September 14, 1976

Dockets Nos.: 50-280
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Virginia Electric & Power Company
ATTN: Mr. W. L. Proffitt
Senior Vice President
P. O. Box 26666
Richmond, Virginia 23261

Gentlemen:

The Commission has issued the enclosed Amendment No. 25 to Facility Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station, Units Nos. 1 and 2. The amendments consist of changes to your Technical Specifications for each license and are in response to your request dated March 22, 1976.

The amendments incorporate provisions into the Technical Specifications related to limiting conditions for operation and surveillance of shock suppressors (snubbers). We have made certain changes in the Technical Specifications you proposed and have discussed these changes with your staff on August 24, 1976.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 25 to DPR-32
2. Amendment No. 25 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures: See next page

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Mr. Sherlock Holmes, Chairman
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cc w/enclosure(s) & incoming
dtd: 3/22/76
Commonwealth of Virginia
Council on the Environment
903 9th Street Office Building
Richmond, Virginia 23219

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated March 22, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 14, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Insert pages TS 3.20-1 through 3.20-2 and TS 4.17-1 through 4.17-14. These are all new pages in the Technical Specifications and are shown by marginal lines.

3.20 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to all shock suppressors (snubbers) which are required to protect the reactor coolant system and safety related systems. The specific snubbers to which this specification applies are listed in Technical Specification 4.17.

Objective

To define those limiting conditions for operation that are necessary to ensure that all snubbers required to protect the reactor coolant system or any other safety related system or component are operable during reactor operation.

Specifications

A. During all modes of operation except Cold Shutdown and Refueling, all safety-related snubbers listed in Technical Specification 4.17 shall be operable except as noted in 3.20.B and 3.20.C below.

B. If any snubber listed in Technical Specification 4.17 is found to be inoperable, it must be repaired and made operable, or otherwise replaced with one which is operable within 72 hours.

C. If the requirements of specification B cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in the hot shutdown condition within 36 hours.

D. If a snubber is determined to be inoperable while the reactor is in the shutdown or refueling mode, the snubber shall be made operable or replaced prior to reactor startup.

E. Snubbers may be added to safety related systems provided that a revision to Technical Specification 4.17 is included with the next license amendment.

Basis

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety related system or component be operable during reactor operation.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacement. In case a shutdown is required, the allowance of 36 hours to reach a hot shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.20.D prohibits startup with inoperable snubbers.

4.17 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to all hydraulic shock suppressors (snubbers) which are required to protect the reactor coolant system and safety related systems.

Objective

To specify the minimum frequency and type of surveillance to be applied to the hydraulic snubbers listed in Table 4.17-1 and 4.17-2.

Specification

A. All hydraulic shock suppressors whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected.

This inspection shall include but not necessarily be limited to, inspection of the hydraulic-fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 Months ± 25%
1	12 Months ± 25%
2	6 Months ± 25%

3, 4	124 Days	$\pm$ 25%
5, 6, 7	62 Days	$\pm$ 25%
$\geq$ 8	31 Days	$\pm$ 25%

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

Snubbers may be categorized into two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

B. All hydraulic snubbers whose seal material are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

C. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule into specification 4.17-A, it shall be assumed that the facility had been on a 6 month inspection schedule.

D. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock-up and bleed.

For each unit and subsequent unit found inoperable, an additional 10% or 10 hydraulic snubbers shall be so tested until no more failures are found, or all units have been tested. Snubbers of rated capacity greater than 50,000 lb. need not be functionally tested.

Basis

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required 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.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 1) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Tables 4.17-1 and 4.17-2 as being in high radiation areas during shutdown or those especially difficult to remove need not be selected for functional tests provided operability was previously verified. Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the impracticability of testing such large units.

LEGEND

ACCESSIBILITY CATEGORY

A = Accessible

I = Inaccessible

RADIATION CATEGORY

H = High radiation area only during periods of reactor operation.
In acceptable radiation work area during periods of reactor shutdown.*

N = Acceptable radiation work area during periods of both reactor operation and shutdown.

REMOVAL CATEGORY

D = Difficult to remove, i.e. large line size, large component, physical location (overhead), lines under load, jacks necessary

R = Can be removed

*Modifications to this table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

Amendment No. 25

TABLE 4.17-1

UNIT NO. 1 SUPPRESSOR DATA

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Main Steam System	1-SHP-HSS-1A & 1B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam System	1-SHP-HSS-2A & 2B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-3A & 3B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-4A & 4B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-5A & 5B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-6A & 6B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-7-12	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-13A	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-14A & 14B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam System	1-SHP-HSS-15-20	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-21-26	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam System	1-SHP-HSS-27-32	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-33A & 33B	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-34A & 34B	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-35A & 35B	Safeguards Bldg.	A	N	D
Main Feed System	1-WFPD-HSS-1	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-2	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-3	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-4	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-5	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-6	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-7	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-8	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-9	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-10	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-11	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-12	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-13	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-14	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-15	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-16	Containment Operating Level	I	H	D

TABLE 4.17-1

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Auxiliary Feed System	1-WAPD-HSS-140	Containment Operating Level	A	H	R
Auxiliary Feed System	1-WAPD-HSS-141	Containment Operating Level	A	H	R
Auxiliary Feed System	1-WAPD-HSS-142	Containment Operating Level	A	H	R
Auxiliary Feed System	1-WAPD-HSS-143	Containment Operating Level	A	H	R
Reactor Coolant System	1-RC-HSS-101	Pressurizer Room-Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-102	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-104	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-105	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-106	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-107	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-108	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-109	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-110	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-111	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-112	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-113	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-114	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-115	Pressurizer Room Spray Line, Operating Level	I	H	R
Reactor Coolant System	1-RC-HSS-118	Pressurizer Room Relief Line, Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-119	Pressurizer Room Relief Line, Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-120	Pressurizer Room Relief Line, Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-121	Pressurizer Room Relief Line, Operating Level	I	H	D

TABLE 4.17-1

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Reactor Coolant System	**Four (4)**	**Suppressors around base of pressurizer with common reservoir**	**A**	**H**	**D**
Reactor Coolant System	Two (2)	Reservoirs in each loop room-one for steam generator suppressor, one for two large suppressors on RCP	I	H	D
Reactor Coolant System	Four (4)	Small suppressors on RCP holding assembly	I	H	D
Low Head Safety Injection System	1-SI-HSS-(19) 2	Containment, Basement, By "A" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-20	Containment, Basement, By "A" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-21	Containment, Basement, By "A" Accumulator	I	H	R
Low Head Safety Injection System	1-SI-HSS-(22) 2	Containment, Basement, By "B" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-23	Containment, Basement, By "B" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-24	Containment, Basement, Overhead, By "B" Accumulator	I	H	D
Low Head Safety Injection System	1-SI-HSS-25-26	Containment, Basement, By "C" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-27	Containment, Basement, Overhead, By "C" Accumulator	I	H	D
Low Head Safety Injection System	1-SI-HSS-100	Safeguards, Second Level, Valve Pit	A	N	D
Low Head Safety Injection System	1-SI-HSS-101	Safeguards, Second Level, Valve Pit	A	N	D
Recirc. Spray System	1-RS-HSS-101-104	Containment, -3' Level	I	H	D
Chemical Volume & Control System	1-CH-HSS-301-304	Aux. Bldg., "C" Chg. Pump Cube	A	N	D
Residual Heat Removal System	1-RH-HSS-1-4	Containment, RHR Flat, RHR Pump Support	A	H	D
Residual Heat Removal System	1-RH-HSS-5-8	Containment, Top of RHR HX's A&B	I	H	D

TABLE 4.17-1

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Residual Heat Removal System	1-RH-HSS-9-11	Containment, RHR Flat, RHR Pump Discharge Line	I	H	D
Residual Heat Removal System	1-RH-HSS-12-14	Containment, RHR Pump Suction Line, Basement	I	H	D
Residual Heat Removal System	1-RH-HSS-15	Containment, RHR Flat, Outlet of "B" HX	I	H	R
Residual Heat Removal System	1-RH-HSS-17-18	Containment, "A" Loop Rm. Near MOV-1700	I	H	R
Residual Heat Removal System	1-RH-HSS-19-20	Containment, RHR Flat, Near MOV-1701	I	H	D
Residual Heat Removal System	1-RH-HSS-21-23	Containment, Basement, Near MOV-1720A & B	I	H	D
Residual Heat Removal System	1-RH-HSS-24-25	Containment, Basement, Under RHR Flat on Return Line	I	H	D
Residual Heat Removal System	1-RH-HSS-100	Containment, 15' Level on RV-1721 Discharge Line	I	H	R
Residual Heat Removal System	1-RH-HSS-101A & B	Containment, Basement on RHR Let-down Line	I	H	D
Residual Heat Removal System	1-RH-HSS-102	Containment, Basement on RHR Re-turn Hd-MOV-1720A	I	H	D
Component Cooling System	1-CC-C-65, 331, 332, 340A & B	Aux. Bldg. Basement in Overhead on Unit No. 1 Side	A	N	D
Component Cooling System	1-CC-C-356A & B	Aux. Bldg. Basement, on Unit No. 2 Side	A	N	R
Component Cooling System	1-CC-C-60A & 330	Aux. Bldg. Basement Component Cooling Water Pumps	A	N	D

TABLE 4.17-2

UNIT NO. 2 SUPPRESSOR DATA

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Main Steam Lines	2-SHP-HSS-1A, 1B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-14A, 14B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-21-26	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-27-32	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-15-20	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-2A, 2B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-3A, 3B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-4A, 4B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-5A, 5B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-6A, 6B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-7-12	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-13A	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-33A, 33B	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-34A, 34B	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-35A, 35B	Safeguards Bldg.	A	N	D
Main Feed System	2-WFPD-HSS-1-13	Containment Operating Level	I	H	D
Aux. Feed System	2-WAPD-HSS-140-143	Containment Operating Level	A	H	R
Reactor Coolant System	2-RC-HSS-100	On Spray Line in Overhead of -18' Level	I.	H	D
Reactor Coolant System	2-RC-HSS-101	Overhead of -18' Level Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-103	Overhead of -18' Level Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-102	In Prz. Relief Tank Room Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-106	In Prz. Relief Tank Room Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-104	On Spray Line in Overhead of -18' Level	I	H	D
Reactor Coolant System	2-RC-HSS-105	On Spray Line in Overhead of -18' Level	I	H	D

Amendment No. 25

TABLE 4.17-2

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Reactor Coolant System	2-RC-HSS-108	On Spray line in Overhead of -18' Level	I	H	D
Reactor Coolant System	2-RC-HSS-107	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-109	On Spray Line in Overhead of -18' Level	I	H	D
Reactor Coolant System	2-RC-HSS-110	On Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-111	On Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-112	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-113	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-114	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-115	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-116	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-117	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-118	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-119	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-120	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-121	On Line Above Pressurizer	I	H	D
Reactor Coolant System			I	N	R
Reactor Coolant System		There are two (2) large suppressors on each steam generator support ring and six (6) suppressors in the reactor coolant pump support structure	I	H	D
Base of Pressurizer	Four (4)	Large suppressors around base of pressurizer on 47'-4" Level	A	H	D
Safety Injection System	2-SI-HSS-(19) 2	At "C" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-20	At "C" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-21	At "C" Accumulator	I	H	D
Safety Injection System	2-SI-HSS-27	At "C" Accumulator	I	H	D
Safety Injection System	2-SI-HSS-25	At "A" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-26	At "A" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-24	At "A" Accumulator	I	H	D
Safety Injection System	2-SI-HSS-23	At "B" Accumulator	A	H	R

TABLE 4.17-2

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Safety Injection System	2-SI-HSS-24	At "B" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-(22) 2	At "B" Accumulator	A	H	R
Residual Heat Removal System	2-RH-HSS-1	On 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-2	On 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-3	On 2-RH-P-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-4	On 2-RH-P-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-5	On 2-RH-E-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-6	On 2-RH-E-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-7	On 2-RH-E-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-8	On 2-RH-E-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-9	Above 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-10	Above 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-11	Above 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-12	In Overhead Below RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-13	In Overhead Below RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-14	In Overhead Below RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-15	Near 2-RH-E-1B at -3'-11" Level	I	H	D
Residual Heat Removal System	2-RH-HSS-17	Back of Loop Room "A" Near Floor	I	H	R
Residual Heat Removal System	2-RH-HSS-18	Back of Loop Room "A" Near Floor	I	H	R
Residual Heat Removal System	2-RH-HSS-19	Under Loop Room "A" Near RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-20	Under Loop Room "A" Near RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-21	At Loop Room Level Over RHR Flat	I	H	R
Residual Heat Removal System	2-RH-HSS-22	At Loop Room Level Over RHR Flat	I	H	R
Residual Heat Removal System	2-RH-HSS-23	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-24	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-25	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-26	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-27	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-28	On 12" Discharge from RHR -15-6 Elevation	I	H	D
Residual Heat Removal System	2-RH-HSS-29	On 12" Discharge from RHR -15-6 Elevation	I	H	D

TABLE 4.17-2

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Recirc. Spray System	2-RS-HSS-101	Outside of Crane Wall (10' El) on Southside	I	H	D
Recirc. Spray System	2-RS-HSS-102	Outside of Crane Wall (10' El) on Southside	I	H	D
Recirc. Spray System	2-RS-HSS-103	Outside of Crane Wall (10' El) on Southside	I	H	D
Recirc. Spray System	2-RS-HSS-104	Outside of Crane Wall (10' El) on Southside	I	H	D
Low Head Safety Injection System	2-SI-HSS-100	In Valve Pit on 10" Safety Injection Line	A	N	D
Low Head Safety Injection System	2-SI-HSS-101	In Valve Pit on 10" Safety Injection Line	A	N	D
Low Head Safety Injection System	2-SI-HSS-102A	In Valve Pit on 10" Safety Injection Line	A	N	D
Low Head Safety Injection System	2-SI-HSS-102B	In Valve Pit on 10" Safety Injection Line	A	N	D
Chemical & Volume Control System	2-CH-HSS-301	Unit No. 2 "C" Charging Pump Cube	A	N	D
Chemical & Volume Control System	2-CH-HSS-302	Unit No. 2 "C" Charging Pump Cube	A	N	D
Chemical & Volume Control System	2-CH-HSS-303	Unit No. 2 "C" Charging Pump Cube	A	N	D
Chemical & Volume Control System	2-CH-HSS-304	Unit No. 2 "C" Charging Pump Cube	A	N	D

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated March 22, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 14, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Insert pages TS 3.20-1 through 3.20-2 and TS 4.17-1 through 4.17-14.

These are all new pages in the Technical Specifications and are shown by marginal lines.

3.20 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to all shock suppressors (snubbers) which are required to protect the reactor coolant system and safety related systems. The specific snubbers to which this specification applies are listed in Technical Specification 4.17.

Objective

To define those limiting conditions for operation that are necessary to ensure that all snubbers required to protect the reactor coolant system or any other safety related system or component are operable during reactor operation.

Specifications

A. During all modes of operation except Cold Shutdown and Refueling, all safety-related snubbers listed in Technical Specification 4.17 shall be operable except as noted in 3.20.B and 3.20.C below.

B. If any snubber listed in Technical Specification 4.17 is found to be inoperable, it must be repaired and made operable, or otherwise replaced with one which is operable within 72 hours.

C. If the requirements of specification B cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in the hot shutdown condition within 36 hours.

D. If a snubber is determined to be inoperable while the reactor is in the shutdown or refueling mode, the snubber shall be made operable or replaced prior to reactor startup.

E. Snubbers may be added to safety related systems provided that a revision to Technical Specification 4.17 is included with the next license amendment.

Basis

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety related system or component be operable during reactor operation.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacement. In case a shutdown is required, the allowance of 36 hours to reach a hot shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.20.D prohibits startup with inoperable snubbers.

4.17 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to all hydraulic shock suppressors (snubbers) which are required to protect the reactor coolant system and safety related systems.

Objective

To specify the minimum frequency and type of surveillance to be applied to the hydraulic snubbers listed in Table 4.17-1 and 4.17-2.

Specification

A. All hydraulic shock suppressors whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected.

This inspection shall include but not necessarily be limited to, inspection of the hydraulic-fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 Months ± 25%
1	12 Months ± 25%
2	6 Months ± 25%

3, 4	124 Days	± 25%
5, 6, 7	62 Days	± 25%
≥ 8	31 Days	± 25%

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

Snubbers may be categorized into two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

B. All hydraulic snubbers whose seal material are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

C. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule into specification 4.17-A, it shall be assumed that the facility had been on a 6 month inspection schedule.

D. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock-up and bleed.

For each unit and subsequent unit found inoperable, an additional 10% or 10 hydraulic snubbers shall be so tested until no more failures are found, or all units have been tested. Snubbers of rated capacity greater than 50,000 lb. need not be functionally tested.

Basis

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required 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.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 1) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Tables 4.17-1 and 4.17-2 as being in high radiation areas during shutdown or those especially difficult to remove need not be selected for functional tests provided operability was previously verified. Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the impracticability of testing such large units.

LEGEND

ACCESSIBILITY CATEGORY

A = Accessible

I = Inaccessible

RADIATION CATEGORY

H = High radiation area only during periods of reactor operation.
In acceptable radiation work area during periods of reactor shutdown.*

N = Acceptable radiation work area during periods of both reactor operation and shutdown.

REMOVAL CATEGORY

D = Difficult to remove, i.e. large line size, large component, physical location (overhead), lines under load, jacks necessary

R = Can be removed

*Modifications to this table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

TABLE 4.17-1

UNIT NO. 1 SUPPRESSOR DATA

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Main Steam System	1-SHP-HSS-1A & 1B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam System	1-SHP-HSS-2A & 2B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-3A & 3B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-4A & 4B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-5A & 5B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-6A & 6B	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-7-12	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-13A	Containment Operating Level	I	H	D
Main Steam System	1-SHP-HSS-14A & 14B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam System	1-SHP-HSS-15-20	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-21-26	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam System	1-SHP-HSS-27-32	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-33A & 33B	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-34A & 34B	Safeguards Bldg.	A	N	D
Main Steam System	1-SHP-HSS-35A & 35B	Safeguards Bldg.	A	N	D
Main Feed System	1-WFPD-HSS-1	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-2	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-3	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-4	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-5	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-6	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-7	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-8	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-9	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-10	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-11	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-12	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-13	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-14	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-15	Containment Operating Level	I	H	D
Main Feed System	1-WFPD-HSS-16	Containment Operating Level	I	H	D

TABLE 4.17-1

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Auxiliary Feed System	1-WAPD-HSS-140	Containment Operating Level	A	H	R
Auxiliary Feed System	1-WAPD-HSS-141	Containment Operating Level	A	H	R
Auxiliary Feed System	1-WAPD-HSS-142	Containment Operating Level	A	H	R
Auxiliary Feed System	1-WAPD-HSS-143	Containment Operating Level	A	H	R
Reactor Coolant System	1-RC-HSS-101	Pressurizer Room-Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-102	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-104	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-105	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-106	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-107	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-108	Basement Overhead-Spray Line	I	H	D
Reactor Coolant System	1-RC-HSS-109	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-110	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-111	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-112	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-113	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-114	Pressurizer Room Relief Line Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-115	Pressurizer Room Spray Line, Operating Level	I	H	R
Reactor Coolant System	1-RC-HSS-118	Pressurizer Room Relief Line, Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-119	Pressurizer Room Relief Line, Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-120	Pressurizer Room Relief Line, Operating Level	I	H	D
Reactor Coolant System	1-RC-HSS-121	Pressurizer Room Relief Line, Operating Level	I	H	D

TABLE 4.17-1

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Reactor Coolant System	Four (4)	Suppressors around base of pressurizer with common reservoir	A	H	D
Reactor Coolant System	Two (2)	Reservoirs in each loop room-one for steam generator suppressor, one for two large suppressors on RCP	I	H	D
Reactor Coolant System	Four (4)	Small suppressors on RCP holding assembly	I	H	D
Low Head Safety Injection System	1-SI-HSS-(19) 2	Containment, Basement, By "A" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-20	Containment, Basement, By "A" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-21	Containment, Basement, By "A" Accumulator	I	H	R
Low Head Safety Injection System	1-SI-HSS-(22) 2	Containment, Basement, By "B" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-23	Containment, Basement, By "B" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-24	Containment, Basement, Overhead, By "B" Accumulator	I	H	D
Low Head Safety Injection System	1-SI-HSS-25-26	Containment, Basement, By "C" Accumulator	A	H	R
Low Head Safety Injection System	1-SI-HSS-27	Containment, Basement, Overhead, By "C" Accumulator	I	H	D
Low Head Safety Injection System	1-SI-HSS-100	Safeguards, Second Level, Valve Pit	A	N	D
Low Head Safety Injection System	1-SI-HSS-101	Safeguards, Second Level, Valve Pit	A	N	D
Recirc. Spray System	1-RS-HSS-101-104	Containment, -3' Level	I	H	D
Chemical Volume & Control System	1-CH-HSS-301-304	Aux. Bldg., "C" Chg. Pump Cube	A	N	D
Residual Heat Removal System	1-RH-HSS-1-4	Containment, RHR Flat, RHR Pump Support	A	H	D
Residual Heat Removal System	1-RH-HSS-5-8	Containment, Top of RHR HX's A&B	I	H	D

TABLE 4.17-1

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Residual Heat Removal System	1-RH-HSS-9-11	Containment, RHR Flat, RHR Pump Discharge Line	I	H	D
Residual Heat Removal System	1-RH-HSS-12-14	Containment, RHR Pump Suction Line, Basement	I	H	D
Residual Heat Removal System	1-RH-HSS-15	Containment, RHR Flat, Outlet of "B" HX	I	H	R
Residual Heat Removal System	1-RH-HSS-17-18	Containment, "A" Loop Rm. Near MOV-1700	I	H	R
Residual Heat Removal System	1-RH-HSS-19-20	Containment, RHR Flat, Near MOV-1701	I	H	D
Residual Heat Removal System	1-RH-HSS-21-23	Containment, Basement, Near MOV-1720A & B	I	H	D
Residual Heat Removal System	1-RH-HSS-24-25	Containment, Basement, Under RHR Flat on Return Line	I	H	D
Residual Heat Removal System	1-RH-HSS-100	Containment, 15' Level on RV-1721 Discharge Line	I	H	R
Residual Heat Removal System	1-RH-HSS-101A & B	Containment, Basement on RHR Let-down Line	I	H	D
Residual Heat Removal System	1-RH-HSS-102	Containment, Basement on RHR Re-turn Hd-MOV-1720A	I	H	D
Component Cooling System	1-CC-C-65, 331, 332, 340A & B	Aux. Bldg. Basement in Overhead on Unit No. 1 Side	A	N	D
Component Cooling System	1-CC-C-356A & B	Aux. Bldg. Basement, on Unit No. 2 Side	A	N	R
Component Cooling System	1-CC-C-60A & 330	Aux. Bldg. Basement Component Cooling Water Pumps	A	N	D

TABLE 4.17-2

UNIT NO. 2 SUPPRESSOR DATA

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Main Steam Lines	2-SHP-HSS-1A, 1B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-14A, 14B	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-21-26	Main Steam Line Area Turb. Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-27-32	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-15-20	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-2A, 2B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-3A, 3B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-4A, 4B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-5A, 5B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-6A, 6B	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-7-12	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-13A	Containment Operating Level	I	H	D
Main Steam Lines	2-SHP-HSS-33A, 33B	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-34A, 34B	Safeguards Bldg.	A	N	D
Main Steam Lines	2-SHP-HSS-35A, 35B	Safeguards Bldg.	A	N	D
Main Feed System	2-WFPD-HSS-1-13	Containment Operating Level	I	H	D
Aux. Feed System	2-WAPD-HSS-140-143	Containment Operating Level	A	H	R
Reactor Coolant System	2-RC-HSS-100	On Spray Line in Overhead of -18' Level	I.	H	D
Reactor Coolant System	2-RC-HSS-101	Overhead of -18' Level Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-103	Overhead of -18' Level Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-102	In Prz. Relief Tank Room Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-106	In Prz. Relief Tank Room Between 2-RS-E-1C and 2-RS-E-1D	I	H	D
Reactor Coolant System	2-RC-HSS-104	On Spray Line in Overhead of -18' Level	I	H	D
Reactor Coolant System	2-RC-HSS-105	On Spray Line in Overhead of -18' Level	I	H	D

TABLE 4.17-2

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Reactor Coolant System	2-RC-HSS-108	On Spray line in Overhead of -18' Level	I	H	D
Reactor Coolant System	2-RC-HSS-107	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-109	On Spray Line in Overhead of -18' Level	I	H	D
Reactor Coolant System	2-RC-HSS-110	On Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-111	On Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-112	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-113	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-114	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-115	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-116	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-117	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-118	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-119	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-120	On Line Above Pressurizer	I	H	D
Reactor Coolant System	2-RC-HSS-121	On Line Above Pressurizer	I	H	D
Reactor Coolant System			I	N	R
Reactor Coolant System		There are two (2) large suppressors on each steam generator support ring and six (6) suppressors in the reactor coolant pump support structure	I	H	D
Base of Pressurizer	Four (4)	Large suppressors around base of pressurizer on 47'-4" Level	A	H	D
Safety Injection System	2-SI-HSS-(19) 2	At "C" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-20	At "C" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-21	At "C" Accumulator	I	H	D
Safety Injection System	2-SI-HSS-27	At "C" Accumulator	I	H	D
Safety Injection System	2-SI-HSS-25	At "A" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-26	At "A" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-24	At "A" Accumulator	I	H	D
Safety Injection System	2-SI-HSS-23	At "B" Accumulator	A	H	R

TABLE 4.17-2

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Safety Injection System	2-SI-HSS-24	At "B" Accumulator	A	H	R
Safety Injection System	2-SI-HSS-(22) 2	At "B" Accumulator	A	H	R
Residual Heat Removal System	2-RH-HSS-1	On 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-2	On 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-3	On 2-RH-P-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-4	On 2-RH-P-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-5	On 2-RH-E-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-6	On 2-RH-E-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-7	On 2-RH-E-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-8	On 2-RH-E-1A	I	H	D
Residual Heat Removal System	2-RH-HSS-9	Above 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-10	Above 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-11	Above 2-RH-P-1B	I	H	D
Residual Heat Removal System	2-RH-HSS-12	In Overhead Below RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-13	In Overhead Below RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-14	In Overhead Below RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-15	Near 2-RH-E-1B at -3'-11" Level	I	H	D
Residual Heat Removal System	2-RH-HSS-17	Back of Loop Room "A" Near Floor	I	H	R
Residual Heat Removal System	2-RH-HSS-18	Back of Loop Room "A" Near Floor	I	H	R
Residual Heat Removal System	2-RH-HSS-19	Under Loop Room "A" Near RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-20	Under Loop Room "A" Near RHR Flat	I	H	D
Residual Heat Removal System	2-RH-HSS-21	At Loop Room Level Over RHR Flat	I	H	R
Residual Heat Removal System	2-RH-HSS-22	At Loop Room Level Over RHR Flat	I	H	R
Residual Heat Removal System	2-RH-HSS-23	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-24	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-25	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-26	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-27	Below RHR Flat on 10" Line	I	H	D
Residual Heat Removal System	2-RH-HSS-28	On 12" Discharge from RHR -15-6 Elevation	I	H	D
Residual Heat Removal System	2-RH-HSS-29	On 12" Discharge from RHR -15-6 Elevation	I	H	D

TABLE 4.17-2

SYSTEM	DESIGNATION	LOCATION	ACCESSIBILITY CATEGORY	RADIATION CATEGORY	REMOVAL CATEGORY
Recirc. Spray System	2-RS-HSS-101	Outside of Crane Wall (10' El) on Southside	I	H	D
Recirc. Spray System	2-RS-HSS-102	Outside of Crane Wall (10' El) on Southside	I	H	D
Recirc. Spray System	2-RS-HSS-103	Outside of Crane Wall (10' El) on Southside	I	H	D
Recirc. Spray System	2-RS-HSS-104	Outside of Crane Wall (10' El) on Southside	I	H	D
Low Head Safety Injection System	2-SI-HSS-100	In Valve Pit on 10" Safety Injection Line	A	N	D
Low Head Safety Injection System	2-SI-HSS-101	In Valve Pit on 10" Safety Injection Line	A	N	D
Low Head Safety Injection System	2-SI-HSS-102A	In Valve Pit on 10" Safety Injection Line	A	N	D
Low Head Safety Injection System	2-SI-HSS-102B	In Valve Pit on 10" Safety Injection Line	A	N	D
Chemical & Volume Control System	2-CH-HSS-301	Unit No. 2 "C" Charging Pump Cube	A	N	D
Chemical & Volume Control System	2-CH-HSS-302	Unit No. 2 "C" Charging Pump Cube	A	N	D
Chemical & Volume Control System	2-CH-HSS-303	Unit No. 2 "C" Charging Pump Cube	A	N	D
Chemical & Volume Control System	2-CH-HSS-304	Unit No. 2 "C" Charging Pump Cube	A	N	D

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 25 TO LICENSES NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNITS NOS. 1 AND 2

DOCKETS NOS. 50-280 AND 50-281

Introduction

By letter dated March 22, 1976, Virginia Electric & Power Company (the licensee) requested amendments to Facility Operating Licenses Nos. DPR-32 and DPR-37. The purpose of the request is to incorporate provisions in the Surry Units Nos. 1 and 2 Technical Specifications related to limiting conditions for operation and surveillance of shock suppressors (snubbers). We made changes in the licensee's March 22, 1976, submittal after discussions with the licensee, by phone, on August 24, 1976.

Background

During the summer of 1973, inspections at two reactor facilities revealed a high incidence of inoperable hydraulic shock suppressors (snubbers) manufactured by Bergen Paterson Pipesupport Corporation. As a result of those findings, the Office of Inspection and Enforcement required each operating reactor licensee to immediately inspect all Bergen Paterson snubbers utilized on safety systems and to reinspect them 45 to 90 days after the initial inspection. Snubbers supplied by other manufacturers were to be inspected on a lower priority basis.

Since a long term solution to eliminate recurring failures was not immediately available, the Division of Reactor Licensing sent a letter dated October 2, 1973, to operating facilities (including Surry) utilizing Bergen Paterson snubbers specifying continuing surveillance requirements. By letter dated December 18, 1975, we requested that the licensee submit proposed Technical Specification requirements for hydraulic snubbers. On March 22, 1976, the licensee proposed Technical Specifications for hydraulic snubbers at the reactors. During our review of the proposed change, we found that certain modifications were necessary. These modifications were discussed with the licensee and have been incorporated into the proposed Technical Specifications.

Evaluation

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal movement during startup and shutdown.

The consequences of an inoperable snubber is an increase in the probability of structural damage to piping resulting from a seismic or other postulated event which initiates dynamic loads. It is, therefore, necessary that snubbers installed to protect safety system piping be operable during reactor operation and be inspected at appropriate intervals to assure their operability.

Examination of defective snubbers at reactor facilities has shown that the high incidence of failures observed in the summer of 1973 was caused by severe degradation of seal materials and subsequent leakage of the hydraulic fluid. The basic seal materials used in Bergen Paterson snubbers were two types of polyurethane; a millable gum polyester type containing plasticizers and an unadulterated molded type. Material tests performed at several laboratories (Reference 1) established that the millable gum polyurethane deteriorated rapidly under the temperature and moisture conditions present in many snubber locations. Although the molded polyurethane exhibited greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. The investigation indicated that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

An extensive seal replacement program has been carried out at many reactor facilities. Experience with ethylene propylene seals has been very good with no serious degradation reported thus far. Although the seal replacement program has significantly reduced the incidence of snubber failures, some failures continue to occur. These failures have generally been attributed to faulty snubber assembly and installation, loose fittings and connections and excessive pipe vibrations. The failures have been observed in both PWR's and BWR's and have not been limited to units manufactured by Bergen Paterson. Because of the continued incidence of snubber failures we have concluded that snubber operability and surveillance

(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

requirements should be incorporated into the Technical Specifications. We have further concluded that these requirements should be applied to all safety related snubbers, regardless of manufacturer, in all light water cooled reactor facilities.

The proposed Technical Specifications and Bases provide additional assurance of satisfactory snubber performance and reliability. The specifications require that snubbers be operable during reactor operation and prior to startup. Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repair or replacement of defective units before the reactor must be shut down. The corrective actions may be performed with the reactor in the hot shutdown condition; hot shutdown will reduce thermal cycling effects in the materials of the reactor coolant system. The licensee will be expected to commence repair or replacement of a failed snubber expeditiously. However, the allowance of 72 hours is consistent with that provided for other safety-related equipment and provides for remedial action to be taken in accordance with 10 CFR §50.36(c)(2). Failure of a pipe, piping system or major component would not necessarily result from the failure of a single snubber to operate as designed, and even a snubber devoid of hydraulic fluid would provide support for the pipe or component and reduce pipe motion. The likelihood of a seismic event or other initiating event occurring during the time allowed for repair or replacement is very small. Considering the large size and difficult access of some snubber units, repair or replacement in a shorter time period is not practical. Therefore, the 72 hour period provides a reasonable and realistic period for remedial action to be taken.

An inspection program is specified to provide additional assurance that the snubbers remain operable. The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The longest inspection interval allowed in the Technical Specifications after a record of no snubber failures has been established is nominally 18 months. Experience at operating facilities has shown that the required surveillance program should provide an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. Snubbers containing seal material which has not been demonstrated to be compatible with the operating environment are required to be inspected every 31 days until the compatibility is established or an appropriate seal change is completed.

To further increase the level of snubber reliability, the proposed Technical Specifications require functional tests once each refueling cycle. The tests will verify proper piston movement, lock up and bleed.

We have concluded that the proposed Technical Specifications, as modified, increase the probability of successful snubber performance, increase reactor safety and we therefore find them acceptable.

We made determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 14, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 25 to Facility Operating Licenses Nos. DPR-32 and DPR-37 issued to Virginia Electric & Power Company, which revised Technical Specifications for operation of the Surry Power Station Units Nos. 1 and 2, located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments incorporate provisions into the Technical Specifications related to limiting conditions for operation and surveillance of shock suppressors (snubbers).

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated March 22, 1976, (2) Amendments No. 25 to Licenses Nos. DPR-32 and DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street N.W., Washington, D. C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of September 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors