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PARAMETER IE-145

Closeout of IE Bulletin 81-01: Surveillance of Mechanical Snubbers

Prepared by R. S. Dean, W. J. Foley, A. Hennick

PARAMETER, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

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NRC FIN B1302

ABSTRACT

In the period from August 1974 to May 1980, failures of mechanical snubbers were described in event reports issued for nine facilities and in a NRC/IE study of the DOE Fast Flux Test Facility. In most failures, the snubbers were frozen and would not permit free piping motions during thermal transients. In some cases, the failed snubbers no longer provided seismic shock restraint. Because of concern about the reported failures of mechanical snubbers, standard technical specification revisions for snubber surveillance were issued by NRC/DL on November 20, 1980. IE Bulletin 81-01 was issued January 27, 1981 to require examination and testing of mechanical snubbers in safety-related systems at licensed facilities and at selected facilities under construction. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out per specific criteria for 73 (95%) of the 77 facilities to which it was issued for action. Followup items are proposed for use by NRC/IE to ensure satisfactory completion of corrective action at the remaining four (4) facilities.

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CLOSEOUT OF IE BULLETIN 81-01: SURVEILLANCE OF MECHANICAL SNUBBERS

INTRODUCTION

In accordance with the Statement of Work in Task Order 63 under Contract NRC 05-82-249, this report provides documentation for the closeout status of IE Bulletin 81-01. The following documentation is based on the records obtained from the IE File, the NRC Document Control System and the cognizant engineer's file.

IE Bulletin 81-01 was issued January 27, 1981 to require examination and testing of mechanical snubbers in safety-related systems at licensed facilities and at selected facilities under construction. Revision 1 of the bulletin was issued March 4, 1981 to change the list of selected facilities under construction.

For background information, IE Bulletin 81-01, revised pages of the bulletin, and Temporary Instruction (TI) 2515/49 with enclosures are included in Appendix A. Evaluation of utility responses and NRC/IE inspection reports is documented in Appendix B as the basis for bulletin closeout. Followup items are proposed in Appendix C for use by NRC/IE in ensuring satisfactory completion of corrective action. Abbreviations used in this report and associated documents are presented in Appendix E.

SUMMARY

1. The bulletin has been closed out for 21 facilities which have no mechanical snubbers of any make in safety-related systems, per Criterion 1.

   All of the facilities which have no mechanical snubbers have operating licenses.

2. The bulletin has been closed out for no facilities for which corrective action is to be tracked by NRC/IE on a separate system, per Criterion 2.

3. The bulletin has been closed out for 52 facilities for which NRC/IE inspection reports verify that corrective action has been completed satisfactorily, per Criterion 3.

These facilities are listed in Table B.2 on pages B-7 and B-8.

Refer to Note 5 on Page B-8. No closeout criterion applies completely to LaCrosse, for which an inspection report verifies that there were no mechanical snubbers. Combining summary items 3 and 4 and taking LaCrosse into account, note that 55 facilities had mechanical snubbers in safety-related systems.

4. The bulletin is being called open for the following four (4) facilities. Followup items are proposed in Appendix C for use by NRC/IE.

   Diablo Canyon 2   Duane Arnold   Zion 1,2

5. The 55 facilities which had mechanical snubbers in safety-related systems reported as follows:

   (a) At least 29 facilities had defective snubbers. Because of incomplete response, it was not possible to determine whether FitzPatrick had defective snubbers.

   (b) Three hundred seventy-six (376) (3.4%) of the 11071 snubbers reported were defective.

   Forty-three (43) (13.0%) of the 331 INC snubbers reported were defective.

   Three hundred twenty-three (323) (3.2%) of the 10237 PSC snubbers reported were defective.

   Ten (10) (2.0%) of the 503 A/D snubbers reported were defective.

   (c) Two hundred twenty-six (226) (5.9%) of 3857 PSC snubbers rated at .65 kips or less were defective.

   Eighty-three (83) (1.7%) of 4969 PSC snubbers rated at more than .65 kips but less than 50 kips were defective.

   Twelve (12) (0.9%) of 1330 PSC snubbers rated at 50 kips or greater were defective.

   Ten (10) (2.5%) of 394 A/D snubbers rated at 5 kips or less were defective. None of the 109 A/D snubbers rated at more than 5 kips were defective.

   Eighty-one (81) PSC snubbers are not included in this summary since quantities of failed and acceptable snubbers per rating are unknown. Two of these 81 snubbers are known to be defective.

6. The modes of failures and applicable numbers were:

For INCs,	low drag	28
	fatigue	8
	locked	3
	jamming	3
	high breakaway	1
For PSCs,	locked	112
	damage	22
	setting	20
	rough action	5
	restricted	5
	stiff	4
	partial stroke	4
	low drag	2
	high drag	2
	erratic, no restraint	2
	inertia mass separation	2
	high breakaway	1
	wrong safety wire	1
	no drag	1
	disengaged	1
	broken	1
	missing snap ring	1
	damaged bearings	1
	rotating barrel	1
	not reported	135
For A/Ds,	no drag	6
	restricted	4

7. The causes of failure and applicable numbers were:

For INCs,	vibration	8
	corrosion	2
	usage	1
	not reported	32
For PSCs,	abuse	39
	overload	29
	installation	24
	moisture	7
	dirty bushings	5
	corrosion	4
	water hammer	2
	immersion	2
	degraded lubricant	1
	broken anchor bolt	1
	not reported	209

For A/Ds	overload	6
	not reported	4

8. The most common forms of corrective action were:

> replacing INCs with equivalent PSCs
> replacing INCs with rigid struts
> replacing INCs with equivalent or larger hydraulic snubbers

> replacing PSCs with new PSCs, equivalent or larger
> making minor mechanical repairs to PSCs at site
> repairing PSCs at site by clearing and lubricating
> returning PSCs to factory for reconditioning

> replacing A/Ds with new A/Ds

CONCLUSIONS

1. The bulletin has been closed out for 73 (95%) of the 77 facilities to which it was issued for action.

2. The percentage of failed snubbers was much greater for INCs (13.0%) than for PSCs (3.2%) and A/Ds (2.0%).

3. The percentage of failed PSCs rated at .65 kips or less (5.9%) was much greater than those rated between .65 kips and 50 kips (1.7%) and those 50 kips or greater (0.9%).

4. All of the failed A/Ds were rated at 5 kips or less.

5. The most common modes of failure for INCs were low drag and fatigue.

6. The most common modes of failure for PSCs were locking, damage and incorrect setting.

7. The modes of failure of A/Ds were no drag and restriction.

8. The most common cause of failure of INCs was vibration.

9. The most common causes of failure of PSCs were abuse, overload and faulty installation.

10. The only reported cause of failure of A/Ds was overload.

REMAINING AREAS OF CONCERN

The bulletin is being called open for the four (4) facilities identified in preceding Summary Item 4. Followup items are proposed in Appendix C for use by NRC/IE, to ensure satisfactory completion of corrective action.

DEFINITIONS USED WITH CLOSEOUT CRITERIA

1. An acceptable response is a clear, written reply by utility personnel indicating compliance with actions required by the bulletin.

2. For holders of operating licenses when the bulletin was issued, corrective action is a remedial process involving separate bulletin action items 1.d, 2.d and 3.d applicable to licensees. These facilities are identified per Note 1 in Table B.1. Some facilities about to be issued operating licenses at that time are included.

3. For selected holders of construction permits when the bulletin was issued, corrective action is a remedial process involving bulletin action Item 1.c applicable to permit holders. These facilities are identified per Note 2 in Table B.1.

CRITERIA FOR CLOSEOUT OF BULLETIN

The bulletin is closed for facilities to which one of the following criteria applies:

1. Facilities for which an acceptable response has been received indicating that it has no mechanical snubbers of any make in safety-related systems.

2. Facilities for which an acceptable response has been received indicating that all mechanical snubbers in safety-related systems have been inspected and tested and that corrective action is planned or has been completed, and for which assurance of the followup of corrective action is provided by an NRC/IE tracking system such as the LER or 10CFR 50.55(e) tracking system.

3. Facilities for which an acceptable response has been received indicating that all mechanical snubbers in safety-related systems have been inspected and tested and that corrective action is planned or has been completed, and for which an NRC/IE inspection report has been received that indicates that corrective action has been completed satisfactorily.

Notes:

a.) Facilities to which Criterion 3 has been applied are listed in Table B.2.

b.) A letter from an NRC region indicating that the bulletin is considered closed for a facility and that a favorable inspection report will be issued suffices for application of Criterion 3. Letters of this type are noted in Table B.2.

APPENDIX A

Background Information

SSINS No.: 6820
Accession No.:
8005050075
IEB 81-01

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

January 27, 1981

IE Bulletin No. 81-01: SURVEILLANCE OF MECHANICAL SNUBBERS

Description of Circumstances:

Several instances of failures of mechanical snubbers supplied by International Nuclear Safeguards Corporation (INC) have been identified that indicate possible deficiencies in these snubbers. A summary of the failures that have occurred is provided below:

1. On August 9, 1974, the Tennessee Valley Authority submitted event report BFAO-50-260/741W identifying 11 of 14 INC Model MSVA-1A snubbers that were found inoperable on Browns Ferry Nuclear Power Station Unit 2 and subsequently identified 5 of 14 inoperable units on Browns Ferry Nuclear Power Station Unit No 3. All of these units were found to be frozen, and the cause was attributed to a failure to lubricate the parts during assembly. The failed snubbers were replaced with new units produced by the same manufacturer.

2. On April 12, 1976, the St. Lucie Plant Unit 1 facility of Florida Power and Light Corporation submitted event report No. 50-335-76-9 wherein five INC Model MSVA-1 snubbers were identified as inoperable because they were found to be frozen. The failures were caused by oxidation on the internals and by improper assembly. All INC mechanical snubbers were replaced with units produced by another manufacturer.

3. On April 8, 1977, Iowa Electric Light and Power Company submitted event report No. 77-23 for the Duane Arnold Energy Center facility that identified 13 INC Model 1MSVA-1 Type AS snubbers to be frozen; the cause of failure was attributed to large amounts of interior oxidation. The units were replaced with those produced by another manufacturer.

4. On December 5, 1979, personnel from the Nuclear Regulatory Commission visited Department of Energy (DOE) facilities at Richland, Washington, to obtain information on DOE experience with INC snubbers at the Fast Flux Test Facility (FFTF). The DOE-owned FFTF was equipped with more than 4,000 mechanical pipe restraints (snubbers) supplied by INC. In 1978, FFTF examined more than 800 of these mechanical snubbers by removing them from their installation and found that 43, or about 5% of those examined, were frozen. The plant was still under construction so the snubbers had seen no service and had been subjected to only normal construction environments for 1 to 2 years.

Tests were conducted on three operable snubbers by installing them on a Hanford Engineering and Development Laboratory (HEDL) process line. The three snubbers were subjected to flow-induced low-amplitude vibration (0.003 inches or less). These snubbers were of both the combined carbon steel and stainless steel construction and the all stainless steel construction. Detailed test data are not available to the NRC at this time. However, all three snubbers froze after being subjected to the vibration for periods of 3 to 30 days.

The failure modes on all units inspected and tested involved a number of different mechanisms leading to the freezing of the snubbers. Following disassembly of some of the snubbers, inspections showed the failures were caused by improper assembly; overheating of internal components caused by welding (during fabrication); and sensitivity of the design to dirt, corrosion, and inadequate or excessive lubrication. DOE concluded that there were generic deficiencies in the design of the snubbers of this specific manufacturer for application to the FFTF facility and for pipes subjected to vibration. All INC mechanical snubbers in FFTF have been replaced with snubbers produced by another manufacturer.

5. On May 31, 1980, Georgia Power Company reported eight INC snubbers located on instrument and drain lines at Edwin I. Hatch Nuclear Plant Unit 1 were identified as inoperable (LER 321-80-55). The cause of the failures was identified as internal corrosion that caused a frozen condition. In an attempt to free a snubber (750-pound capacity), forces up to 1500 pounds were applied in both the "extend" and "retract" directions and the snubber did not move. The inspection of INC snubbers was completed at the Hatch facility and, on June 30, 1980, NRC received a supplemental report that 45 of the 61 snubbers that had been inspected on Unit 1 had been identified as inoperable and three of the 42 snubbers that were inspected on Unit 2 were inoperable. All inoperable snubbers were replaced prior to startup of the affected unit. Some were replaced with mechanical units produced by another manufacturer, some were replaced with later-model INC snubbers, and three were replaced with rigid restraints. Plans are being made to replace all INC snubbers during upcoming refueling outages. Analyses are also being performed on the piping affected by the locked up snubbers.

In addition to INC snubber failures, failures of mechanical snubbers by another manufacturer are identified below:

1. On September 7, 1979, Public Service Electric and Gas Company reported the failure of three Model PSA-3 mechanical snubbers manufactured by Pacific Scientific Company that were located on a main feedwater line of Salem Nuclear Generating Station Unit 1 (LER 79-54). These three snubbers could not be rotated around their spherical rod end bearings. The snubbers were removed and inspection revealed that the lead screw and traveling nut assembly, which translates linear to rotational motion, had failed. The snubbers no longer provided seismic shock restraint under this condition. These snubbers are directly upstream of the nuclear Class II piping boundary and are included in the stress calculations for the seismic analysis of the nuclear portion of the main feedwater piping. Failure of the snubbers

appeared to result from a force many times greater than the design load of the snubbers. This force was either an extreme shock load or occurred when the snubber was in the fully retracted condition. The snubbers were replaced with units produced by the same manufacturer.

2. On April 10, 1979, Consumers Power Company reported a failure of eight Model PSA-3 Pacific Scientific snubbers at their Big Rock Point Nuclear Plant facility (LER 79-017/03L-0). The cause of the failure was improper installation in that a spherical washer was omitted from the transition tube.

3. On March 15 and June 11, 1979, Florida Power and Light reported failures of Pacific Scientific Company mechanical snubbers on main steam and feedwater systems at Turkey Point Plant Units 3 and 4 (LER 79-006/03L-0 and 79-009/03L-0 respectively). The cause in both cases was attributed to excessive loading.

The nature of the above mechanical snubber failures is to prevent the piping systems, to which they are attached, from moving freely during the normal thermal heat up and cool down associated with plant operations. Restraining this thermal motion results in higher than normal stresses which, if high enough and repeated frequently enough, can lead to a premature fatigue failure of the piping system.

These mechanical snubbers have been installed for a number of years without any NRC requirements for periodic surveillance to determine their condition. As a result, their current condition is unknown to NRC and therefore NRC is requesting a prompt examination of all mechanical snubbers installed to date. Because of the high percentage of failures discovered with the INC snubbers, the time frame for their examination is the shortest and additional operability tests are called for.

Actions to be Taken by Licensees of Operating Reactors:

1. Within 30 days of the issuance date of this bulletin, all normally accessible* INC mechanical snubbers installed on safety-related systems or in storage shall be visually examined and tested as follows:

   a. Perform a visual examination for damage and, without causing the system to be inoperable except as permitted by the facility technical specifications, verify that the snubbers have freedom of movement by performing a manual test over the range of the stroke in both compression and tension.

   b. Perform an operability test to confirm that (1) activation (restraining action) occurs in both compression and tension and (2) the drag forces are within the specified range in both compression and tension. The tests shall be performed on all snubbers in storage and on a representative sample (10% of the total of this type of snubber in use in the plant or 35, which ever is less) of the

*"Normally accessible" refers to those areas of the plant that can be entered during reactor operation.

normally accessible snubbers that are in service and can be individually removed without causing the system to be inoperable, except as permitted by the facility technical specifications. For each snubber that does not meet the test acceptance criteria, an additional representative sample (as defined above) of this type of snubber shall be tested. For each of these additional snubbers that do not meet the test acceptance criteria, another representative sample of this type of snubber shall be tested. This cycle shall be repeated until no more failures have been found or until all snubbers of this type have been tested. The samples should be made up of snubbers representing the various sizes.

c. Snubbers which have been examined and tested in a manner comparable to Items 1a and 1b above within the last six months may be exempted.

d. If any failures are identified in Items 1a or 1b above, take corrective action and evaluate the effect of the failure on the system operability pursuant to the facility technical specifications for continued operation.

e. If failures are identified in Items 1a and 1b above, and if INC snubbers are known to be located in any inaccessible areas, a plant shutdown shall be performed within 30 days after the discovery of the first inoperable snubber and inspections conducted in accordance with Item 2a and 2b below, unless justification for continued operation has been provided to the NRC.

2. Visually examine and test all inaccessible INC mechanical snubbers installed on safety related systems at the next outage of greater than five days duration as follows:

   a. Visually examine and manually test all inaccessible snubbers as described in Item 1a above.

   b. Perform an operability test on a representative sample of inaccessible snubbers as described in Item 1b above.

   c. Snubbers which have been examined and tested in a manner comparable to Items 2a and 2b above within the last six months may be exempted.

   d. If any failures are identified in Items 2a or 2b above, take corrective action to evaluate the effect of the failure on system operability pursuant to the facility technical specifications for resuming operation.

3. Provide a schedule for an inspection program covering mechanical snubbers produced by other manufactures. As a minimum, this inspection program shall:

   a. Include all snubbers installed on safety-related systems;

   b. Include the visual examination and manual test described in Item 1a above for all snubbers;

c. Snubbers which have been examined and tested in a manner comparable to Item 3b above within the last twelve months may be exempted:

d. Require the corrective action and evaluations described in Items 1d and 2d above; and

e. Be completed prior to the completion of the next refueling outage. Plants which are currently in a refueling outage should perform the visual examination and manual tests of inaccessible mechanical snubbers before resumption of operations unless some other basis for assurance of snubber operability is provided to the NRC.

4. Submit a report of the results of the inspections, testing and evaluation requested in Item 1 to NRC within 45 days of the issuance date of this bulletin. Report the results of the inspections, testing and evaluation requested in Item 2 within 30 days after the inspection and testing have been completed. The response to Item 3 shall be submitted within 60 days of the issuance date of this Bulletin. The results of the inspections performed for Item 3 shall be submitted within 60 days after the completion of the inspection.

The reports shall contain the following:

a. A description of the visual examinations and tests performed.

b. Number of snubbers examined and tested. Grouping by manufacturer name, model number, and size is acceptable.

c. Number of failures identified; manufacturer name, model number, size, mode of failure, cause of failure, corrective action, snubber location, effect of failure on plant and system safety, and justification for continuing or resuming operation.

d. The above information shall also be provided for the snubbers exempted by Items 1c, 2c, and 3c above.

Actions to be Taken by the Following Licensees Holding Construction Permits:

Diablo Canyon Nuclear Power Plant Unit 1; San Onofre Nuclear Station Unit 2; Watts Bar Nuclear Plant Units 1 and 2; and Virgil C. Summer Nuclear Station Unit 1.

1. After preoperational and/or hot functional testing and preceding fuel loading, visually examine and test the mechanical snubbers installed on safety-related systems as follows:

a. For all snubbers perform a visual examination for damage and verify that the snubbers have freedom of movement by performing a manual test over the range of the stroke in both compression and tension.

b. For INC snubbers, perform an operability test to confirm that (1) activation (restraining action) occurs in both compression and tension and (2) the drag forces are within the specified range in both compression and tension. The tests shall be performed on a representative sample (10% of the total of this type of snubber in use in the plant or 35, which ever is less). For each snubber that does not meet the test acceptance criteria, an additional representative sample (as defined above) of this type of snubber shall be tested. For each of these additional snubbers that do not meet the test acceptance criteria, another representative sample of this type of snubber shall be tested. This cycle shall be repeated until no more failures have been found or until all snubbers of this type have been tested. The samples should be made up of snubbers that represent the various sizes.

c. If any failures are identified in Items a or b above, take corrective action prior to fuel loading.

2. The schedule for the inspections and tests requested in Item 1 above, shall be submitted within 60 days of the issuance date of this bulletin. The results of the inspections, testing, and evaluation requested in Item 1 shall be reported to NRC within 30 days after the inspection and testing have been completed.

   The reports shall contain the following:

   a. A description of the visual examinations and tests performed.

   b. Number of snubbers examined and tested. Grouping by manufacturer name, model number, and size is acceptable.

   c. Number of failures identified; manufacturer name, model number, size, mode of failure, cause of failure, corrective action, and snubber location.

Reports, signed under oath or affirmation, under the provisions of Section 182a of the Atomic Energy Act of 1954, shall be submitted to the Director of the appropriate NRC Regional Office and a copy shall be forwarded to the Director of the NRC Office of Inspection and Enforcement, Washington, D. C. 20555.

If you desire additional information regarding this matter, please contact the IE Regional Office.

Approved by GAO B-180225 (S81003) expires December 31, 1981.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 0 5 1981

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: J. H. Sniezek, Director, Division of Resident and Regional Reactor Inspection, IE

SUBJECT: IE BULLETIN NO. 81-01, Revision 1: SURVEILLANCE OF MECHANICAL SNUBBERS

Pages 5 and 6 of IE Bulletin 81-01 have been revised to change the identification of the power reactor facilities with construction permits to which the bulletin is applicable. This revision is prompted by the fact that the Office of Nuclear Reactor Regulation generic letter concerning snubber preoperational testing received a slightly different distribution than originally envisioned by IE.

A copy of the bulletin, with revised pages 5 and 6, should be issued for action to the power reactor facilities with construction permits that have been added to the list of those requiring action on March 4, 1981. Also, a copy of revised pages 5 and 6 should be issued to the power reactor facilities with construction permits that have been deleted from the list of those requiring action, as formal notification of their revised status. Finally, a copy of revised pages 5 and 6 should be transmitted for information to all other power reactor facilities with construction permits and all facilities with operating permits.

A copy of the bulletin with revised pages 5 and 6 and draft transmittal letters to licensees and permit holders are also enclosed.

This revision was discussed with your cognizant staff members during early and mid-February 1981. At that time they were also requested to informally alert the affected power reactor facilities of their change in status.

J H Sniezek
James H. Sniezek, Director
Division of Resident and Regional
Reactor Inspection, IE

CONTACT: R. J. Kiessel
49-27551

Enclosures:
1. Draft transmittal letter for action
2. Draft transmittal letter for deletion of action
3. Draft transmittal letter for information
4. IE Bulletin No. 81-01 with revised pages 5 and 6

c. Snubbers which have been examined and tested in a manner comparable to Item 3b above within the last twelve months may be exempted:

d. Require the corrective action and evaluations described in Items 1d and 2d above; and

e. Be completed prior to the completion of the next refueling outage. Plants which are currently in a refueling outage should perform the visual examination and manual tests of inaccessible mechanical snubbers before resumption of operations unless some other basis for assurance of snubber operability is provided to the NRC.

4. Submit a report of the results of the inspections, testing and evaluation requested in Item 1 to NRC within 45 days of the issuance date of this bulletin. Report the results of the inspections, testing and evaluation requested in Item 2 within 30 days after the inspection and testing have been completed. The response to Item 3 shall be submitted within 60 days of the issuance date of this Bulletin. The results of the inspections performed for Item 3 shall be submitted within 60 days after the completion of the inspection.

The reports shall contain the following:

a. A description of the visual examinations and tests performed.

b. Number of snubbers examined and tested. Grouping by manufacturer name, model number, and size is acceptable.

c. Number of failures identified; manufacturer name, model number, size, mode of failure, cause of failure, corrective action, snubber location, effect of failure on plant and system safety, and justification for continuing or resuming operation.

d. The above information shall also be provided for the snubbers exempted by Items 1c, 2c, and 3c above.

<u>Actions to be Taken by the Following Licensees Holding Construction Permits</u>:

Diablo Canyon Nuclear Power Plant Units 1 and 2; Grand Gulf Nuclear Station, R1
Unit 1; LaSalle County Station, Unit 1; Virgil C. Summer Nuclear Station, R1
Unit 1; and Susquehanna Steam Electric Station, Unit 1 shall perform the R1
following: R1

1. After preoperational and/or hot functional testing and preceding fuel loading, visually examine and test the mechanical snubbers installed on safety-related systems as follows:

   a. For all snubbers perform a visual examination for damage and verify that the snubbers have freedom of movement by performing a manual test over the range of the stroke in both compression and tension.

b. For INC snubbers, perform an operability test to confirm that (1) activation (restraining action) occurs in both compression and tension and (2) the drag forces are within the specified range in both compression and tension. The tests shall be performed on a representative sample (10% of the total of this type of snubber in use in the plant or 35, which ever is less). For each snubber that does not meet the test acceptance criteria, an additional representative sample (as defined above) of this type of snubber shall be tested. For each of these additional snubbers that do not meet the test acceptance criteria, another representative sample of this type of snubber shall be tested. This cycle shall be repeated until no more failures have been found or until all snubbers of this type have been tested. The samples should be made up of snubbers that represent the various sizes.

c. If any failures are identified in Items a or b above, take corrective action prior to fuel loading.

2. The schedule for the inspections and tests requested in Item 1 above, shall be submitted within 60 days of the issuance date of this bulletin.* The R1
results of the inspections, testing, and evaluation requested in Item 1 shall be reported to NRC within 30 days after the inspection and testing have been completed.

The reports shall contain the following:

a. A description of the visual examinations and tests performed.

b. Number of snubbers examined and tested. Grouping by manufacturer name, model number, and size is acceptable.

c. Number of failures identified; manufacturer name, model number, size, mode of failure, cause of failure, corrective action, and snubber location.

Reports, signed under oath or affirmation, under the provisions of Section 182a of the Atomic Energy Act of 1954, shall be submitted to the Director of the appropriate NRC Regional Office and a copy shall be forwarded to the Director of the NRC Office of Inspection and Enforcement, Washington, D. C. 20555.

If you desire additional information regarding this matter, please contact the IE Regional Office.

Approved by GAO B-180225 (S81003) expires December 31, 1981.

*The "issurance date of this bulletin" shall be considered to be the date R1
of issuance of revision 1 for the following licensees holding construction R1
permits: Diablo Cnayon Nuclear Power Plant, Unit 2; Grand Gulf Nuclear R1
Station, Unit 1; LaSalle County Station, Unit 1; and Susquehanna Steam R1
Electric Station, Unit 1. R1

INSPECTION REQUIREMENTS TO REVIEW LICENSEE ACTIONS TAKEN IN RESPONSE TO IE BULLETIN 81-01, SURVEILLANCE OF MECHANICAL SNUBBERS

I. OBJECTIVE

Verify that the licensees have taken the action required by IE Bulletin 81-01 for the surveillance of mechanical snubbers.

II. BACKGROUND

IE Bulletin 81-01 was issued on January 27, 1981 (copy is enclosed for reference). The generic implications of defects, at several facilities, of the mechanical snubbers manufactured by the International Nuclear Safeguards Corporation (INC) and Pacific Scientific Company cause us to be concerned about the adequacy of mechanical snubbers at other facilities. Because of the large number of INC mechanical snubber defects, they are singled out for immediate action by the bulletin. Mechanical snubbers produced by other manufacturers are addressed on a less immediate basis.

This concern is shared by the Office of Nuclear Reactor Regulation which has prepared a revision to the technical specifications addressing mechanical snubber surveillance (copy is enclosed for reference).

The bulletin requirements follow those of the revised technical specifications, and therefore can be viewed as early implementation of the program. The major difference is that modified functional testing need only be performed on the INC snubbers. Requiring this testing on all snubbers could cause a lengthy delay while new testing equipment is purchased.

The Office of Nuclear Reactor Regulation is also modifying its licensing review to require preservice inspection and testing of snubbers. However, the new licensing criteria will only be applicable to applicants who have not commenced pre-operational testing. As such, the new licensing criteria will not be applicable to the near term operating license plants. Hence the need for including these plants in the bulletin. Copies of the new licensing criteria were forwarded to the Regional Reactor Operations and Nuclear Support Branches on October 21, 1980 (copy is enclosed for reference). The bulletin requirements follow those of the preservice portion of the new licensing criteria.

III. INSPECTION REQUIREMENTS

1. Review the responses from licensees and permit holders to ensure that the examinations and tests are being completed, or are scheduled for completion within the time limits set in the bulletin.

2. Review the responses to determine which of the reported failures are generic in nature. A generic mechanical snubber failure is one resulting from a potential defect in manufacturing or design that gives cause to suspect other snubbers. This includes generic failure of any snubbers that fail to sustain the environment or application for which they are designed. The information on generic failures shall be forwarded to the Reactor Engineering Branch, Division of Resident and Regional Reactor Inspection.

3. Requests for delays in implementation of the bulletin due to a lack of testing equipment should be considered on the basis of individual mechanical snubber size and type rather than on a blanket approval for all sizes and types. For example, if equipment for activation testing of a particular large-sized mechanical snubber is not immediately available, this does not mean that small-sized mechanical snubbers need not be activation tested. Similarly, other examinations and tests, such as determining the range of stroke or the drag forces, should not be precluded for those large-sized mechanical snubbers.

4. Portions of the bulletin that are duplicated by the new technical specifications concerning mechanical snubbers need not be performed if the licensee or permit holder has committed to implement these new technical specifications and the applicable examination and/or test will be performed within the time limits set by the bulletin.

5. Normally accessible refers to those areas of the plant which can be entered during reactor operation. No specific radiological criteria are contained in or implied in the bulletin. Rather, the criteria should be whether or not plant personnel routinely enter the area for inspections and/or maintenance during normal reactor operations.

6. In addition to the above, the inspection and review for this bulletin should also satisfy the requirements set forth in MC 92703.

IV. REPORTING REQUIREMENTS

1. Upon receipt and evaluation of all of the inspection schedules required by Item 3 of the bulletin, a report indicating the current status of each plant with respect to completion of the bulletin requirements and the licensees' and permit holders' stated corrective actions shall be submitted to the Reactor Engineering Branch, Division of Resident and Regional Reactor Inspection. Plants requesting delays in implementation (item III.3 above), continued operation or resumption of operation, and/or exemption from portions of the bulletin (item III.4 above) should be specifically identified along with the resolution of the request.

2. The results of any inspections conducted in accordance with MC 92703 (item III.6 above) shall be included in the routine inspection report(s) and compiled in the above report.

3. Compile the licensees' estimates of manpower expended in response to this bulletin and include in the above report.

V. EXPIRATION

This technical instruction shall remain in effect until December 31, 1982.

VI. IE HEADQUARTERS CONTACT

Questions concerning this technical instruction should be directed to: R. J. Kiessel (492-7551).

VII. MODULE TRACKING SYSTEM INPUT (766 DATA)

For Module tracking system input, record actual inspection time agains Module No. 25549

Enclosures:

1. IE Bulletin No. 81-01
2. Letter from D. G. Eisenhut to All Power Reactor Licensees (Except SEP Licensees) dated November 20, 1980, w/encls.
3. Letter from E. L. Jordan, IE, to RO&NS Branches dated October 21, 1980, w/encls.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Enclosure 2 to TI 2515/49

November 20, 1980

TO ALL POWER REACTOR LICENSEES (EXCEPT SEP LICENSEES)

RE: TECHNICAL SPECIFICATION REVISIONS FOR SNUBBER SURVEILLANCE

Enclosed is a copy of Revision 1 of the Inservice Surveillance Requirements for snubbers under the Standard Technical Specifications. This revision embodies several changes, clarifications, and improvements from the previous version based on recent operating experience. This revision also includes surveillance requirements for mechanical snubbers.

We request that you submit a license amendment application to incorporate the applicable portion of these model Technical Specifications into your Appendix "A" Technical Specification within the next 120 days.

In case you have questions, please contact H. Shaw, 492-7364.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Standard Technical Specifications Snubber
Surveillance Requirements

cc w/encl. for information:
See Service List
SEP Licensees

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers listed in Tables 3.7-4a and 3.7-4b shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after four months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7-4a and 3.7-4b. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.9.d or 4.7.9.e, as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.9.d or 4.7.9.e, an additional 10% of that type of snubber shall be functionally tested).
or
(that number of snubbers which follows the expression 35 $[1 + \frac{c}{2}]$,
where c* is the allowable number of snubbers not meeting the

*The value c will be arbitrarily chosen by the applicant and incorporated into the expressions for the representative sample and for the resample prior to the issuance of the Technical Specifications. The expressions are intended for use in plants with larger numbers of safety-related snubbers (>500) and provide a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptable limits. That is, the confidence level will be provided no matter what value is chosen for c. It is advised, however, that discretion be used when initially choosing the value for c because the lower the value of c (the lower the amount of snubbers in the representative sample), the higher the amount of snubbers required in the re-sample will be. To illustrate: If c = 2 and 3 snubbers are found not to meet the functional test acceptance criteria, there will be 70 snubbers in the representative sample and 31 snubbers required for testing in the re-sample; If c = 2 and 4 snubbers fail the functional test, there will be 70 snubbers in the representative sample and 62 snubbers required for testing in the re-sample; If c = 0 and 1 snubber fails the functional test, there will be 35 snubbers in the representative sample and 140 snubbers required for testing in the re-sample; If c = 0 and 2 snubbers fail the functions test, there will be 35 snubbers in the representative sample and 280 snubbers required for testing in the re-sample.

acceptance criteria selected by the operator, shall be functionally tested either in-place or in a bench test. For each number of snubbers above c which does not meet the functional test acceptance criteria of Specifications 4.7.9.d. or 4.7.9.e, an additional sample selected according to the expression $35 (1 + \frac{c}{2}) (\frac{2}{c + 1})^2 (a - c)$ shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the representative sample.

Functional testing shall continue according to the expression $b [35(1 + \frac{c}{2}) (\frac{2}{c + 1})^2]$ where b is the number of snubbers found inoperable in the previous re-sample, until no additional inoperable snubbers are found within a sample or until all snubbers in Table 3.7-4a and 3.7-4b have been functionally tested).

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)

3. Snubbers within 10 feet of the discharge from a safety relief valve

Snubbers identified in Tables 3.7-4a and 3.7-4b as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.* Tables 3.7-4a and 3.7-4b may be used jointly or separately as the basis for the sampling plan.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. Drag force shall not have increased more than 50% since the last functional test.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Tables 3.7-4a and 3.7-4b shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

TABLE 3.7-4a

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4a provided that a revision to Table 3.7-4a is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4a is included with the next License Amendment request.

TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4b is included with the next License Amendment request.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample according to the expression $35 (1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the (URG) and the (CNRAG).

l. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

Enclosure 3 to TI 2515/49

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SSINS #6900

OCT 21 1980

MEMORANDUM FOR: E. J. Brunner, Chief, RO&NSB, RI
R. C. Lewis, Acting Chief, RO&NSB, RII
R. F. Heishman, Chief, RO&NSB, RIII
G. L. Madsen, Chief, RO&NSB, RIV
J. L. Crews, Chief, RO&NSB, RV

FROM: E. L. Jordan, Assistant Director for Technical Programs, Division of Reactor Operations Inspection, IE

SUBJECT: REQUEST FOR PRESERVICE INSPECTION AND TESTING OF SNUBBERS

The enclosed memorandum is forwarded for your information.

E L Jordan

Edward L. Jordan, Assistant Director
for Technical Programs
Division of Reactor Operations Inspection

Enclosure: Memo J. P. Knight to
R. L. Tedesco dtd 9/25/80

CONTACT: ~~L.~~ H. A. Wilber, IE
49-28180

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 2 5 1980

MEMORANDUM FOR: R. L. Tedesco, Assistant Director for Licensing, DL

FROM: J. P. Knight, Assistant Director for Components and Structures Engineering, DE

SUBJECT: PRESERVICE INSPECTION AND TESTING OF SNUBBERS

Reference: (a) Memorandum from R. L. Tedesco, DL dated August 29, 1980

The Office of Inspection and Enforcement has requested that preservice inspection and test requirements for snubbers be included in the licensing process. Based on this request and the long history of snubber problems as documented by Licensee Event Reports on the subject of inoperable and incorrectly installed snubbers, it is requested that the enclosed requirements to insure snubber operability be sent to all applicants who have not commenced pre-operational testing.

J. P. Knight, Assistant Director
for Components & Structures Engineering
Division of Engineering

cc: R. Vollmer, DE
D. Eisenhut, DL
E. Jordan, IE
J. Collins, IE
W. Mills, IE
V. Noonan, DE
S. Pawlicki, DE
B. Youngblood, DL
C. Moon, DL
C. Stahle, DL
D. Sells, DL
J. Wilson, DL
M. D. Lunch
H. Rood
D. Hood
A. Bournia
S. Wurwell
W. Kane
L. Kintner
R. Stark
J. Martore
J. Kerrigan

Contact: E. Hemminger, DE:MEB, x29480

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9 This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a mimimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifictions.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1,4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

APPENDIX B

Documentation of Bulletin Closeout

TABLE B.1 BULLETIN CLOSEOUT STATUS

Facility	Utility	Docket Number	Facility Status	NRC Region	Utility Response Date	Closeout Status and Criterion	Note
Arkansas 1	AP&L	50-313	OL	IV	03-26-81	Closed 3	1
Arkansas 2	AP&L	50-368	OL	IV	03-26-81	Closed 3	1
					08-18-81		
Beaver Valley 1	DLC	50-334	OL	I	03-27-81	Closed 3	1
					03-31-81		
					09-13-82		
Big Rock Point 1	CPC	50-155	OL	III	03-13-81	Closed 3	1
					04-28-82		
					09-12-83		
Browns Ferry 1	TVA	50-259	OL	II	03-13-81	Closed 3	1
					07-23-81		
					02-10-82		
					02-16-83		
Browns Ferry 2	TVA	50-260	OL	II	03-13-81	Closed 3	1
					07-23-81		
					02-10-82		
					02-16-83		
Browns Ferry 3	TVA	50-296	OL	II	03-13-81	Closed 3	1
					07-23-81		
					02-10-82		
					02-16-83		
Brunswick 1	CP&L	50-325	OL	II	02-18-81	Closed 1	1
Brunswick 2	CP&L	50-324	OL	II	02-18-81	Closed 1	1
Calvert Cliffs 1	BG&E	50-317	OL	I	02-19-81	Closed 1	1
Calvert Cliffs 2	BG&E	50-318	OL	I	02-19-81	Closed 1	1
Cook 1	IMECO	50-315	OL	III	03-12-81	Closed 1	1
Cook 2	IMECO	50-316	OL	III	03-12-81	Closed 1	1
Cooper Station	NPPD	50-298	OL	IV	03-06-81	Closed 3	1
					04-02-81		
					06-01-81		
					08-03-81		
					01-19-82		
Crystal River 3	FP	50-302	OL	II	02-19-81	Closed 1	1

See notes at end of table.

TABLE B.1 (contd.)

Facility	Utility	Docket Number	Facility Status	NRC Region	Utility Response Date	Closeout Status and Criterion	Note
Davis-Besse 1	TECO	50-346	OL	III	03-06-81 09-14-82	Closed 3	1
Diablo Canyon 1	PG&E	50-275	OL	V	04-03-81 07-21-81 09-08-81 10-20-81	Closed 3	2
Diablo Canyon 2	PG&E	50-323	CP	V	05-05-81 05-14-82 02-15-85 05-28-85	Open	2
Dresden 2	CECO	50-237	OL	III	03-02-81 03-27-81 06-09-81	Closed 3	1
Dresden 3	CECO	50-249	OL	III	03-02-81 03-27-81 06-15-82	Closed 3	1
Duane Arnold	IELPCO	50-331	OL	III	03-13-81 06-01-81 11-10-83	Open	1
Farley 1	APCO	50-348	OL	II	03-30-81 04-06-81	Closed 3	1
Farley 2	APCO	50-364	OL	II	03-30-81 04-06-81	Closed 3	1
FitzPatrick	PASNY	50-333	OL	I	03-11-81 05-28-82	Closed 3	1
Fort Calhoun 1	OPPD	50-285	OL	IV	02-27-81 03-27-81 11-27-81 12-08-81	Closed 3	1
Fort St. Vrain	PSCC	50-267	OL	IV	03-13-81	Closed 3	1
Ginna	RG&E	50-244	OL	I	03-17-81 06-25-81	Closed 3	1
Grand Gulf 1	MP&L	50-416	OL	II	05-01-81 10-27-81 04-21-82 06-04-82 08-10-84 09-25-84	Closed 3	2

See notes at end of table.

TABLE B.1 (contd.)

Facility	Utility	Docket Number	Facility Status	NRC Region	Utility Response Date	Closeout Status and Criterion	Note
Haddam Neck	CYAPCO	50-213	OL	I	02-19-81 02-24-81 03-13-81 08-11-81 01-08-82	Closed 3	1
Hatch 1	GP	50-321	OL	II	03-13-81 07-31-81	Closed 3	1
Hatch 2	GP	50-366	OL	II	03-13-81 07-31-81	Closed 3	1
Indian Point 2	ConEd	50-247	OL	I	03-16-81	Closed 1	1
Indian Point 3	PASNY	50-286	OL	I	02-11-81	Closed 1	1
Kewaunee	WPS	50-305	OL	III	03-13-81	Closed 1	1
LaCrosse	DPC	50-409	OL	III	02-27-81	Closed 3	1
LaSalle 1	CECO	50-373	OL	III	04-30-81 05-17-82	Closed 3	2
Maine Yankee	MYAPCO	50-309	OL	I	03-06-81	Closed 1	1
McGuire 1	DUPCO	50-369	OL	II	03-27-81 06-01-81	Closed 3	1
Millstone 1	NNECO	50-245	OL	I	02-19-81 02-24-81 04-03-81 08-11-81	Closed 3	1
Millstone 2	NNECO	50-336	OL	I	02-19-81 02-24-81 03-11-81 03-27-81 04-16-81 04-27-81 05-11-81 05-18-81 06-22-81 04-07-82	Closed 3	1
Monticello	NSP	50-263	OL	III	02-04-81 03-09-81	Closed 1	1
Nine Mile Point 1	NMP	50-220	OL	I	03-24-81 06-24-81	Closed 3	1

See notes at end of table.

TABLE B.1 (contd.)

Facility	Utility	Docket Number	Facility Status	NRC Region	Utility Response Date	Closeout Status and Criterion	Note
North Anna 1	VEPCO	50-338	OL	II	03-26-81 01-03-83	Closed 3	1
North Anna 2	VEPCO	50-339	OL	II	03-26-81 01-03-83	Closed 1	1
Oconee 1	DUPCO	50-269	OL	II	03-20-81 06-22-81 01-10-84	Closed 3	1
Oconee 2	DUPCO	50-270	OL	II	03-20-81 06-22-81 01-10-84	Closed 3	1
Oconee 3	DUPCO	50-287	OL	II	03-20-81 06-22-81 01-10-84	Closed 3	1
Oyster Creek 1	JCP&L	50-219	OL	I	03-30-81 03-01-85	Closed 3	1
Palisades	CPC	50-255	OL	III	03-19-81 12-15-81	Closed 3	1
Peach Bottom 2	PECO	50-277	OL	I	03-25-81 10-27-81 04-22-82 10-04-82	Closed 3	1
Peach Bottom 3	PECO	50-278	OL	I	03-25-81 10-27-81	Closed 3	1
Pilgrim 1	BECO	50-293	OL	I	03-13-81 08-11-81 06-10-82	Closed 3	1
Point Beach 1	WEPCO	50-266	OL	III	02-12-81	Closed 1	1
Point Beach 2	WEPCO	50-301	OL	III	02-12-81	Closed 1	1
Prairie Island 1	NSP	50-282	OL	III	02-06-81	Closed 1	1
Prairie Island 2	NSP	50-306	OL	III	02-06-81	Closed 1	1
Quad Cities 1	CECO	50-254	OL	III	03-02-81 03-27-81 06-15-82	Closed 3	1
Quad Cities 2	CECO	50-265	OL	III	03-02-81 03-27-81 01-14-82 06-15-82	Closed 3	1

See notes at end of table.

TABLE B.1 (contd.)

Facility	Utility	Docket Number	Facility Status	NRC Region	Utility Response Date	Closeout Status and Criterion	Note
Rancho Seco 1	SMUD	50-312	OL	V	03-06-81 08-10-82	Closed 1	1
Robinson 2	CP&L	50-261	OL	II	02-18-81	Closed 1	1,7
Salem 1	PSE&G	50-272	OL	I	03-12-81 07-17-81	Closed 3	1
Salem 2	PSE&G	50-311	OL	I	03-12-81 07-17-81	Closed 3	1
San Onofre 1	SCE	50-206	OL	V	02-27-81 02-29-82 06-10-82 08-12-82(2)	Closed 3	1
Sequoyah 1	TVA	50-327	OL	II	03-13-81 03-31-81 05-01-81	Closed 3	1
St. Lucie 1	FPL	50-335	OL	II	03-27-81 03-10-82	Closed 3	1
Summer 1	SCE&G	50-395	OL	II	03-17-81 08-31-81 02-15-82 06-22-82 03-04-83 04-22-83 06-30-83	Closed 3	2
Surry 1	VEPCO	50-280	OL	II	03-26-81 07-30-81	Closed 3	1
Surry 2	VEPCO	50-281	OL	II	03-26-81 07-01-81	Closed 3	1
Susquehanna 1	PP&L	50-387	OL	I	05-12-81 07-12-82	Closed 3	2
TMI 1	Met-Ed	50-289	OL	I	03-23-81	Closed 1	1
Trojan	PGE	50-344	OL	V	03-30-81 04-23-81 07-20-81	Closed 3	1
Turkey Point 3	FPL	50-250	OL	II	03-31-81 07-06-81	Closed 3	1
Turkey Point 4	FPL	50-251	OL	II	03-31-81 02-19-82	Closed 3	1

See notes at end of table.

TABLE B.1 (contd.)

Facility	Utility	Docket Number	Facility Status	Utility NRC Region	Response Date	Closeout Status and Criterion	Note
Vermont Yankee 1	VYNP	50-271	OL	I	02-10-81	Closed 1	1
Yankee-Rowe 1	YAECO	50-029	OL	I	02-27-81 06-01-81	Closed 3	1
Zion 1	CECO	50-295	OL	III	03-02-81 03-27-81 06-09-81	Open	1
Zion 2	CECO	50-304	OL	III	03-02-81 03-27-81 01-14-82	Open	1

Notes:

1. This facility had an operating license (OL) or was considered to be a "near term operating license" plant when the bulletin was issued and was required to perform bulletin actions 1 through 4 for OLs.
2. This selected facility had a construction permit (CP) when the bulletin was issued and was required to perform bulletin actions 1 and 2 for CPs.
3. Facility status is based on references 1, 2 and 3, Page B-19.
4. The following abbreviations apply to facility status:
   CP, Construction Permit
   OL, Operating License
5. For Bulletin Closeout Criteria, refer to Page 5 of this report.
6. Facilities to which the bulletin was issued for information are listed in Table B.4 on Page B-17.
7. Referring to Page 8 of NRC/IE Inspection Report 50-261/82-27 (9-2-82) for Robinson 2, note that mechanical snubbers were added after the utility response to IEB 81-01 was issued 2-18-81. A licensee action item and an NRC/IE inspection followup item ensure completion of a Technical Specification amendment and development of an inspection program.
8. The bulletin was issued for action to licensees and selected holders of construction permits. Only facilities affected by this action requirement are included in this table.

TABLE B.2 LIST OF NRC/IE INSPECTION REPORTS USED FOR BULLETIN CLOSEOUT PER CRITERION 3

Facility	Report Number	Date of Approval	NRC Region
Arkansas 1	50-313/81-28	10-23-81	IV
Arkansas 2	50-368/81-27	10-23-81	IV
Beaver Valley 1	50-334/81-28	01-07-82	I
	50-334/83-14	08-19-83	
Big Rock Point 1	50-155/84-07	09-13-84	III
Browns Ferry 1	50-259/85-15	See Note 2	II
Browns Ferry 2	50-260/85-15	See Note 2	II
Browns Ferry 3	50-296/85-15	See Note 2	II
Cooper Station	50-298/81-17	09-28-81	IV
	50-298/82-03	02-12-81	
Davis-Besse 1	50-346/83-17	09-14-83	III
Diablo Canyon 1	50-275/83-14	05-09-83	V
	50-275/84-24	10-10-84	
Dresden 2	50-237/84-027	See Note 3	III
Dresden 3	50-249/85-013	See Note 3	III
Farley 1	50-348/82-18	08-02-82	II
Farley 2	50-364/82-17	08-02-82	II
FitzPatrick	50-333/82-12	07-13-82	I
Fort Calhoun 1	50-285/81-14	07-23-81	IV
Fort St. Vrain	50-267/81-04	04-15-81	IV
Ginna	50-244/81-13	08-17-81	I
Grand Gulf 1	50-416/85-06	--------	II
Haddam Neck	50-213/84-03	04-18-84	I
Hatch 1	50-321/81-23	10-16-81	II
Hatch 2	50-366/81-23	10-16-81	II
LaCrosse	50-409/85-004	04-26-85	III
LaSalle 1	50-373/81-29	09-28-81	III
McGuire 1	50-369/84-05	03-23-84	II
Millstone 1	------------	See Note 1	I
Millstone 2	------------	See Note 1	I
Nine Mile Point 1	50-220/81-29	01-15-82	I
North Anna 1	50-338/83-29	11-22-83	II
Oconee 1	50-269/83-24	08-26-83	II
Oconee 2	50-270/83-24	08-26-83	II
Oconee 3	50-287/83-24	08-26-83	II
Oyster Creek 1	50-219/85-06	04-04-85	I
Palisades	50-255/83-25	11-21-83	III
Peach Bottom 2	------------	See Note 1	I
Peach Bottom 3	50-278/81-26	12-10-81	I
Pilgrim 1	50-293/82-23	09-09-82	I
Quad Cities 1	50-254/82-19	12-23-82	III
Quad Cities 2	50-265/82-04	03-08-82	III

TABLE B.2 (contd.)

Facility	Report Number	Date of Approval	NRC Region
Salem 1	-------------	See Note 1	I
Salem 2	50-311/81-13	07-13-81	I
San Onofre 1	50-206/82-05	03-19-82	V
Sequoyah 1	50-327/85-08	03-15-85	II
St. Lucie 1	50-335/82-18	06-08-82	II
Summer 1	50-395/85-09	03-11-85	II
Surry 1	50-280/84-03	02-17-84	II
Surry 2	50-281/84-03	02-17-84	II
Susquehanna 1	50-387/82-19	08-12-82	I
Trojan	50-344/82-24	09-02-82	V
Turkey Point 3	50-250/84-15	05-31-84	II
Turkey Point 4	50-251/84-15	05-31-84	II
Yankee-Rowe 1	50-029/81-06	07-09-81	I

Notes:
1. Refer to memorandum dated 3-14-85 for E. L. Jordan (NRC/IE HQ) from R. W. Starostecki (RI) and to memorandum dated 4-29-85 (with enclosures 1 and 2) for H. Kister, S. Collins and E. Wennzinger (Project Branches 1, 2 and 3) from J. Durr (DRS) indicating that closeout had been approved.
2. Refer to memorandum dated 4-9-85 for E. L. Jordan (NRC/IE HQ) from R. D. Walker (RII) indicating that the inspection reports noted are to be issued for closeout.
3. Refer to memorandum dated 4-19-85 for E. L. Jordan (NRC/IE HQ) from C. E. Norelius (RIII) indicating that the inspection reports noted are to be issued for closeout.
4. All of the facilities in Table B.2 have operating licenses.
5. Verification that LaCrosse had no mechanical snubbers is documented in the inspection report noted. The utility response was incomplete. No corrective action was required.

TABLE B.3 IDENTIFICATION OF AFFECTED SNUBBERS AND EVALUATION OF FAILURES

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Arkansas 1	PSC	PSA-1/2	0.65	4	0			1,3
	PSC	PSA-1	1.50	1	0			
	PSC	PSA-3	6	4	2	Locked	Immersion	
	PSC	PSA-35	50	18	0			
The failed snubbers were removed from the RC pump P34B suction drain line and replaced with new PSA-3s. Even if this normally isolated 1-1/2" line were to break, the plant would be safe.								
Arkansas 2	PSC	PSA-1/2	0.65	61	15	Locked	Abuse(10),moisture(5)	1,3
	PSC	PSA-1	1.50	26	0			
	PSC	PSA-3	6	52	0			
	PSC	PSA-10	15	59	0			
	PSC	PSA-35	50	52	0			
	PSC	PSA-100	120	34	0			
The failed snubbers were removed from seven lines and replaced with new PSA-1/2s. Although the bolts of support 2EBC-1-H4 may have experienced above normal thermal loads, their operability limit was not exceeded. One tee may have to be replaced after about 15 years of service.								
Beaver Valley 1	PSC	PSA-3	6	3	0			1,3
	PSC	PSA-10	15	3	0			
Big Rock Point 1	PSC	PSA-1/4	0.35	2	1	Damage	Abuse	1,3
	PSC	PSA-1/2	0.65	3	2	Low Drag		
	PSC	PSA-3	6	11	0			
These snubbers are used or intended for use in the RDS. As noted in IEB 81-01, the cause of failure was omission of a spherical washer from the transition tube.								
Browns Ferry 1	INC	MSVA-1	0.75	14	0			1,3
	PSC	PSA-3	6	1	0			
	PSC	PSA-10	15	52	0			
Browns Ferry 2	INC	MSVA-1	0.75	14	14	Low drag		1,3
	PSC	PSA-10	15	46	1	High breakaway	Overload	
All INCs were replaced with equivalent PSCs. The failed PSA-10 was replaced with a new PSA-10. There was no evidence of damage to the support and piping adjacent to the failed PSA-10.								
Browns Ferry 3	INC	MSVA-1	0.75	14	14	Low drag		1,3
	PSC	PSA-10	15	48	3	High breakaway,low drag		
All INCs were replaced with equivalent PSCs. The failed PSA-10s were replaced with new PSA-10s. There was no evidence of damage to supports and piping adjacent to the failed PSA-10s.								
Cooper Station	PSC	PSA-3	6	3	0			1,3
	PSC	PSA-10	15	74	2	Locked	Overload	
	PSC	PSA-35	50	18	0			
The failed PSA-10s were replaced with new PSA-10s. A water hammer event caused failure of snubber MS-SNUB-SSA3 in the "A" main steam line. A water hammer event probably caused failure of snubber MS-SNUB-SS7A1 in the "A" RR pump discharge line. Procedures were reviewed to mitigate water hammer events. No damage to adjacent piping and supports was observed.								
Davis-Besse 1	PSC	PSA-1/4	0.35	63	9	Locked(3),setting(6)	Twisting,corrosion, installation	1,3
	PSC	PSA-1/2	0.65	4	0			
	PSC	PSA-1	1.50	9	0			
	PSC	PSA-3	6	2	0			
The three locked snubbers were replaced with new PSA-1/4s. The six snubbers with out-of-tolerance piston settings were modified as necessary. Operability of piping and supports was not affected adversely.								

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Diablo Canyon 1	PSC	PSA-1/4	0.35	313	12	Locked(7), high drag(2), damage(3)	Installation, abuse	2,3
	PSC	PSA-1/2	0.65	85	1	Locked	Installation	
	PSC	PSA-1	1.50	127	1	Locked	Installation	
	PSC	PSA-3	6	142	2	Locked	Moisture(1), installation(1)	
	PSC	PSA-10	15	33	0			
	PSC	PSA-35	50	12	0			
	PSC	PSA-100	120	6	0			

The failed snubbers were repaired by replacing damaged parts, except that one of the failed PSA-3s was replaced. The locations of failed snubbers were identified.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Diablo Canyon 2	PSC	PSA-1/4 (NF)*	0.35	23	7	Partial stroke(3), locked(2), restricted(1), rotating barrel(1)	Overload(6), not determined(1)	2,4
	PSC	PSA-1/4 (PRE-NF)	0.25	104				
	PSC	PSA-1/2 (NF)	0.65	7	2	Locked(1), missing snap ring(1)	Overload(1), not determined(1)	
	PSC	PSA-1/2 (PRE-NF)	0.50	46				
	PSC	PSA-1(NF)	1.50	58	0			
	PSC	PSA-1 (PRE-NF)	1.00	14	0			
	PSC	PSA-3(NF)	6	81	1	Damaged bearings	Not determined	
	PSC	PSA-3 (PRE-NF)	3	29				
	PSC	PSA-10(NF)	15	11	2	Locked(2)	Overload(2)	
	PSC	PSA-10 (PRE-NF)	10	32				
	PSC	PSA-35(NF)	50	5	0			
	PSC	PSA-35 (PRE-NF)	50	3	0			
	PSC	PSA-100(NF)	120	4	0			
	PSC	PSA-100 (PRE-NF)	100	1	0			
	A/D	AD-41	0.40	146	2	Restricted	Not determined	
	A/D	AD-71	0.70	72	5	No drag(4), restricted(1)	Overload(4), not determined(1)	
	A/D	AD-151	1.50	75	2	No drag	Overload	
	A/D	AD-501	5	101	1	Restricted	Not determined	
	A/D	AD-1601	16	54	0			
	A/D	AD-5501	55	41	0			
	A/D	AD-12501	125	14	0			

*Nuclear Function

All snubbers with impaired operability were replaced prior to fuel load. The locations of replaced snubbers were identified.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Dresden 2	PSC	PSA-3	6	3	0			1,3
	PSC	PSA-10	15	36	0			
	PSC	PSA-35	50	19	0			
Dresden 3	PSC	PSA-1/4	0.35	6	0			1,3
	PSC	PSA-1/2	0.65	1	0			
	PSC	PSA-1	1.50	7	0			
	PSC	PSA-10	15	29	0			
	PSC	PSA-35	50	19	0			

One snubber failed full load testing and was replaced with a new hydraulic snubber (HSSA-10). Failure probably was caused by one time shock overload. The size of the replaced snubber was not given.

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Duane Arnold	INC	MSVA-1	0.75	6	1	High breakaway	Usage	1,4
	PSC	PSA-1	1.5	58	4	Restricted (3), wrong safety wire(1)	Moisture(1), degraded lubricant(1), unknown(1) installation(1)	
	PSC	PSA-3	6	26	1	Restricted	Overload	
	PSC	PSA-10	15	23	5	Frozen	Dirty bushings	
	PSC	PSA-35	50	23	0			
	PSC	PSA-100	120	2	0			
The MSVA-1 was replaced with a PSA-1. The PSA-10s were repaired by cleaning and lubrication. The restricted PSA-1s and PSA-3s were replaced with the same models. One PSA-1 was repaired by installing the correct size of safety wire.								
Farley 1	PSC	PSA-1/4	0.35	170	8	Locked	Installation, abuse	1,3
	PSC	PSA-1/2	0.65	16	0			
	PSC	PSA-1	1.50	23	0			
	PSC	PSA-3	6	6	0			
The failed snubbers were replaced. No unusual indication of damage to piping systems affected by the failed snubbers was found.								
Farley 2	PSC	PSA-1/4	0.35	129	11	Locked or restricted	Installation, abuse	1,3
	PSC	PSA-1/2	0.65	39	5	Locked or restricted	Installation, abuse	
	PSC	PSA-1	1.50	26	1	Locked	Installation	
	PSC	PSA-3	6	20	0			
	PSC	PSA-10	15	1	0			
The failed snubbers were replaced with spares. The functional testing of snubbers took place during pre-operational test and had no affect on plant or system safety.								
FitzPatrick	There were 16 PSCs. The number of defective snubbers cannot be determined from the responses.							1,3
Fort Calhoun 1	INC			26	0			
	PSC	PSA-3	6	4	0			1,3
	PSC	PSA-10	15	4	0			
Nineteen INCs were replaced with hydraulic snubbers of equivalent size or larger (3.0 kips). Six INCs were replaced with rigid struts per reanalysis of the AFS. One INC was removed.								
Fort St. Vrain	INC	MSVA-1A	0.75	34	0			1,3
	INC	MSVA-1AS	0.75	14	0			
	INC	MSVA-3AS	10	2	0			
	INC	MSVA-4AS	20	14	0			
Ginna	PSC	PSA-3	6	1	0			
Additional PSCs stored for safety-related use met test requirements.								1,3
Grand Gulf 1	PSC	PSA-1/4	0.35	1	0			
	PSC	PSA-3	6	1	0			2,3
	PSC	PSA-10	15	4	0			
Haddam Neck	PSC	PSA-1/4	0.35	15	0			
	PSC	PSA-3	6	9	0			1,3
	PSC	PSA-10	15	2	0			
	PSC	PSA-35	50	8	0			
Hatch 1	INC			42	0			
All INCs were replaced with PSCs.								1,3

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Hatch 2	PSC	PSA-1/4	0.35	18	0			1,3
	PSC	PSA-1	1.50	53	2	Locked(1),no drag(1)	Unknown	
	PSC	PSA-3	6	73	0			
	PSC	PSA-10	15	132	0			
	PSC	PSA-35	50	23	1	Locked	Corrosion	

Replacement of INCs with PSCs was in process when IEB 81-01 was issued and was completed shortly thereafter. The failed PSCs were replaced prior to startup.
See notes at end of table.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
LaSalle 1	PSC	PSA-1/4	0.35	350	13	Damage	Abuse	2,3
	PSC	PSA-1/2	0.65	96	1	Damage	Abuse	
	PSC	PSA-1	1.50	70	0			
	PSC	PSA-3	6	207	2	Damage	Abuse	
	PSC	PSA-10	15	288	0			
	PSC	PSA-35	50	253	2	Damage	Abuse	
	PSC	PSA-100	120	28	0			

All damaged snubbers were replaced with the same models.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
McGuire 1	PSC	PSA-1/4	0.35	3	0			1,3
	PSC	PSA-1/2	0.65	326	3	Locked	Abuse	
	PSC	PSA-1	1.50	158	2	Rough action		
	PSC	PSA-3	6	96	0			
	PSC	PSA-10	15	38	0			
	PSC	PSA-35	50	21	0			

Failed snubbers were sent to PSC for reconditioning. Piping and hangers adjacent to failed snubbers were checked visually and no damage was found.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Millstone 1	INC	MSVA-3	10	10	1	Locked		1,3
	PSC	PSA-3	6	1	0			
	PSC	PSA-10	15	4	2	Inertia mass separation	Water hammer	

All INCs were located in condensate lines and were replaced with PSA-10s. The failed PSA-10s were replaced with PSA-35s, which are designed for water hammer loads. Piping systems were not identified.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Millstone 2	INC	MSVA-1	0.75	24	3	Jamming		1,3
	INC	MSVA-2	3	44	2	Locked	Corrosion	
	INC	MSVA-3	10	12	0			
	INC	MSVA-4	20	1	0			
	PSC	PSA-1/4	0.35	5	0			
	PSC	PSA-10	15	6	0			

All INCs were replaced with snubbers of a different design. The piping systems were identified.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Nine Mile Point 1	PSC	PSA-10	15	1	0			1,3
	PSC	PSA-35	50	35	0			

The PSA-35s were installed on the reactor recirculation system during the spring of 1981.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
North Anna 1	PSC	PSA-1/4	0.35	4	0			1,3

Four snubbers were installed during the spring of 1981; they were removed and tested satisfactorily by Wyle in July of 1982. The piping systems were not identified.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Oconee 1	PSC	PSA-1/4	0.35	6	0			1,3
	PSC	PSA-1/2	0.65	1	0			

Quantities of accessible PSCs were not reported; however, it was stated that none failed. Piping systems were not identified.

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Oconee 2	PSC	PSA-1/4	0.35	4	0			1,3
		PSA-1/2	0.65	7	0			
		PSA-1	1.50	5	0			
		PSA-3	6	1	0			
Quantities of accessible PSCs were not reported; however, it was stated that none failed. Piping systems were not identified.								
Oconee 3	PSC	PSA-35	50	12	0			1,3
Quantities of accessible PSCs were not reported; however, it was stated that one mechanical snubber No. 3-07A-6-0-2400A-H78 was discovered locked up on the LP Condensate System. Other piping systems were not identified.								
Oyster Creek 1	PSC	PSA-11	11	91	2	Locked	Corrosion	1,3
In one case, interference with an adjacent snubber contributed to failure. The failed snubbers were replaced and the causes of failure were corrected. The affected piping systems were not overstressed.								
Palisades	PSC			7	0			1,3
Model numbers and piping systems were not identified.								
Peach Bottom 2	PSC	PSA-1/2	0.65	8	0			1,3
	PSC	PSA-1	1.50	1	0			
	PSC	PSA-3	6	14	0			
	PSC	PSA-10	15	24	0			
	PSC	PSA-35	50	1	0			
Piping systems were not identified.								
Peach Bottom 3	PSC	PSA-1/2	0.65	8	0			1,3
	PSC	PSA-1	1.50	1	0			
	PSC	PSA-3	6	3	0			
	PSC	PSA-10	15	50	0			
	PSC	PSA-35	50	1	0			
Piping systems were not identified.								
Pilgrim 1	PSC	PSA-10	15	35	0			1,3
All snubbers are located in the drywell to support the four branch lines of the main steam relief discharge system.								
Quad Cities 1								1,3
Quad Cities 2	PSC			33	0			1,3
Model numbers and piping systems were not identified.								
Salem 1	INC	MSVA-2	3	20	3	Fatigue	Vibration	1,3
	INC	MSVA-2A	3	3	1	Fatigue	Vibration	
	INC	MSVA-3	10	2	0			
	INC	MSVA-4	20	5	0			
	PSC	PSA-1/4	0.35	15	2	Locked	Abuse	
	PSC	PSA-1/2	0.65	4	0			
	PSC	PSA-1	1.50	21	0			
	PSC	PSA-3	6	35	0			
	PSC	PSA-10	15	10	0			
	PSC	PSA-35	50	3	0			
The failed MSVAs were planned for replacement with strut restraints; they were located at the RHR pump motors. One PSA-1/4 was replaced with a larger capacity unit, two PSA-1/4s were removed and one PSA-1/4 was not specifically addressed. These four PSA-1/4s were located in the steam generator blowdown system. The utility indicated that system integrity was not impaired by the failed snubbers.								

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Salem 2	INC	MSVA-2	3	20	3	Fatigue	Vibration	1,3
	INC	MSVA-2A	3	3	1	Fatigue	Vibration	
	INC	MSVA-3	10	1	0			
	INC	MSVA-4	20	5	0			
	PSC	PSA-1/4	0.35	14	2	Locked	Abuse	
	PSC	PSA-1/2	0.65	3	0			
	PSC	PSA-1	1.50	21	0			
	PSC	PSA-3	6	35	0			
	PSC	PSA-10	15	11	0			
	PSC	PSA-35	50	2	0			

The failed MSVAs were planned for replacement with strut restraints; they were located at the RHR pump motors. One PSA-1/4 was replaced with a larger capacity unit, two PSA-1/4s were removed and one PSA-1/4 was not specifically addressed. These four PSA-1/4s were located in the steam generator blowdown system. The utility indicated that system integrity was not impaired by the failed snubbers.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
San Onofre 1	PSC	PSA-1	1.50	2	0			1,3
	PSC	PSA-3	6	13	1	Locked, missing screw	Corrosion	
	PSC	PSA-10	15	7	1	Locked, brinelled	Installation	
	PSC	PSA-35	50	9	0			

The failed PSA-3 was repaired, tested and returned to service. The failed PSA-10 was repaired and was to be reinstalled correctly. Thermal stresses with the snubbers locked were less than allowable.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Sequoyah 1	PSC	PSA-1/4	0.35	188	0			1,3
	PSC	PSA-1/2	0.65	109	0			
	PSC	PSA-1	1.50	107	0			
	PSC	PSA-3	6	137	0			
	PSC	PSA-10	15	107	0			
	PSC	PSA-35	50	26	0			
	PSC	PSA-100	120	7	0			

Piping systems were not identified.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
St. Lucie 1	PSC	PSA-1/4	0.35	79	27	Locked(12),stiff(4), setting(11)	Corrosion,fractures, installation	1,3
	PSC	PSA-1	1.50	6	3	Setting	Installation	

Piping systems with failed snubbers were identified; evaluation of these systems indicated that stresses were allowable.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Summer 1	PSC	PSA-1/4	0.35	257	10	Locked(9),disengaged(1)	Installation(6),overload(4)	2,3
	PSC	PSA-1/2	0.65	148	6	Locked	Installation(1),overload(5)	
	PSC	PSA-1	1.50	335	2	Locked	Installation	
	PSC	PSA-3	6	331	5	Locked	Installation(3),abuse(2)	
	PSC	PSA-10	15	196	1	Locked	Installation	
	PSC	PSA-35	50	109	1	Locked	Installation	
	PSC	PSA-100	120	4	0			

Piping systems with failed snubbers were identified. There was no evidence to indicate that the overloads were caused by improper operation.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Surry 1	PSC	PSA-1/4	0.35	7	0			1,3
	PSC	PSA-1/2	0.65	20	0			
	PSC	PSA-1	0.50	6	0			
	PSC	PSA-3	6	23	1	Locked	Installation	
	PSC	PSA-10	15	6	0			
	PSC	PSA-35	50	6	0			

All piping systems were identified. The failed snubber was replaced with a new PSA-3 before startup.

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Surry 2	PSC	PSA-1/4	0.35	7	2(size			1,3
	PSC	PSA-1/2	0.65	2	not			
	PSC	PSA-1	1.50	4	given)			
	PSC	PSA-3	6	12				

Two failed snubbers were reported but not identified; they were replaced with new snubbers. One of the failed snubbers was locked, the other was stiff. The system with the locked snubber was reanalyzed and was found not to have been overstressed.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Susquehanna 1	PSC	PSA-1/4	0.35	722	70			2,3
	PSC	PSA-1/2	0.65	236	11			
	PSC	PSA-1	1.50	187	12			
	PSC	PSA-3	6	207	12			
	PSC	PSA-10	15	194	1			
	PSC	PSA-35	50	455	5			
	PSC	PSA-100	120	68	3			

The failed snubbers were replaced. Minor repairs were made at the site. Piping systems were not identified. Most failures were caused by abuse during construction.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Trojan	INC			1	0			1,3
	PSC	PSA-1/4	0.35	42	3	Locked(2),partial stroke(1)		
	PSC	PSA-1/2	0.65	76	5	Locked(3),rough(1), broken(1)	Installation,abuse	
	PSC	PSA-1	1.50	35	0			
	PSC	PSA-3	6	111	1	Rough	Abuse	
	PSC	PSA-10	15	101	3	Locked(2),rough(1)	Overload	
	PSC	PSA-35	50	5	0			

The solitary INC was replaced with a PSC. The failed PSCs were replaced. The cause of overload had not been determined. Stress analyses indicated that the systems with failed snubbers had suffered no ill effects.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Turkey Point 3	PSC	PSA-3	6	20	1	Locked	Broken anchor bolts	1,3
	PSC	PSA-10	15	39	2	Locked	Overload	
	PSC	PSA-35	50	5	0			

The failed PSA-3 and anchor bolts were replaced; charging system accumulators were installed to reduce vibration greatly. The failed PSA-10s were replaced with new PSA-10s, which were to be replaced with PSA-35s during the outage of July 1981. The utility reported that plant safety was not affected.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Turkey Point 4	PSC	PSA-3	6	18	0			1,3
	PSC	PSA-10	15	37	2	Erratic,no restraint	Overload	
	PSC	PSA-35	50	5	0			

The failed snubbers were located on the "B" steam generator feedwater line. Stress evaluations of the water hammer event indicated no piping damage. During the ongoing steam generator repair (February 1982), the damaged reducer joint was to be repaired. The new steam generators were expected to prevent water hammer. It was planned to replace the failed PSA-10s with PSA-35s.

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Yankee-Rowe 1	PSC	PSA-3	6	8	0			1,3
	PSC	PSA-100	120	8	0			
Zion 1	PSC	PSA-1/2	0.65	1	0			1,4
	PSC	PSA-1	1.50	6	0			
	PSC	PSA-3	6	8	0			
	PSC	PSA-10	15	6	0			
	PSC	PSA-35	50	8	0			

See notes at end of table.

TABLE B.3 (contd.)

Facility	Manuf.	Model Number	Size, kips	Number Total	Number Failed	Mode of Failure	Cause of Failure	Note
Zion 2	PSC	PSA-1/4	0.35	1	0			1,4
	PSC	PSA-1/2	0.65	7	0			
	PSC	PSA-1	1.50	4	0			
	PSC	PSA-3	6	14	0			
	PSC	PSA-10	15	9	0			
	PSC	PSA-35	50	12	0			

Notes:
1. This facility had an operating license (OL) or a near term OL when the bulletin was issued and was required to perform bulletin actions 1 through 4 for OLs.
2. This selected facility had a construction permit (CP) when the bulletin was issued and was required to perform bulletin actions 1 and 2 for CPs.
3. Bulletin status is closed per Criterion 3.
4. Bulletin closeout status is open.

TABLE B.4 LIST OF FACILITIES ISSUED BULLETIN FOR INFORMATION ONLY

Facility	Utility	Docket Number	Facility Status	NRC Region
Bailly 1	NIPSCO	50-367	CD	III
Beaver Valley 2	DLC	50-412	CP	I
Bellefonte 1	TVA	50-438	CP	II
Bellefonte 2	TVA	50-439	CP	II
Braidwood 1	CECO	50-456	CP	III
Braidwood 2	CECO	50-457	CP	III
Byron 1	CECO	50-454	LPTL	III
Byron 2	CECO	50-455	CP	III
Callaway 1	UE	50-483	OL	III
Callaway 2	UE	50-486	CD	III
Catawba 1	DUPCO	50-413	OL	II
Catawba 2	DUPCO	50-414	CP	II
Cherokee 1	DUPCO	50-491	CD	II
Cherokee 2	DUPCO	50-492	CD	II
Cherokee 3	DUPCO	50-493	CD	II
Clinton 1	IP	50-461	CP	III
Clinton 2	IP	50-462	CHI	III
Comanche Peak 1	TUGCO	50-445	CP	IV
Comanche Peak 2	TUGCO	50-446	CP	IV
Dresden 1	CECO	50-010	SDI	III
Fermi 2	DECO	50-341	CP	III
Forked River	JCP&L	50-363	CD	I
Grand Gulf 2	MP&L	50-417	CHI	II
Harris 1	CP&L	50-400	CP	II
Harris 2	CP&L	50-401	CHI	II
Harris 3	CP&L	50-402	CHI	II
Harris 4	CP&L	50-403	CHI	II
Hartsville A-1	TVA	50-518	CD	II
Hartsville A-2	TVA	50-519	CD	II
Hartsville B-1	TVA	50-520	CD	II
Hartsville B-2	TVA	50-521	CD	II
Hope Creek 1	PSE&G	50-354	CP	I
Hope Creek 2	PSE&G	50-355	CHI	I
Humboldt Bay 3	PG&E	50-133	SDI	V
Indian Point 1	ConEd	50-003	SDI	I
Jamesport 1	LILCO	50-516	CD	I
Jamesport 2	LILCO	50-517	CD	I
LaSalle 2	CECO	50-374	OL	III
Limerick 1	PECO	50-352	LPTL	I
Limerick 2	PECO	50-353	CP	I

See notes at end of table.

TABLE B.4 (contd.)

Facility	Utility	Docket Number	Facility Status	NRC Region
Marble Hill 1	PSI	50-546	CHI	III
Marble Hill 2	PSI	50-547	CHI	III
McGuire 2	DUPCO	50-370	OL	II
Midland 1	CPC	50-329	CHI	III
Midland 2	CPC	50-330	CHI	III
Millstone 3	NNECO	50-423	CP	I
Nine Mile Point 2	NMP	50-410	CP	I
North Anna 3	VEPCO	50-404	CD	II
North Anna 4	VEPCO	50-405	CD	II
Palo Verde 1	APSCO	50-528	LPTL	V
Palo Verde 2	APSCO	50-529	CP	V
Palo Verde 3	APSCO	50-530	CP	V
Perkins 1	DUPCO	50-488	CD	II
Perkins 2	DUPCO	50-489	CD	II
Perkins 3	DUPCO	50-490	CD	II
Perry 1	CEI	50-440	CP	III
Perry 2	CEI	50-441	CP	III
Phipps Bend 1	TVA	50-553	CD	II
Phipps Bend 2	TVA	50-554	CD	II
River Bend 1	GSU	50-458	CP	IV
River Bend 2	GSU	50-459	CD	IV
San Onofre 2	SCE	50-361	OL	V
San Onofre 3	SCE	50-362	OL	V
Seabrook 1	PSNH	50-443	CP	I
Seabrook 2	PSNH	50-444	CP	I
Sequoyah 2	TVA	50-328	OL	II
Shoreham	LILCO	50-322	CP	I
South Texas 1	HL&P	50-498	CP	IV
South Texas 2	HL&P	50-499	CP	IV
St. Lucie 2	FPL	50-389	OL	II
Sterling	RG&E	50-485	CD	I
Susquehanna 2	PP&L	50-388	OL	I
TMI 2	Met-Ed	50-320	SDI	I
Vogtle 1	GP	50-424	CP	II
Vogtle 2	GP	50-425	CP	II
WNP 1	WPPSS	50-460	CP	V
WNP 2	WPPSS	50-397	OL	V
WNP 3	WPPSS	50-508	CP	V
WNP 4	WPPSS	50-513	CHI	V
WNP 5	WPPSS	50-509	CHI	V
Waterford 3	LP&L	50-382	LPTL	IV
Watts Bar 1	TVA	50-390	CP	II
Watts Bar 2	TVA	50-391	CP	II
Wolf Creek 1	KG&E	50-482	CP	IV
Yellow Creek 1	TVA	50-566	CHI	II
Yellow Creek 2	TVA	50-567	CHI	II
Zimmer 1	CG&E	50-358	CD	III

See notes at end of table.

TABLE B.4 (contd.)

Notes:

1. CD, Cancelled; CHI, Construction Halted Indefinitely; CP, Construction Permit; LPTL, Low Power Testing License; OL, Operating License; SDI, Shut Down Indefinitely; based on references 1, 2 and 3 (Page B-19).

2. The following nine operating facilities did not have licenses for operation or low power testing when the bulletin was issued in 1981:

Callaway 1	San Onofre 2,3	Susquehanna 2
LaSalle 2	Sequoyah 2	WNP 2
McGuire 2	St. Lucie 2	

REFERENCES

1. United States Nuclear Regulatory Commission, Licensed Operating Reactors, Status Summary Report, Data as of 02-28-85, NUREG-0020, Volume 9, Number 3, March 1985

2. United States Nuclear Regulatory Commission, Nuclear Power Plants, Construction Status Report, Data as of 06-30-82, NUREG-0030, Volume 6, Number 2, October 1982

3. United States Nuclear Regulatory Commission, Listing of Inactive Current Holders of Construction Permits, Letter dated May 29, 1985, to Richard A. Lofy (Parameter, Inc.) from Robert L. Baer (NRC/IE HQ)

APPENDIX C

Proposed Followup Items

Region III

1. Duane Arnold
   Utility personnel responded acceptably March 13, 1981, June 1, 1981, and November 10, 1983, indicating that (a) one INC snubber tested by Wyle Laboratories and found to be inoperable was subsequently replaced by a PSC snubber (b) ten PSC snubbers were considered to be inoperable and (c) four PSC snubbers were replaced with snubbers of the same size, five were reconditioned by cleaning and oiling frozen bushings and one was repaired by using safety wire of specified diameter to secure transition tube bolts.

   Verification is incomplete or not fully documented that visual examinations, operability tests and corrective actions have been performed acceptably.

2. Zion 1
   Utility personnel responded acceptably March 2, March 17 and June 9, 1981, indicating that (a) INC snubbers were not used, (b) all snubbers were manufactured by PSC, (c) all snubbers were considered operable and (d) no further action was necessary.

   Verification is incomplete or not fully documented that visual examinations and operability tests have been performed acceptably.

3. Zion 2
   Utility personnel responded acceptably March 2, 1981, March 19, 1981 and January 14, 1982, indicating that (a) INC snubbers were not used, (b) all snubbers were manufactured by PSC, (c) results of testing showed that all snubbers were undamaged, (d) all snubbers were considered operable and (e) no further action was necessary.

   Verification is incomplete or not fully documented that visual examinations and operability tests have been performed acceptably.

Region V

Diablo Canyon 2

Utility personnel responded acceptably May 5, 1981, May 15, 1982, February 15, 1985 and May 28, 1985, indicating that (a) inspection of mechanical snubbers per the bulletin was completed April 26, 1985, (b) all snubbers with impaired operability were replaced prior to fuel load, which began on May 7, 1985 and (c) inspection results for each snubber were filed separately in the Maintenance Department. The utility response of May 28, 1985 is the only response in which mechanical snubbers manufactured by Anchor/Darling Industries are mentioned.

Verification is incomplete or not fully documented that visual examinations, operability tests and corrective actions have been performed acceptably.

APPENDIX D

Utility Manhours Expended on IEB 81-01

TABLE D.1

Facility	Review & Reporting	Corrective Action	Total	Closeout Status and Criterion
Arkansas 1	95	NR*	NR	Closed 3
Arkansas 2	260	4825	5085	Closed 3
Beaver Valley 1	90	NR	NR	Closed 3
Big Rock Point 1	200	NR	NR	Closed 3
Browns Ferry 1,2,3	NR	NR	200	Closed 3
Brunswick 1,2	8	0	8	Closed 1
Cooper Station	60	NR	NR	Closed 3
Crystal River 3	16	0	16	Closed 1
Diablo Canyon 1	2400	NR	NR	Closed 3
Diablo Canyon 2	740	2400	3140	Open
Duane Arnold	1100	235	1335	Open
Farley 1	2620	48	2668	Closed 3
Farley 2	0	0	0	Closed 3
Ginna	6	NR	NR	Closed 3
LaCrosse	1	0	1	Closed 3
Maine Yankee	10	0	10	Closed 1
Millstone 1	220	700	920	Closed 3
Millstone 2	NR	NR	$740,000	Closed 3
Monticello	2	C	2	Closed 1
Nine Mile Point 1	16	NR	NR	Closed 3
Oyster Creek 1	1000	NR	NR	Closed 3
Peach Bottom 2,3	60	NR	NR	Closed 3
Point Beach 1,2	4	0	4	Closed 1
Robinson 2	8	0	8	Closed 1
Sequoyah 1	190	NR	NR	Closed 3
Susquehanna 1	20,500	NR	NR	Closed 3
TMI 1	3	0	3	Closed 1
Trojan	150	NR	NR	Closed 3
Yankee-Rowe 1	31	NR	NR	Closed 3

*NR signifies "not reported"

APPENDIX E

Abbreviations

A/D	Anchor/Darling Industries
AFS	Auxiliary Feedwater System
APCO	Alabama Power Company
AP&L	Arkansas Power and Light Company
APSCO	Arizona Public Service Company
BECO	Boston Edison Company
BG&E	Baltimore Gas and Electric Company
CD	Cancelled
CECO	Commonwealth Edison Company
CEI	Cleveland Electric Illuminating Company
CG&E	Cincinnati Gas and Electric Company
CHI	Construction Halted Indefinitely
ConEd	Consolidated Edison Company of New York, Inc.
CP	Construction Permit
CPC	Consumers Power Company
CP&L	Carolina Power and Light Company
CR	Contractor Report
CYAPCO	Connecticut Yankee Atomic Power Company
DECO	Detroit Edison Company
DL	Division of Licensing (NRC)
DLC	Duquesne Light Company
DPC	Dairyland Power Cooperative
DUPCO	Duke Power Company
FP	Florida Power Corporation
FPL	Florida Power & Light Company
GAO	Government Accounting Office
GP	Georgia Power Company
GSU	Gulf States Utilities Company
HL&P	Houston Lighting and Power Company
HQ	Headquarters (NRC)
IE	Inspection and Enforcement, Office of (NRC)
IEB	Inspection and Enforcement Bulletin (NRC)
IEC	Inspection and Enforcement Circular (NRC)
IEIN	Inspection and Enforcement Information Notice (NRC)
IELPCO	Iowa Electric Light and Power Company
IMECO	Indiana and Michigan Electric Company
INC	International Nuclear Safeguards Corporation
IP	Illinois Power Company
JCP&L	Jersey Central Power and Light Company

KG&E	Kansas Gas and Electric Company
kips	Units of 1000 pounds
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LILCO	Long Island Lighting Company
LP&L	Louisiana Power and Light Company
LPTL	Low Power Testing License
Manuf.	Manufacturer
Met-Ed	Metropolitan Edison Company
MP&L	Mississippi Power and Light Company
MSIV	Main Steam Isolation Valve
MYAPCO	Maine Yankee Atomic Power Company
NIPSCO	Northern Indiana Public Service Company
NMP	Niagara Mohawk Power Company
NNECO	Northeast Nuclear Energy Company
NPPD	Nebraska Public Power District
NRC	Nuclear Regulatory Commission
NSP	Northern States Power Company
NU	Northeast Utilities
OL	Operating License
OPPD	Omaha Public Power District
PASNY	Power Authority of the State of New York
PECO	Philadelphia Electric Company
PGE	Portland General Electric Company
PG&E	Pacific Gas and Electric Company
PP&L	Pennsylvania Power and Light Company
PSC	Pacific Scientific Company
PSCC	Public Service Company of Colorado
PSCO	Public Service Company of Oklahoma
PSE&G	Public Service Electric and Gas Company
PSI	Public Service Indiana
PSNH	Public Service Company of New Hampshire
R	Region (NRC)
RC	Reactor Coolant
RDS	Reactor Depressurization System
RG&E	Rochester Gas and Electric Corporation
RPV	Reactor Pressure Vessel
RR	Reactor Recirculation
SCE	Southern California Edison Company
SCE&G	South Carolina Electric and Gas Company
SDI	Shut Down Indefinitely
SMUD	Sacramento Municipal Utility District
SNUPPS	Standardized Nuclear Unit Power Plant Systems
SRV	Safety Relief Valve
TECO	Toledo Edison Company
TI	Temporary Instruction (NRC/IE)
TMI	Three Mile Island
TUGCO	Texas Utilities Generating Company
TVA	Tennessee Valley Authority

UE	Union Electric Company
VEPCO	Virginia Electric and Power Company
VYNP	Vermont Yankee Nuclear Power Corporation
W	Westinghouse Electric Corporation
WEPCO	Wisconsin Electric Power Company
WNP	Washington Nuclear Project
WPPSS	Washington Public Power Supply System
WPS	Wisconsin Public Service Corporation
YAECO	Yankee Atomic Electric Company

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BIBLIOGRAPHIC DATA SHEET

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13. ABSTRACT (200 words or less)

In the period from August 1974 to May 1980, failures of mechanical snubbers were described in event reports issued for nine facilities and in a NRC/IE study of the DOE Fast Flux Test Facility. In most failures, the snubbers were frozen and would not permit free piping motions during thermal transients. In some cases, the failed snubbers no longer provided seismic shock restraint. Because of concern about the reported failures of mechanical snubbers, standard technical specification revisions for snubber surveillance were issued by NRC/DL on November 20, 1980. IE Bulletin 81-01 was issued January 27, 1981 to require examination and testing of mechanical snubbers in safety-related systems at licensed facilities and at selected facilities under construction. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out per specific criteria for 73 (95%) of the 77 facilities to which it was issued for action. Followup items are proposed for use by NRC/IE to ensure satisfactory completion of corrective action at the remaining four (4) facilities.

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