REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8206210539 DOC.DATE: 82/06/17 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387
50-388 Susquehanna Steam Electric Station, Unit 2, Pennsylva 05000388
AUTH.NAME AUTHOR AFFILIATION
CURTIS,N.W. Pennsylvania Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards responses to NRC questions re sample representativeness & interpretation for core damage.Action completed.Response allows for closeout of TMI Item II.B.3 & SER Item 81.

DISTRIBUTION CODE: A046S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 4
TITLE: Response to NUREG -0737/NUREG-0660 TMI Action Plan Rqmts (OL's)

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	LIC BR #2 BC 01	7	7			
INTERNAL:	ELD/HDS4	1	0	IE/DEP DIR 33	1	1
	IE/DEP EPDS	1	1	IE/DEP/EPLB	3	3
	NRR/DE DIR 21	1	1	NRR/DE/ADCSE 22	1	1
	NRR/DE/ADMQE 23	1	1	NRR/DHFS DIR 28	1	1
	NRR/DHFS/DEPY29	1	1	NRR/DL DIR 14	1	1
	NRR/DL/ADL 16	1	1	NRR/DL/ADOR 15	1	1
	NRR/DL/ADSA 17	1	1	NRR/DL/ORAB 18	3	3
	NRR/DSI DIR 24	1	1	NRR/DSI/ADDPS25	1	1
	NRR/DSI/ADRP 26	1	1	NRR/DSI/ADRS 27	1	1
	NRR/DSI/AEB	1	1	NRR/DSI/ETSB	1	1
	NRR/DSI/RAB	1	1	NRR/DST DIR 30	1	1
	NRR/DST/ADGP 31	1	1	NRR/DST/ADT 32	1	1
	REG FILE 04	1	1	RGN1	1	1
EXTERNAL:	ACRS 34	10	10	FEMA-REP DIV	1	1
	INPO,J.STARNES	1	1	LPDR 03	2	2
	NRC PDR 02	1	1	NSIC 05	1	1
	NTIS	1	1			

TOTAL NUMBER OF COPIES REQUIRED: LTTR 54 ENCL 53

PP&L Pennsylvania Power & Light Company

Two North Ninth Street • Allentown, PA 18101 • 215 / 770-5151

Norman W. Curtis
Vice President-Engineering & Construction-Nuclear
215 / 770-5381

JUN 17 1982

Mr. A. Schwencer, Chief
Licensing Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO QUESTIONS ON POST
ACCIDENT SAMPLING
ER 100450 FILE 841-2, -12
PLA-1133

Docket Nos. 50-387
50-388

Dear Mr. Schwencer:

The attachments to this letter provide responses to NRC questions about sample representativeness and interpretation (for core damage). This submittal completes our action and should allow close-out of NUREG 0737, item II.B.3 and SER item no. 81.

Very truly yours,

N W Curtis

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

DPM/mks

Attachments

cc: R. Perch
G. Rhoads

A046

Attachment A: RCS Sample Representativeness

NRC Concern:

That the reactor coolant liquid sample which is taken from the jet pump diffuser will be diluted to an uncertain degree by the RCS make-up water source. This condition occurs when low volumes of steam are being generated significantly reduce the amount of moisture which leaves the core and are subsequently returned to the downcomer via the moisture separators. This condition can result in the samples being analyzed at lower concentrations of soluble species (chloride, boron, iodine, etc.) than are actually present in the core area.

PP&L Response:

The Post Accident Sample Station (PASS) accepts reactor coolant for sampling from the non-calibrated jet pump instrument line for jet pump number 14. Use of this sample point to obtain representative reactor coolant samples, when the reactor is at high pressures, requires that certain preconditions be met. These are as follows:

A. Reactor accident conditions typify those following non-break or small break accidents. Means exist to maintain reactor water level such as to allow the circulation of water from inside the shroud area to the jet pump intake.

B. Reactor residual power is greater than or equal to 1 percent of rated power in order to assure adequate recirculation flow.

These preconditions are verified through use of plant procedures which currently exist or are under development. A synopsis of these procedures is provided as an attachment to this letter.

Preconditions are met by maintaining reactor water level at or near normal water level using Emergency Procedures. At decay powers of greater than 1% rated power, core flow is estimated to be 10% of rated core flow during natural circulation conditions, thus assuring that a representative fluid will exist at the jet pump instrument/sample tap.

For small steam line breaks or non-break accidents sufficient make-up water will be added to maintain level in the reactor vessel. This make-up water amounts to very small fractions of the total core flow a representative fluid will exist at the jet pump instrument/sample tap.

For small steam line breaks or non-break accidents, makeup water is pumped in to remove decay heat and to make up for steam loss through the break. This makeup water amounts to approximately 2% of the core flow present. Even for small liquid line breaks, the makeup water flow rate is estimated to be less than 18% of the core flow present. The resultant dilution of the RCS sample is deemed to be acceptable.

Attachment B: Suppression Pool Sample Representativeness

NRC Concern:

That the suppression chamber samples, due to the location of the sample points relative to RCS safety valve discharge points, will either be excessively diluted or virtually undiluted resulting in erroneous estimates of core damage. We have requested the applicants to provide information to demonstrate that these sample points are located such that adequate mixing will occur and the samples are representative of the mixture rather than only the discharged fluid.

PP&L Response:

The PASS is designed to accept samples from either of two Residual Heat Removal (RHR) loops. The sample points are taken down stream of the RHR heat exchangers from loop piping which is in service and sees flow during suppression pool cooling, shutdown cooling, or Low Pressure Coolant Injection (LPCI) modes of operation. The attached synopsis of the plant procedures demonstrates that the operators confirm proper RHR system alignment and operation (including the allowance of a 30 minute system flush) in order to assure that a representative sample is taken. When RCS samples are to be taken with the RHR system in the shutdown cooling mode, sample representativeness will be assured by raising the water level in the reactor vessel to the point where water flows directly from the steam separators to the shroud area. This corresponds to a water level increase of approximately 18 inches above normal water level.

The Susquehanna suppression pool is designed such that during steam relief modes, thermal, and therefore radioisotope mixing is optimized by location and orientation of the T-quenchers. In addition, a review of the SRV discharge points relative to the RHR pump suction locations was performed and no direct discharge flow path from the SRV to the RHR pump suction inlets was discovered.

The suppression pool atmospheric samples are taken from taps also used for containment hydrogen and radiation analysis. They are located on opposite sides of the wetwell vapor space. Their location is optimized to obtain representative samples.

Attachment C: Core Damage Calculation

NRC Concern:

That a procedure be provided to relate specific radionuclide concentrations to the estimated extent of core damage.

PP&L Response:

A procedure for the determination of the extent of core damage, under accident conditions, has been developed by General Electric Company for use by BWR plants possessing a GE PASS, such as Susquehanna. A copy of this procedure, RPE 81CCL01, is attached. This procedure provides a method for determining the degree of reactor core damage from the measured fission product concentrations in either the water or gas samples taken with the PASS. For reasons discussed previously in this letter we believe the proper operation of PASS and the respective sample source systems will result in sample data representative of actual plant conditions. Use of the enclosed GE procedure will then allow a core damage percentage calculation to be performed. It must be noted, that use of techniques and methodology such as this does not result in precise knowledge of actual core damage. For this reason precautions are being added to the appropriate emergency manuals which require that other factors be considered and balanced with the PASS results in the final quantification of core damage. These factors include:

1. Comparison to hydrogen levels inside the primary containment

2. Reference to reactor water level transients

3. Reference to containment radiation monitors

4. Consideration of recent core power history and the relative time in the core life

5. Reference to the type of accident and how it would perturb expected radioisotope releases.

In addition, PP&L intends to implement and maintain on a weekly basis a program to be used to calculate various radioisotopes in the core. When, this program is in place, it will also be used as a source of base-line information for the initial core activity content, to be used in the calculation of core damage percentages.