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AUTH.NAME AUTHOR AFFILIATION
ZIMMERMAN,S.R. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Forwards responses to request for addl info re TMI Action Item II.D.1, "Performance Testing of Relief & Safety Valves.

DISTRIBUTION CODE: A046L COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 22
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NOTES: 05000261
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Carolina Power & Light Company

JUN 1 4 1985

SERIAL: NLS-85-217
NRC TAC NO. 44616

Director of Nuclear Reactor Regulation
Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
NUREG 0737, ITEM II.D.1 -
PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES

Dear Mr. Varga:

Enclosed are responses to your request for additional information regarding TMI Item II.D.1, Performance Testing of Relief and Safety Valves. Please contact Mr. Stephen D. Floyd at (919) 836-6901 if you have any questions concerning this issue.

Yours very truly,

S. R. Zimmerman

S. R. Zimmerman
Manager
Nuclear Licensing Section

SRZ/SDF/mf (1507SDF)

Enclosure

cc: Dr. J. Nelson Grace (NRC-RII)
Mr. G. Requa (NRC)
Mr. H. Krug (NRC Resident Inspector - RNP)

A046
1/1

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

UNRESOLVED QUESTIONS ON TMI ACTION NUREG 0737, II.D.1 FOR ROBINSON 2

Question 1

The response to Question 1 states that the feedwater line break accident is not a part of the Robinson 2 licensing basis since the plant was licensed prior to issuance of Regulatory Guide 1.70, Rev. 2. This response is not acceptable. Item II.D.1 in NUREG-0737 specifically requires that PORVs and safety valves be qualified for fluid conditions resulting from transients and accidents referenced in Regulatory Guide 1.70, Rev. 2. The feedwater line break accident is specifically defined in Regulatory Guide 1.70, Rev. 2. Additionally, from the staff review of other plant-specific responses to Item II.D.1, it is clear that for many Westinghouse plants, the feedwater line break accident is the limiting case for providing high pressure liquid to the safety valves, a fluid for which they were not specifically designed originally. This is exactly the type of concern that NUREG-0737, II.D.1 was established to address. In accordance with the requirements of the NUREG, we require that information be provided to demonstrate that the PORVs and safety valves will function as required to assist in safe shutdown of the plant and will not experience any degradation that would inhibit safe plant shutdown if exposed to the Feedwater Line Break Accident.

Response

At H. B. Robinson, feedwater line break results in an immediate cooldown of the Reactor Coolant System. This is reflected in:

1. Section 15.2.8 of the Updated HBR FSAR

2. Section 15.2.8 of XN-NF-83-72, Revision 2, Supplement 1; an Exxon report submitted to the NRC as a part of Cycle 10 reload analyses

3. Section 15.2.8 of Attachment I to the SER from the NRC for Cycle 10 operation.

As a cooldown event, the primary coolant shrinks instead of challenging the Safety and Power Operated Relief Valves.

Unlike newer Westinghouse plants that feature steam generators with integral preheaters, the feedwater enters the steam generators at H. B. Robinson through a feedring (sparger) that is located above the U-tubes. Because the elevation of this feedring is comparable to the initial water level in the steam generator shell, it will be primarily steam (instead of water) that is removed from the steam generator. The latent heat of vaporization is removed from the liquid and cools the Reactor Coolant System in much the same manner as the main steam line break transient. Therefore, the feedwater line break accident is enveloped by the main steam line break accident which demonstrates that no high pressure liquid is discharged from the PORVs and safety valves.

Question 3

The response to Question 3 stated that blowdowns of greater than 10 percent have been shown to be acceptable. However, a report was not provided that discussed the calculations. In addition, the EPRI tests showed that for the two Crosby valves that bracketed the Robinson 2 valves, blowdowns of significantly greater than 10 percent were observed. The response to Question 5 and 7 stated that the ring settings used will produce 5 percent blowdown, but this conclusion is based on production tests rather than the full-flow EPRI tests. The staff position is that the expected blowdown for the in-plant ring settings should be established based on the EPRI test data and that a report be provided that demonstrates that with this expected blowdown, the core can be adequately cooled. Also, if the pressurizer fills, the operability of the safety valves while discharging liquid must also be addressed.

Response

As stated in our earlier response, Crosby Production Tests on the H. B. Robinson safety valves demonstrated blowdown values of 5 percent. The ring settings for the H. B. Robinson safety valves were established by the same methods as the "as stamped" final ring settings for the safety valves tested at EPRI. These methods include performance of a steam operational test on each safety valve to determine the best suited ring setting to assure proper and stable valve performance. A review of the EPRI long pipe data for 3K6 and 6M6 Safety Valves indicates for test valves using manufacturers ring settings, blowdowns were less than 13 percent for the 6M6 valve and less than 10 percent for the 3K6 valve.

As noted in Table 4.4 of EPRI Report NP-2770-LD, Volume 6, the Crosby 6M6 test valve achieved rated flow for each of the tests reported at 3 percent accumulation regardless of the ring setting used in the test. A review of EPRI Tables 4-3 and 4-4 in Volume 5 of EPRI Report NP-2770-LD reveals that for steam tests of the 3K6 valve where blowdown was measured to be less than 10 percent, flow rates of 119-122 percent of rated flow at 3 percent accumulation were reported. The EPRI tables indicate that lower than rated flows occurred at blowdowns greater than 15 percent using lowered ring settings.

Therefore, if the blowdown is below 15 percent, the EPRI data indicates rated flow can be achieved. Since 'as-shipped' ring settings resulted in blowdowns less than 13 percent for the 3K6 and 6M6 safety valves using manufacturer's ring settings, and the H. B. Robinson safety valves were installed using manufacturer's ring settings, rated flow is, therefore, expected for the H. B. Robinson safety valves.

A calculation (attached) was performed using a very conservative value of blowdown (20 percent) to determine the effect on the pressurizer water level. The results of the calculation show that the peak pressurizer water level remains below the inlet piping to the safety valves. Consequently, reactor coolant is not discharged through these valves and the core cooling capability of the Reactor Coolant System is not reduced.

Question 5

In response to Question 5, ring settings were given from the upper or locked position rather than from the level position that was used for reference in identifying position in the EPRI test reports. The reference stated that comparisons of the ring settings were not necessary since the ring settings for the in-plant valves were established from production tests rather than from the EPRI tests. The staff position is that full flow tests such as the EPRI tests are required to justify successful operation. Provide the equivalent ring setting referenced to the level position so that comparisons can be made with the settings used in the EPRI tests. Also, if the ring settings do not correspond to those used in the tests, the effect on operability, stability, and ability to pass rated flow should be discussed.

Response

The following lists the H. B. Robinson safety valve ring settings referenced to the level position for the guide ring.

Valve	Nozzle Ring (Notches from Locked Position)	Guide Ring (Notches from Level Position)
RC-551A	-7	-54
RC-551B	-7	-42
RC-551C	-7	-45

The Robinson valve ring settings developed by the Crosby Production test methods should have performance characteristics similar to those EPRI test valves operated at 'as-shipped' (manufacturer's preset) ring settings. The guide ring positions for the Robinson safety valves are different from valve to valve (as shown above) due to each valve having a different guide ring level position. The difference is due to part tolerance stack-up within the individual valves. The difference in guide ring level positions for the Robinson safety valves compared to the EPRI test valves is due to different ring movement per notch for each valve size.

The EPRI test program was formulated to resolve this issue without the need for individual valve testing. It is requested that the NRC reconsider the need for more extensive information regarding ring settings that would require additional testing estimated to take approximately two years and cost approximately $470K. As stated in our response to Question 3 above, the H. B. Robinson valves are expected to be bounded by the range of blowdowns experienced in the EPRI testing program and an analysis using an even more conservative blowdown value demonstrates adequate core cooling. Therefore, it is our conclusion that additional testing to further correlate our valves to the EPRI-tested valves is not necessary.

Question 7

In the EPRI tests for the two Crosby valves that bracketed the Robinson 2 valves, several tests demonstrated valve chatter and the tests were interrupted by opening the valves manually to limit valve damage. The response to Question 7 stated that the EPRI tests used bounding conditions of ring settings, pressures, and pipe lengths. The response also stated that the plant valves had better settings and shorter pipe lengths and, therefore, plant valves would have stable operation. The response stated there is a smaller volume of water in the loop seal. Provide numerical comparisons and explain the bases for the conclusions.

Response

Our response to Question 7 did not state the plant valves had better ring settings, but that the H. B. Robinson loop seals are shorter than those tested by EPRI, and more stable results can, therefore, be expected due to the reduced acoustic pressure drop.

Loop seal arrangement information for the H. B. Robinson valves was provided in Table 2-2 of the original submittal and are reproduced below. The table also shows loop seal volume for H. B. Robinson versus the 6M6 test valve.

SAFETY VALVE INLET PIPING COMPARISON

	Typical H. B. Robinson Inlet Piping	3K6 Inlet Piping	6M6 Inlet Piping
Length of straight pipe, in.	89	60	61
Number of 90° elbows	3	4	-
Number of 180° bends	-	-	2
Number of 45° elbows	1	-	-
Misc. fittings, in.	-	72	71
Loop seal water volume, ft^3	0.44	0.27	1.02

PROJECT RECORD (DAI 1.0) CP&L Nuclear Fuel Section	Page 1 of 17 Project No.: 85-0026
No. of Pages: 1 through 17 and a 15 page attachment	File No.: 1485-0026 2485-0026

Subject:
Effect of maximum pressurizer safety valve blowdown on pressurizer level swell for H.B. Robinson plant

Project Leader:
R.C. Gorman

Performing Personnel:
R.C. Gorman

Table of Contents | Page No.

This Activity ☐ supersedes ☒ supplements the following project(s):
Exxon report XN-NF-84-74, supplement 2

Project Complete: (Project Leader) R.C. Gorman	Date:
Project Verified: (Verifier) Mark A Pope	Date: 5/21/85
Reviewed and Accepted: (RPE) Talmage Clements (TBC)	Date: 5/21/85

PRCDR8-Z

Rev. 0

PROJECT PLAN (DAI 1.0) CP&L Nuclear Fuel Section	Page 2 of 17 Project No.: 85-0026	
Subject: Effect of maximum PZR safety valve blowdown on PZR level swell for HBR		
Description: As a part of qualifying the PZR safety valves, this calculation is performed to complement XN-NF-84-74, Supplement 2 (the latest Loss of Normal Feedwater Analysis) in response to an NRC question/request regarding TMI action NUREG 0737, II.D.1.		
Project Leader: R. C. Gorman	Target Date: 5-17-85	Acceptance: R. C. Gorman
Supporting Projects: Are supporting projects required? ____ Yes X No. If yes, a project plan shall be completed for each new project. Supporting projects shall be referenced below: (Project number).		
Safety Related: Is this design activity safety related? X Yes ____ No If no, Explain:		
Approved: (RPE) Talmage Clements (TBC) Verification shall be performed: ✓ Yes ____ No		
Verified by: (Date) Mark A Pope 5/21/85		

Rev. 0

PRCDR8-Z

B.) Purpose

The objective of this calculation is to show that the PZR does not completely fill with liquid. As a part of qualifying the PZR safety valves, maintaining a steam bubble in the PZR assures that there is no liquid relief through the safety valves. As the most severe "loss of heat sink" transient, Loss of Normal Feedwater is the case analyzed because it results in the maximum heatup and expansion of the primary coolant.

This design activity is necessary because the Exxon analysis assumed no blowdown (i.e., 0%) while the EPRI safety + relief valve testing program measured blowdowns significantly greater than 10%.

FORM NO. 2036 1/83

CP&L
Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R C Gorman	DATE: 5-13-85	PAGE NO.
	CHECKED BY: Mark A Pope	DATE: 5/20/85	4 OF 17
SUBJECT:			

c.) Method

Page 6 of the Exxon report XN-NF-84-74, Supplement 1 presents the minimum PZR steam volume resulting from heatup and expansion of the primary coolant. This design activity complements that report by calculating the incremental decrease in the steam volume due to depressurization of the RCS. The depressurization is from the opening setpoint pressure for the valves down to the minimum pressure where the safety valves reseat. The resulting increase in primary liquid volume is conservatively estimated as the difference of the corresponding state point specific volumes (times the RCS fluid mass).

FORM NO. 2036 1/83

CP&L Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R C Gorman	DATE: 5-21-85	PAGE NO. 5 OF 17
	CHECKED BY: Mark A Pope	DATE: 5/21/85	

SUBJECT:

Method (continued)

The effect of the increased valve blowdown upon the flowrate through the valve is not important for this calculation. Although prolonging the time the valve is open will allow more fluid to escape, as a conservative simplification this additional reduction in PZR fluid inventory is not considered:

a.) Although this additional loss of inventory will contribute to the depressurization of the PZR, it is already (inherently and implicitly) included in considering the effect of depressurization alone.

b.) By ignoring this additional loss of inventory more mass is retained in the PZR. This must necessarily result in a maximum estimate for level swell for a valve reseating at the same pressure.

FORM NO. 2036 1/83

CP&L
Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R. C. Gorman	DATE: 5-13-85	PAGE NO.
	CHECKED BY: Mark A Pope	DATE: 5/20/85	6 OF 17

SUBJECT:

D.) Assumptions

1.) The results presented on p. 6 of XN-NF-84-74, Supplement 2 are correct.

2.) There is complete separation of liquid and vapor phases within the PZR.

3.) As a conservative simplification, the average temperature of the primary coolant remains constant while the pressure drops: this is a "separate effects" calculation to add valve blowdown to the other factors considered in the Exxon analysis (i.e., temperature and inventory changes.)

4.) The results of the EPRI safety and relief valve testing program are applicable to HBR, i.e., the valve models and conditions chosen to be "representative" are truly so.

FORM NO. 2036 1/83

CP&L
Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.:	PREPARED BY: R. C. Gorman	DATE: 5-13-85	PAGE NO.
85-0026	CHECKED BY: Mark A Pope	DATE: 5/22/85	7 OF 17

SUBJECT:

E.) Computer Codes Used

none

F.) Computers Used

none

FORM NO. 2036 1/83

CP&L Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R. C. Gorman	DATE: 5-14-85	PAGE NO. 8 OF 17
	CHECKED BY: Mark A Pope	DATE: 5/20/85	

SUBJECT:

G) Calculations

the volume occupied by the primary coolant = (total mass)(average specific volume)

$$V = M v$$

$$\frac{\partial V}{\partial P} = M \left.\frac{\partial v}{\partial P}\right|_T$$

$$\Delta V \approx M\left(\frac{\partial v}{\partial P}\right)\Delta P$$

$$\approx \left(\frac{V}{v}\right)\left(\frac{\partial v}{\partial P}\right)\Delta P \quad \text{must be} \leq \quad 345\ ft^3$$

from XN-NF-84-74 Supplement 1

FORM NO. 2036 1/83

CP&L Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R.C. Dorman	DATE: 5-14-85	PAGE NO. 9 OF 17
	CHECKED BY: Mark A Pope	DATE: 5/20/85	

SUBJECT:

G.) contined

From Billy Hinton in CP&L NELD, a value of 20% blowdown for the safety valves is to be evaluated, i.e., the pressure drop, ΔP, is from the nominal valve opening setpoint to a pressure 20% less. The nominal setpoint is given in Table 15.2.7-2 of XN-NF-84-74, Supplement 2, as 2500 psia.

$$\Delta P = (2500\ \text{psia})\,.2 = 500\ \text{psi}$$

from 2500 to 2000 psia

FORM NO. 2036 1/83

CP&L Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R. C. Gorman	DATE: 5-14-85	PAGE NO. 10 OF 17
	CHECKED BY: Mark A Pope	DATE: 5/20/85	

SUBJECT:

G.) continued

from Figure 15.2.7-7 in XN-NF-84-74, Supplement 2, the peak average primary coolant temperature is between 580°F and 590°F.

The total RCS liquid volume is given in Table 5.1.0-1 of the HBR FSAR as 9343 ft^3.

The specific volumes are taken from the ASME Steam Tables:

	2000 psia	2500 psia
590°F	0.02286 $\frac{ft^3}{lb}$	0.02260 $\frac{ft^3}{lb}$
580°	0.02245 $\frac{ft^3}{lb}$	0.02222 $\frac{ft^3}{lb}$

FORM NO. 2036 1/83

CP&L Carolina Power & Light Company | INCORE ANALYSIS UNIT | ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R.C. Gorman DATE: 5-14-85 CHECKED BY: Mark a Pope DATE: 5/20/85	PAGE NO. 11 OF 17

SUBJECT:

G.) continued

$$\left.\frac{\partial v}{\partial P}\right|_T \approx \frac{(v @ 2000\,psia + T) - (v @ 2500\,psia + T)}{2000\,psia - 2500\,psia}$$

@ 590°F

$$\Delta V \approx \frac{(9343\ ft^3)}{(0.023\ \frac{ft^3}{lb})} \frac{(0.02286 - 0.02260\ \frac{ft^3}{lb})}{(-500\ psi)} (-500\ psi)$$

$$= 105.6\ ft^3$$

@ 580°

$$\Delta V \approx \frac{9343}{0.022} (0.02245 - 0.02222)$$

$$= 97.7\ ft^3$$

FORM NO. 2036 1/83

CP&L
Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: R. C. Gorman	DATE: 5-14-85	PAGE NO. 12 OF 17
	CHECKED BY: Mark A. Pope	DATE: 5/20/85	

SUBJECT:

H.) Computer Inputs/Outputs

none

I.) References

The following references are provided as an attachment:

a.) telephone minutes setting valve blowdown at 20% — p.1 of 15

b) excerpt from NRC questions and W response — p.2 to 7

c.) excerpt from a manual discussing terminology — p. 8 to 10

d.) telephone minutes with Exxon verifying that SLOTRAX modeled 0% blowdown — p.11

e.) excerpts from XN-NF-84-74, Supplement 2 — p.12 to 14

f.) Table 5.1.0-1 from HBR FSAR giving RCS volume — p.15 of 15

J.)

DESIGN INPUT RECORD (DAI 3.0)	PAGE 13 OF 17
CP&L NUCLEAR FUEL SECTION	PROJECT NO.: 85-0026

SUBJECT:

LIST OF DESIGN INPUTS USED IN PERFORMING DESIGN ACTIVITY	SOURCE OF INPUT	QA RECORD FILE/LOCATION
PZR is designed to accomodate level surges	HBR FSAR 5.4.6	several locations e.g. 285/2C5 for Nuclear Fuel
one of the objectives of LONFW analysis is to show that liquid is not expelled through the PZR safety valves	XN-NF-84-7, supplement 2, section 15.2.7.1, (p.3)	285: D. Cook file area
The consequences of water discharge through the PZR safety valves is not addressed because the peak PZR water level remains below the inlet piping to the safety & relief valves.	NRC "questions" and CP&L responses re TMI action NUREG	filed under 0120-500-XXX-XXX in NELD files on 6th floor (Zelma Ausley)

PROJECT LEADER/DATE	VERIFIER/DATE	RPE/DATE
RCG/5-14-85	MAP 5/20/85	TBC 5/21/85

REV. 0

PRCDR8-W

FORM NO. 2036 1/83

CP&L Carolina Power & Light Company

INCORE ANALYSIS UNIT

ENGINEERING CALCULATION

PROJECT NO.: 85-0026	PREPARED BY: RC Gorman	DATE: 5-14-85	PAGE NO. 14 OF 17
	CHECKED BY: Mark A Pope	DATE: 5/21/85	

SUBJECT:

K.) Summary of Results

As shown by the calculations, the increase in the primary coolant liquid volume resulting from reducing system pressure to account for safety valve blowdown is approximately 100 ft^3. Since the steam bubble that would otherwise occupy this space in the PZR is more than three times this size, the PZR can comfortably accommodate this additional insurge without "going solid".

L.) Conclusions

Maintaining a steam bubble in the PZR assures that the PZR safety valve operability will not be threatened by solid water flow. Loss of sufficient inventory to threaten core cooling is also precluded.

VERIFICATION RECORD (DAI 2.0) CP&L NUCLEAR FUEL SECTION		Page 15 of 17 PROJECT NO.: 85-0026
PROJECT LEADER: Roger Gorman	VERIFIER: Mark Pope	TARGET DATE: 5-23-85

SUBJECT:

VERIFICATION SCOPE (CHECK AS APPLICABLE):

- a. ☒ Complete Verification Checklist (Attachment 2)
- b. ☐ Spot Check of Mathematics
- c. ☒ Complete Check of Mathematics
- d. ☐ Comparison to Previous Calculations of a Similar Nature
- e. ☐ Partial Recalculation (Provide Details Below)
- f. ☐ Alternate Calculations
- g. ☐ Other (Provide Details Below)

OTHER/COMMENTS:

Check Assumptions also.

DESCRIPTION OF VERIFICATION (ACTIVITIES, FINDINGS AND RESOLUTIONS):

A complete check of the mathematics was made and no errors were found. The assumptions were evaluated and found to be reasonable for this application.

VERIFIER'S INITIALS/DATE:	RPE'S INITIALS/DATE:
MAP 5/21/85	TBC 5/21/85

VERIFICATION CHECKLIST (DAI 2.0) CP&L NUCLEAR FUEL SECTION	Page 16 of 17 PROJECT NO.: 85-0026

The listing below is taken from ANSI N45.2.11 and indicates the general subjects to be considered during verification:

		Yes	N/A
1.	Were the design inputs correctly selected, documented, and incorporated into design?	☑	☐
2. a.	Are assumptions necessary to perform the design activity adequately described and reasonable?	☑	☐
b.	Where necessary, are the assumptions identified which could not be verified at the time the Design Activity is performed, but need to be verified at a later stage in the design process?	☐	☑
3.	Are the appropriate quality and quality assurance requirements specified?	☑	☐
4.	Are the applicable codes, standards, and regulatory requirements including issue and addenda properly identified; and are their requirements for design met?	☑	☐
5. a.	Are all computer codes used identified along with version, alias, computer system, inputs and outputs?	☐	☑
b.	Are all codes being used of production status and have they been qualified for the intended end use? or Have these codes received a temporary waiver (CCQI-10) for use in this activity? or Has the use of this code's results been identified as an assumption? or Has the code been designated as a data manipulator and the results verified as a part of this Design Activity?	☐	☑
c.	Are the codes suitable for the activity?	☐	☑
d.	Do all the computer models (Noding, Time Steps, etc.) adequately represent the physical systems?	☐	☑
6.	Has applicable operating experience been considered?	☑	☐
7.	Was an appropriate design method used considering the purpose and type of activities and the use and acceptability of the results (i.e., Margin to Limits)?	☑	☐

VERIFIER'S INITIALS/DATE:
MAP 5/21/85

VERIFICATION CHECKLIST (CONT.) (DAI 2.0)

PROJECT NO.: 85-0026

CP&L NUCLEAR FUEL SECTION

		Yes	N/A
8.	a. Is the output reasonable compared to inputs?	☑	☐
	b. Is the magnitude of the result reasonable?	☑	☐
	c. Are the direction of trends reasonable?	☑	☐
9.	Has the design properly considered radiation exposure to the public and plant personnel?	☐	☑
10.	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?	☑	☐
11.	Are the requirements for record preparation, review, approval, retention, etc., adequately specified?	☑	☐
	Items 12-19 apply to hardware items only.		
12.	Are the specified parts, equipment, and processes suitable for the required application?	☐	☑
13.	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?	☐	☑
14.	Have adequate maintenance features and requirements been specified?	☐	☑
15.	Are accessibility and other design provisions adequate for the performance of needed maintenance and repair?	☐	☑
16.	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?	☐	☑
17.	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?	☐	☑
18.	Are adequate handling, storage, cleaning, and shipping requirements specified?	☐	☑
19.	Are adequate identification requirements specified?	☐	☑

VERIFIER'S INITIALS/DATE:
MAP 5/21/85