGTR with pre-accident iodine spike: The limiting reactor coolant leakage allowed by the plant technical specification (Ref. 19) was assumed (for a total of 11 gpm). The mass leak rate was computed based on standard conditions. The L/D flow was assumed to be at standard conditions (L/D flow is measured downstream of the L/D Heat Exchanger at 100 ^{o}F).

4.08) For the design basis SGTR with pre-accident iodine spike, the initial activity in the reactor coolant was set to 15 μCi/gm. These are part of the above-mentioned administrative controls at Catawba.

4.09) Initial level of radioactivity in the secondary side of the S/Gs was set to 0.055 μCi/gm. This is the last part of the administrative controls in place at Catawba. Initial activity levels in the condenser hotwell and the condensate grade sources for the AFWS were computed based on an S/G iodine partition factor of 100 and perfect scrubbing in the main condenser (the latter assumption being conservative for calculating initial activity levels in the unit secondary systems).

4.10) Main feedwater (MFW) flow rates before unit trip are set to "turbine valves wide open."

4.11) Releases of iodine from the secondary coolant before unit trip were calculated based on an efficiency of 85% for removal of iodine in the main condensers.

4.12) Unit trip is assumed to occur 20 minutes after the initiating event. This is consistent with the methodology for calculation of radiation doses for the design basis SGTR (Re. 20, 21).

4.13) With the exception of unit trip, all assumed initial and boundary conditions are consistent with the limiting design basis SGTR with respect to margin to overfill of the ruptured S/G (Ref. 22, 23). For example, loss of offsite power at trip is assumed. Maximum initial water levels in the S/Gs were assumed. Both trains of ECCS were assumed to be initially in operation (the EDE/EDF failure leads to a Minimum Safeguards scenario).

4.14) The EDF/EDF failure degrades the ability of the operators to stop flow from the turbine driven AFWS pump to the ruptured S/G. It was assumed that the operators did not isolate flow from the turbine driven AFWS pump to the ruptured S/G before it was projected to fill up. The rate of flow from the turbine driven AFWS pump to the ruptured S/G was set at maximum values at the ruptured S/G steam pressure.

4.15) Consequential failure of the MSSV for the ruptured S/G was taken as soon as the ruptured S/G was projected to fill up. No credit was taken for the time required to fill the main steam line segment to the Main Steam Isolation Valves.

4.16) Westinghouse has developed a best-estimate model for iodine transport and release following a postulated SGTR with S/G overfill (Ref. 1). This model was not used. The following conservative and deterministic (non mechanistic) assumptions were made in its place: Following projected overfill of the ruptured S/G, the SGTR break flow was assumed to flash directly to environment. No credit was taken for scrubbing of flashed SGTR break flow. In addition, steaming was taken with an iodine partition factor of 100.

4.17) The offsite χ/Qs (at the Exclusion Area Boundary-EAB and boundary of the Low Population Zone-denoted as the LPX) were set to limiting values (Ref. 11, 12).

4.18) Limiting values were assumed for the performance of the CRAVS. The rate of unfiltered inleakage was set to its limiting value of 100 cfm. (Ref. 12)

4.19) The resultant doses were compared to the NRC expectations of Standard Review Plan (Ref. 28) Sections 15.6.3 and 6.4.II. These expectations for offsite and control room radiation doses are part of the current license basis of Catawba.

The limiting offsite radiation dose was the thyroid radiation dose at the Exclusion Area Boundary (EAB) following the design basis SGTR with concurrent iodine spike and S/G overfill. The EAB thyroid radiation dose for this scenario was found to be 29.3 Rem. The germane NRC guideline value is 30 Rem. This design basis SGTR scenario corresponds to the administrative limits for equilibrium RCS DEI specific activity of 0.099 μCi/gm with one letdown line in operation and 0.064 μCi/gm with two letdown lines in operation.

For a design basis SGTR with pre-accident iodine spike and S/G overfill, the limiting offsite radiation dose was the EAB thyroid radiation dose with 83.9 Rem. The limiting NRC guideline value is 300 Rem. This scenario is limiting for thyroid radiation dose in the control room. This was computed to be at the guideline value of 30 Rem. The associated administrative control is transient RCS DEI specific activity limited to 15 μCi/gm.

5) *Would operator action outside the control room contribute to a response that is more effective than that associated with the dose calculation?*

Response: The dose calculation implicitly credits the following operator action outside the control room.

5.01) The EDE/EDF failure will cause loss of power to the Class 1E solenoids for two S/G PORVs, causing them to open to vent air to ambient to keep the two S/G PORVs closed (the "fail safe" configuration). The worst case scenario is that the affected PORVs are associated with two of the three intact S/Gs following a design basis SGTR with EDE/EDF failure. The dose calculation implicitly assumes that personnel operate these failed closed S/G PORVs with their handwheels. The ability of the operators to complete this action has been validated (Ref. 28).

Additional operator actions outside the control room, not credited in the dose calculation, may also be taken. They include but are not necessarily limited to the following:

5.02) The EDE/EDF failure causes loss of power to the 600 volt motor for the Class 1E isolation valve and the Class 1E solenoids in the in-series control valve from the AFWS turbine driven pump to the ruptured S/Gs. In this scenario, the operators would close the affected Class 1E isolation valve with its handwheel. This action is not specifically credited in the dose calculation (cf. ¶ 4.14 above). Early completion of this local action could delay the time to overfill of the ruptured S/G. These valves are located in the S/G doghouses. These buildings are safety-related structures and are classified as mild environments.

5.03) The EDE/EDF failure causes loss of power to the load sequencer for the associated Class 1E D/G and loss of control power to the Class 1E 4160 volt switchgear. This yields the Minimum Safeguards scenario. The dose calculation implicitly assumes that the operators will manually close the 4160 volt breaker for the affected train of the RHRS. The pathway from the control room or from a staging area near the control room to the room in which the affected switchgear is located inclusive is completely inside safety-related Seismic Category I structures. These rooms also are classified as mild environments. This action need not be taken for at least 4-5 hours after the initiating event. For these reasons, this action appears feasible.

5.04) The design basis SGTR includes loss of offsite power at unit trip. The operators are directed to restore offsite power "when time permits" (Ref. 24-27). Restoration of offsite power would allow use of the reactor coolant pumps. In establishing forced circulation in the RCS, the operators would facilitate the cooldown of the RCS and shorten the time required to bring the RCS to ambient pressure.

5.05) Loss of offsite power would be followed by loss of instrument air. The operators are directed to restore instrument air (Ref. 27). The operators could restore instrument air in at least three different ways. The Instrument Air compressors are blackout non-1E loads. The operators could restore SI (as they are directed to do) and load the blackout buses onto the D/G's. Second, the operators could connect a diesel powered air compressor to the Instrument Air lines and start it. Finally, the operators could restore offsite power (as noted above).

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