Central File
50-261

Carolina Power & Light Company

November 18, 1976

FILE: NG-3513 (R) SERIAL: NG-76-1505

Mr. Norman C. Moseley, Director
U.S. Nuclear Regulatory Commission
Region II, Suite 818
230 Peachtree Street, N.W.
Atlanta, Georgia 30303

Dear Mr. Moseley:

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET 50-261
LICENSE DPR-23
SUPPLEMENTAL RESPONSE TO IE BULLETIN 76-05

The initial response to IE Bulletin 76-05 on May 4, 1976, addressed actions to resolve the problem of thermal degradation in the Westinghouse BFD relays which resulted in the failure leading to Reportable Occurrence 76-2 on H. B. Robinson Unit No. 2. These actions consisted of immediate replacement of the normally energized relays and later replacement of the normally deenergized relays. As you recall, the relay failure was related to thermal degradation resulting from operation at higher voltages during battery charges. Although not normally subject to this higher voltage, it was believed a conservative approach to replace the normally deenergized as well. In the response, the latter replacement was scheduled for the current refueling outage.

Recent developments revealed a potential pin hang-up problem which affected all BFD relays including the replacement model (IE Circular 76-2). As a result of this new development, Westinghouse has issued a recommendation (Attachment) that we delay further preventative replacement of the BFD relays until availability of a new relay, designed to resolve the thermal problem as well as the pin hang-up problem. As commented in the attached letter, the production version of a prototype now being tested, could be available by the end of the year.

Due to the developments addressed above, we do not believe it prudent to replace the normally deenergized relays at this time. We, therefore, request relief from the commitment to replace these relays, made in the referenced response. Replacement of these

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relays will be based on further recommendations from Westinghouse following availability of a relay which will totally remedy the presently recognized BFD problems. These recommendations are expected by the end of the year. Should you require additional information in the interim, please contact us.

Very truly yours,

E. E. Utley

E. E. Utley
Vice President
Bulk Power Supply

JMC/CSB/dmc

Attachment

cc: Messrs. W. G. McDonald
E. Volgenau

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
230 PEACHTREE STREET, N.W. SUITE 818
ATLANTA, GEORGIA 30303

OCT 8 1976

In Reply Refer To:
IE:II:NCM
50-325
50-324
50-261

Carolina Power and Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
Engineering, Construction
and Operation
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The enclosed Circular 76-05 is forwarded to you for action. If there are any questions related to your understanding of the actions required, please contact this office.

Sincerely,

William B. Ravenscroft, Acting
Norman C. Moseley
Director

ENCLOSURE:
IE Circular 76-05

IE Circular 76-05 Date: October 8, 1976

HYDRAULIC SHOCK AND SWAY SUPPRESSORS - MAINTENANCE OF BLEED AND LOCK-UP VELOCITIES ON ITT GRINNELL'S MODEL NOS. - FIG. 200 AND FIG. 201, CATALOG PH-74-R

DESCRIPTION OF CIRCUMSTANCES:

Recent information has become available related to improper lock-up and bleed rates of certain ITT Grinnell shock and sway suppressors (snubbers). The control block or valve block on the snubbers involved is one which contains the mechanisms for a "dual orifice type" snubber valve arrangement. These control blocks or valve blocks can be identified by four locking nuts on the surface of the block. These nuts lock the stem of the valve control into position. Rotation of the stem modifies the orifices which control the lockup and bleed rates. The units in question, at this time, are those snubbers which have control block or valve block serial numbers from B-0001 through B-2000. These snubbers include 1 1/2 inch diameter (3 kip capacity) through 6 inch diameter (72 kip capacity) units.

ITT Grinnell first identified this problem in connection with their own testing program during the last two weeks of August 1975. The origin of the problem was traced by Grinnell to November of 1974 when a design change (known as Revision B) was initiated on Model Nos. - Figure 200 and Figure 201, Catalog PH-74-R snubbers, along with the addition of a Grinnell design specification for lock-up and bleed velocities. These new rates were to be 8 in/min. plus or minus 2 in/min. for lock-up and 4 in/min. plus or minus 1 in/min. for bleed. At that time production testing of completed snubbers included within the above series of serial numbers was completed on the basis of "go" or "no-go" testing and did not specifically determine the lock-up and bleed rates for each snubber.

Grinnell has determined that all snubbers involved should lock under seismic events as designed, but perhaps at a lower velocity. Additionally, the bleed rate has been determined to exist at a value as low as 1/8 in/min. This could result in increased piping stresses depending on the specific design. With the uncertainty of specific lockup and bleed rates, there is the possibility of adverse effects on the piping systems or components the snubbers were designed to protect. It has been stated, however, that to the best of Grinnell's knowledge, there were no lockup and bleed criteria defined by their customers for the plants identified as having utilized snubbers from the B-0001 through B-2000 series.

Even though lock-up and bleed rates on snubbers with adjustable orifices are now determined prior to shipment from the manufacturer, there is always the possibility that subsequent removal or tightening of the locking nuts on the control block or valve block will modify the orifice setting, thus affecting lock-up and bleed rates. At least two models of Grinnell's snubber control valves use some type of seal on the locking washers and in some cases these may have been replaced in the field to change the seal material or correct small leaks. In these cases, if proper procedures were not followed to maintain the stem portion in its original position, the orifice setting could change. The tightening of a lock nut itself could result in a change to the orifice. ITT Grinnell makes the following statement in the printed material furnished with each snubber assembly: "Adjustment of snubber valve requires equipment capable of measuring rate and load. DO NOT ATTEMPT TO RESET ADJUSTMENTS IN THE FIELD."

The recommended corrective action outlined by ITT Grinnell is to replace the entire valve assembly (2 required per snubber) known as the "barrel" in the case of the lock-up control, and the pins or stems (2 required per snubber) used to control the bleed rate. ITT Grinnell has prepared documents which prescribe the replacement procedures. The replacement parts are precalibrated. Therefore, field calibration of the entire snubber unit will not be necessary if the replacement parts are installed in accordance with the prescribed procedures. The tolerances on the precalibrated pins or stems associated with the bleed mechanism will exceed the original value of 4 inches per minute plus or minus 1 inch/min. The new bleed rate will be 4 inches/minute plus or minus 2 inches/minute.

Corrective action could also consist of recalibration of the existing snubbers with adjustments made to bring the operating characteristics of the units back within the tolerences specified by ITT Grinnell. Monitoring of relative rotation of the lock washer and stem could in the future be carried out if position marks were provided on the parts after calibration.

ITT Grinnell has identified the following facilities which have licenses under 10 CFR 50 which received snubbers within the group defined by the bounding serial number.

Plant	Number of Grinnell Snubbers Model Nos.-Fig. 200 & Fig. 201
Farley Nuclear Plant	239
Millstone Point Unit #2	185
North Anna	155
Davis Besse Unit #1	111
D. C. Cook	78
Calvert Cliffs	66
Beaver Valley	53
Diablo Canyon	41
Ginna	8
Kewaunee	2
Peach Bottom	2
Browns Ferry	1

ITT Grinnell has also informed the utilities who own these facilities regarding this matter; however, some of the series may be in use or intended for use at other reactor facilities as a result of utilities loaning or selling units to other utilities or by the fact that subcontractors or other suppliers might be involved.

ACTION TO BE TAKEN BY LICENSEE:

For all power reactor facilities with an operating license or a construction permit using or intending to use the above described snubbers on safety related systems or components:

1. Determine if any of the subject snubbers are installed or scheduled to be installed in safety-related systems at your facility.

   a. Provide a list containing each snubber identified and indicate the piping system or component with which it is associated.

   b. Indicate for each snubber its history of testing and test results as well as its maintenance history.

   c. For each snubber list the current required lock-up and bleed rates and what rates the snubbers were specified to meet by the original construction/purchase specification. Indicate the organization, by name, that was responsible for the preparation of the construction/purchase specification.

   d. If replacement parts are utilized instead of recalibration of the snubbers, provide an assessment of the effect of a bleed rate of 2 inches/minute which represents the new lower bound.

2. For snubbers within the above described series that do not have documentation available defining the lock-up and bleed velocities, provide a description of the actions to be taken to substantiate these characteristics. Also, provide the schedule to complete these verifications and modifications, if needed.

3. Review maintenance and modification procedures for the above described series of snubbers where such actions could affect the performance characteristics of the snubbers. Such actions include but are not limited to the disassembly of snubbers, replacement of seals, changing hydraulic fluid and the changing of any valve springs. Modify the procedures, as appropriate, to assure that any future maintenance or modifications will not change the performance characteristics.

4. Provide a report which contains the responses to items 1, 2 and 3 above within 90 days of the date of this Circular. Reports should be submitted to the Director of the NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Inspection Programs, Washington, D. C. 20555.

Approval of NRC requirements for reports concerning possible generic problems has been obtained under 44 U.S.C. 3152 from the U.S. General Accounting Office. (GAO Approval B-180255 (R0062), expires 7/31/77).