5/15/78

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)
DISTRIBUTION FOR INCOMING MATERIAL 50-261

REC: Y
NRC

ORG: UTLEY E E
CAROLINA PWR & LIGHT

DOCDATE: 05/08/78
DATE RCVD: 05/15/78

DOCTYPE: LETTER NOTARIZED: NO
SUBJECT:

COPIES RECEIVED
LTR 1 ENCL 0

RESPONSE TO NRC REQUEST OF 01/25/78... FURNISHING APPLICANT"S SCHEDULE RE EVALUATION OF THE EFFECTS OF ASYMMETRIC LOSS OF COOLANT ACCIDENT OF SUBJECT FACILITY, ASSESSING THE SAFETY OF THE PLANT.

PLANT NAME: H B ROBINSON - UNIT 2

REVIEWER INITIAL: XJM
DISTRIBUTOR INITIAL: ML

****************** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS ******************

GENERAL DISTRIBUTION FOR AFTER ISSUANCE OF OPERATING LICENSE.
(DISTRIBUTION CODE A001)

FOR ACTION: BR CHIEF SCHWENCER**LTR ONLY(7)

INTERNAL:
REG FILE**LTR ONLY(1)
~~I & E**LTR~~ ONLY(2)
HANAUER**LTR ONLY(1)
EISENHUT**LTR ONLY(1)
BAER**LTR ONLY(1)
EEB**LTR ONLY(1)
J. MCGOUGH**LTR ONLY(1)
NRC PDR**LTR ONLY(1)
OELD**LTR ONLY(1)
CHECK**LTR ONLY(1)
SHAO**LTR ONLY(1)
BUTLER**LTR ONLY(1)
J COLLINS**LTR ONLY(1)

EXTERNAL:
LPDR'S
HARTSVILLE, SC**LTR ONLY(1)
TIC**LTR ONLY(1)
NSIC**LTR ONLY(1)
ACRS CAT B**LTR ONLY(16)

DISTRIBUTION: LTR 40 ENCL 0
SIZE: 2P

CONTROL NBR: 781320182

************************************ THE END ************************************

CP&L

Carolina Power & Light Company

May 8, 1978

FILE: NG 3514 (R)

Office of Nuclear Reactor Regulation
ATTN: Mr. Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
PRIMARY COOLANT LOOP SUPPORT ANALYSIS

Dear Mr. Stello:

Carolina Power & Light Company was requested by your letter of January 25, 1978, to inform you within 90 days of the detailed schedule for providing an evaluation of the effects of asymmetric loss of coolant accident (LOCA) loads as described in Enclosure 2 of your letter. This submittal sets forth a realistic program of analytical evaluations that will, within the two-year time frame you have established, assess the safety of the plant.

Prior to issuance of your letter of January 25, 1978, a task group of utilities was formed and, with the assistance of Westinghouse Electric Corporation, has outlined the steps necessary to complete an evaluation of the type you have requested. The evaluation program prepared by the task group has been broken into three parts, or phases, to facilitate the work effort and to gain some advantage by performing generic analyses applicable to several plants of similar design.

Phase A, which is essentially complete, consists of a detailed examination of primary loop characteristics for each plant involved in the task group effort to categorize and group the plants for the analyses. Plant groupings are based on plant characteristics such as reactor vessel diameter, steam generator type, steam generator and reactor coolant pump support structures and primary loop temperatures.

Phase B will evaluate the postulated ruptures which have the greatest effect upon the steam generator and reactor coolant pump supports. This evaluation will consist of generic scoping analyses for the plant groupings determined in Phase A of the project using standard loop analysis procedures. The breaks to be postulated for the analyses are the hot leg at the reactor outlet nozzle, the crossover leg between the steam generator and the reactor coolant pump, and the cold leg at either the pump discharge or the reactor

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

A001 | S *
1 | 0

vessel inlet nozzle. The results of these scoping analyses will be used to determine if specific plant analyses will be required to better assess the adequacy of the support systems. During this phase, bounding reactor coolant loop hydraulic forcing functions will be developed and generic subcompartment pressurization studies will be performed.

Phase C will address the reactor vessel responses for the postulated pipe ruptures at the reactor vessel inlet and outlet nozzles. Analyses will be performed on a group basis where asymmetric cavity pressure loads, vessel internal hydraulic loads, and vessel support characteristics are similar. The need for specific plant analyses will be based on the results of the generic analyses. Prior to the initiation of the Phase C effort, studies will be performed to determine the effects of increased break opening times, more detailed downcomer annulus nodalization and structural damping on the MULTIFLEX calculations. Also, certain reactor cavity pressurization studies and the development of a mechanistic crack opening analysis will be performed.

The anticipated schedule for the program is presented below:

Program Effort	Completion Date
Phase A	Complete
Phase B Generic Group Analyses	June 15, 1978
Phase B RCL Hydraulic Forcing Functions	June 30, 1978
Phase B Subcompartment Pressurization	November 1, 1978
Phase B Specific Plant Analyses	Early, 1979
MULTIFLEX Modifications	January 1, 1979
Reactor Cavity Pressurization Studies	January 1, 1979
Mechanistic Pipe Break Analysis	January 1, 1979
Phase C Generic Group Analyses	First Quarter 1979
Phase C Specific Plant Analyses	January 1, 1980

Should the generic analyses indicate that some corrective action is necessary, potential plant modifications may be included in specific plant analyses. These modifications may be later incorporated in the plant, or augmented in-service inspection or probability analyses may be proposed as an alternate solution. In either case, it is our intent to demonstrate the safety of long-term continuous operation.

We trust this information is suitable for your use. Please inform us if you have additional questions concerning the program outlined above.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

DBW/gsm