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November 15, 1979

1-119-10

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Nuclear One-Unit 1
Docket No. 50-313
License No. DPR-51
PORV and Safety Valve Lift Frequency and Mechanical Reliability
(File: 1510.1)

Gentlemen:

In reponse to your request for additional information concerning the above subject, the following is provided.

Request 1

According to statements made by B&W, there are approximately 146 documented occasions where PORV actuation occurred at B&W facilities prior to the accident at Three Mile Island, Unit 2 (TMI-2). For each of these events which have occurred at your facility(ies), provide the following information:

a. The cause of the event;

b. the initial power level prior to the transient;

c. indicate which of these transients caused the reactor to trip on high RCS pressure and/or caused the safety valves to lift; and,

d. if you assume that the present setpoints for high RCS pressure trip and PORV actuation were in effect at the time of each of these transients, estimate whether either of the following would have taken place:

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(1) PORV actuation, and

(2) lifting of the safety valves.

(for this item assume no credit for the anticipatory control-grade reactor trip on loss of feedwater or turbine trip).

Response

The requested information has been compiled in Attachment 1.

Request 2

Provide a complete listing of reactor trips for your facility (ies) which have occurred subsequent to the revised setpoints for PORV actuation and high RCS pressure trip. This listing should include the following items:

a. the cause of each event;

b. the initial power level prior to the transient;

c. indicate which of these transients caused the PORV and/or safety valves to open; and,

d. if the old (pre-TMI-2) setpoints for high RCS pressure and PORV actuation were in effect at the time of these transients, estimate whether any or all of the following would have taken place:

    (1) PORV actuation;

    (2) reactor trip on high RCS pressure, and

    (3) lifting of the safety valves.

Response

This information has been compiled and is presented in Attachment 2.

Request 3

Provide an estimate of the increase in reactor trip frequency since lowering the high pressure trip setpoint and adding the anticipatory reactor trip. Include a review of the design

criteria for the number of reactor trips over the plant life and evaluate the effect of the increase in trip frequency on these criteria. Also, provide the basis for the acceptable number of reactor trips in terms of the limiting component(s).

Response

A. An increase in the reactor trip frequency can be expected to occur as a result of lowering of the high pressure trip setpoint and addition of the anticipatory reactor trip. In order to estimate the change in trip frequency due to these changes, specific pieces of information and assumptions are necessary. For instance, the Rx trip frequencies before and after implementation of the design changes are needed.

Following commerical operation through the time of the design changes in question, 24 reactor trips occurred in a period of 53 months for a frequency of 0.45 trips/month.

Following the design changes, 2 reactor trips have occurred in a period of approximately 5 months for a frequency of 0.40 trips/month.

A simple comparison indicates a slight decrease in frequency since the design changes. However, one should consider that as operating experience and history was gained, the frequency of trips on ANO-1 has decreased. For instance, in 1977 and 1978 only 7 reactor trips were experienced yielding a trip frequency of 0.29 trips/month. Comparing this with the past design change frequency of 0.40 trips/month, indicates an increase of 0.11 trips/month.

Probably a more realistic measure of the increase in trip frequency due to the design changes is to evaluate past plant transients which did not result in reactor trips but can reasonably be expected to have caused trips if the PORV setpoint, high RC pressure trip setpoint and anticipatory changes had been in effect. Review of past data shows that 4 such transients occurred during the 53 months following commercial operation and before the design changes. This is an increased frequency of 0.08 trips/month.

In evaluating the consequences of the design changes the following must be considered:

1. There has been a relatively short period of operation since the changes, and

2. As operators become familiar with the revised setpoints and operating conditions, it is reasonable to assume the trip frequencies may decrease.

B. The structural design criterion for the number of reactor trips over the life of the plant is to keep the fatigue usage factors of all RCS components below 1.0 as supported by the component stress analysis. In general, this usage factor is made up of contribution due to all specified transients. Since the largest contribution to the fatigue usage factor is attributable to heatup and cooldown transients, with reactor trips producing only a small effect, the increase in trip frequency should only have a small effect relative to plant life.

As a part of the total allowable transient picture, 400 reactor trips are specified. Assuming a 40-year life, this translates into 10 trips per year or 0.83 per month. Since the overall frequency for ANO-1 commercial operation is 0.45 per month there does not appear to be a reason for concern. However, for the reasons discussed in A above, it is premature to draw any conclusions over the life of the plant based on the little data available with these setpoints.

C. To determine the acceptable number of reactor trips in terms of the limiting component(s), it is necessary to review the stress report for each component and plant and evaluate the fatigue usage factor.

If the number of trips were to exceed 400 on any plant, that plant would have to be re-analyzed based on actual transients and the limiting component would be a function of these actual transients plus those that would be expected throughout the remainder of the plant's life.

Very truly yours,

David C. Trimble

David C. Trimble
Manager, Licensing

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ATTACHMENT 1
ARKANSAS NUCLEAR ONE - UNIT 1
REACTOR TRIPS WITH A PORV ACTUATION

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PZR Safety Valves Lifted?	If present Setpoints Had Been Used: PORV Actuation?	If present Setpoints Had Been Used: Lift Safety Valves?
10-15-74	Loss of Feedwater	Hi RCP	"A" FWP Tripped on High Vibration	45	No	No	No
12-6-74	Loss of Feedwater	Pressure/Temp	Loss of Vacuum Due to "B" Main Chiller Getting Wet and Shorting	60	No	No	No
- - - - - - -Prior to Commercial Operation - - - - - - -							
1-6-75	Load Rejection	Hi RCP	Generator Tripped on Differential Current Due to Loss of Buss Cooling	98	No	No	No
6-6-75	Instrument Failure	PWR/Imbalance Flow	Loose Connection on Loop "B" T_C Signal	100	No	No	No
7-23-75	Loss of Feedwater	Pressure/Temp	Adjustment on FW Heater Level caused Trip of Htr Drain Pump ⇒ FWP Trip	94	No	No	No
7-8-76	Loss of Feedwater/Power Supply Failure	Hi RCP	Inst. Techs Shorted NNI Power Supply	94	No	No	No
9-23-76	Turbine Trip	Pressure/Temp	Turbine Tripped When Vibration Trip Module was Reinserted by Technician	100	No	No	No
12-20-76	Rod Drop/Manual Runback	Hi RCP	Rod 8 in Group 4 Dropped. Manual Runback Too Rapid	100	No	No	No
6-19-78	Turbine Trip	Hi RCP	Technician or Operator Error in Opening Wrong Feeder Breaker	100	No	No	No
10-13-78	Instrument Failure	Pressure/Temp	RPS Channel "B" RC Flow Signal Failed	100	No	No	No
12-20-78	Instrument Failure	Pressure/Temp	Low Steam Pressure Caused by LVDT Linkage Breaking	99	No	No	No

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ATTACHMENT 2

ARKANSAS NUCLEAR ONE-UNIT 1

REACTOR TRIPS SINCE TMI-2

DATE	7-8-79	8-13-79
Transient Classification	Rx Trip on High RCS Press.	Turbine Trip
Cause of Transient	Governor Valve Control Failure	Switchyard Relay Failure
Initial Power Level	75	75
PORV Lifted?	No	No
PZR Safety Valves Lifted?	No	No
<u>If Old Setpoints Had Been Used</u>		
PORV Actuation?	Yes	Yes
Safety Valves Lifted?	No	No
Trip on High Pressure?	Unable to Determine - Too Close	

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