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OCT 15 1976

Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
ATTN: Mr. I. S. Mitchell, III
Vice President and Secretary
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated September 4, 1975, as supplemented by letters dated March 10, 1976, and August 2, 1976.

The amendment to the Technical Specifications adds a section which will incorporate a program for the surveillance of shock suppressors (snubbers) on safety related systems.

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

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 37
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

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DATE➤	10/12/76	10/12/76	10/12/76	10/14/76		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company and Oglethorpe Electric Membership Corporation (the licensees) dated September 4, 1975, as supplemented by letters dated March 10, 1976 and August 2, 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 a change 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

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 15, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 37

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace pages iv and viii with the attached revised pages. Add pages 3.6-10a through 3.6-10g, 3.6-14a through 3.6-14e, 3.6-31 and 3.6-32. (No change has been made on pages iii or vii.)

Amendment No. 37

LIST OF TABLES

LIST OF TABLES
(Concluded)

3.6.L. Shock Suppressors (Snubbers)

1. During all modes of operation except Shutdown (Cold) and Refuel, all snubbers which are required to protect the primary coolant system or any other safety related system or component shall be operable except as noted in 3.6.L.2 through 3.6.L.4 below. These safety related snubbers are listed in Table 3.6.L.

2. From and after the time that a snubber listed in Table 3.6.L is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.L.1 and 3.6.L.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

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

5. Snubbers may be added to safety related systems without prior License Amendment to Table 3.6.L provided that a revision to Table 3.6.L is included with the next License Amendment request.

4.6.L Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in Table 3.6.L.

1. All hydraulic snubbers 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 in two groups, "accessible" or "inaccessible" for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials 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.

4.6.L. Shock Suppressors (Snubbers)(cont'd

3. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.L.1, it shall be assumed that the facility had been on a 6 month inspection interval.

4. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximaetly 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 ten 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.

TABLE 3.6.L

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

A. SNUBBERS NOT ACCESSIBLE DURING REACTOR OPERATION

MAIN STEAM SEISMIC RESTRAINTS

SNUBBER #	SIZE (kips)	LOCATION	ELEVATION	*SHUTDOWN ACCESSIBILITY CODE
SS-1	20	72°	150'	RA
2	20	72°	150'	RA
3	10	0°	140'	RA
4	10	5°	128'	RA
5	10	0°	140'	RA
6	30	108°	150'	RA
7	20	90°	150'	RA
8	3	160°	151'	RA
9	3	95°	144'	RA
10	3	90°	142'	DA
11	10	100°	138'	DA
12	10	85°	144'	DA
13	10	82°	143'	RA
14	10	315°	150'	RA
15	10	90°	145'	RA
16	10	90°	144'	RA
17	3	170°	145'	N
18	3	160°	145'	N
19	3	160°	134'	RA
20	3	160°	145'	N
21	10	135°	158'	RA
22	10	160°	152'	RA
23	30	278°	150'	RA
24	50	270°	150'	RA
25	20	327°	150'	
26	3	275°	145'	RA
27	10	270°	138'	RA
28	10	270°	142'	DA
29	3	270°	140'	RA
31	3	270°	139'	RA
32	3	275°	125'	RA
33A	3	285°	123'	RA
33B	10	285°	123'	RA
34	3	280°	120'	RA
35	3	292°	120'	RA
36	20	307°	148'	RA
37	30	315°	150'	RA
38	10	9°	128'	RA
39	10	9°	128'	RA
40	20	0°	123'	RA
41	3	345°	155'	RA
42	3	347°	151'	RA
43	3	340°	155'	RA
44	3	343°	155'	RA
45	10	67°	118'	RA
46	3	67°	120'	RA

SNUBBER #	SIZE (kips)	LOCATION	ELEVATION	*SHUTDOWN ACCESSIBILITY CODE
		Nuclear Boiler Sys.		
MVVH-23	10	270°	144'	DA
24	10	270°	144'	DA
25	10	270°	144'	DA
27	10	225°	124'	RA
28	10	250°	128'	RA
29	10	250°	128'	RA
31	10	140°	149'	RA
32	10	140°	148'	RA
33	10	140°	145'	RA
35	10	90°	126'	RA
36	10	90°	126'	RA
37	10	90°	126'	RA
FDH-11	10	16°	147'(2)	RA
12	10	0°	147 (2)	RA
13	10	40°	148'	RA
14	10	75°	148'	RA
15	20	53°	148'	RA
16	10	79°	146'	RA
17	10	280°	148'	RA
18	10	281°	146'	RA
19	10	98°	150'	RA
21	3	210°&150°	165' (2)	RA
22	3	120°&240°	165' (2)	RA
23	10	60°	164'	RA
24	10	30°	167'	RA
25	10	330°	164'	RA
26	10	310°	167'	RA
DFDH-28	3	4'SR7-8'ERA	132'	RA
30	3	4'NR7-8'ERA	132'	RA
32	3	14°	132'	RA
36	3	0°	132'	RA
		RHR SYSTEM		
S-1	30	270°	141'	RA
2	30	270°	141'	RA
4	30	210°	141'	RA
5	30	240°	141'	RA
15	20	185°	139'	DA
SM-1	30	180°	134'	RA
2	10	180°	140'	DA
3	30	225°	146'	N
4	30	225°	146'	N
8	30	90°	146'	RA
RHRH-255	3	6'NR7-13'ERF	207'	DA
256	3	6'NR7-13'ERF	207'	DA
257	3	6'NR7'13'ERF	200'	DA
258	3	2'NR7-13'ERF	194'	DA

SNUBBER #	SIZE	LOCATION	ELEVATION	*SHUTDOWN ACCESSIBILITY CODE
SS-A1	35	315°	123'	RA
A2	50	315°	123'	RA
A3	50	315°	123'	RA
A4	50	310°	131'	RA
A5	50	320°	131'	RA
A6	50	315°	134'	RA
A7	21	15°	134'	RA
A8	35	10°	134'	RA
A9	11	270°	127'	RA
A10	--	270°	126'	RA
A11	--	275°	123'	RA
A13	35	270°	145'	RA
A14	50	270°	122'	RA
A12	35	270°	145'	RA
SS-B1	21	140°	120'	RA
B2	50	135°	123'	RA
B3	50	135°	123'	RA
B4	50	145°	131'	RA
B5	50	135°	131'	RA
B6	50	135°	137'	DA
B7	21	185°	140'	RA
B8	35	180°	140'	RA
B9	11	90°	126'	RA
B10	--	90°	125'	RA
B11	--	100°	121'	RA
B12	35	90°	145'	RA
B13	35	90°	145'	RA
B14	50	90°	116'	RA

B. SNUBBERS ACCESSIBLE DURING REACTOR OPERATION

CORE SPRAY SYSTEM

CSH-75	3	10'NR3-7'WRL	125'	RA
71	10	7'NR13-10'WRL	121'	RA
79	10	2'NR9-7'WRH	172'	RA

HPCI SYSTEM

HPCIH-9	20	13'SR1-6'ERG	88'	RA
13	50	7'SR1-2'WRL	94'	RA
12	10	5'NR3-3'ERF	123'	RA
HPSEH-2	10	12'NR2-10'WRL	92'	RA
8	10	6'NR2-4ERG	112'(2)	RA
12	10	5'NR3-3'ERF	123'8"	RA
13	10	4'NR3-3'ERF	123'6"(2)	RA
17	10	5'NR3-14'ERF	123'6"	RA
57	3	1'SR1-18'WRL	99 1/2'	RA
58	3	4'SR1-18'WRL	99'	RA
60	3	4'NR2-4'ERG	120'(2)	RA
61	10	3'NR5-11'ERH	123'	RA
62	10	3'NR5-11'ERH	123'	RA

SNUBBER #	SIZE	LOCATION	ELEVATION	*SHUTDOWN ACCESSIBILITY CODE
63	3	8'NR7-13'ERH	123'	RA
66	10	13'SR9-2'ERH	122'	RA
67	10	8'SR11-6'ERH	123'	RA
72	3	11'SR2-6'ERH	124'	RA
73	20	11'SR2-5'ERH	124'	RA
74	3	7'SR3-11'ERH	122'	RA
76	3	1'NR3-7'WRF	123'	RA
77	3	8'NR3-13'WRF	123'	RA
80	3	11'NR13-6'ERH	126'	RA
81	20	12'NR13-5'ERL	126'	RA
82	3	3'SR11-5'WRH	123'	RA
83	3	6'NR11-1'ERF	123'	RA
84	3	8'NR11-13'WRF	123'	RA
85	3	7'NR11-11'WRF	122'	RA
87	3	1'SR11-7'WRF	125'	RA
88	30	12'NR2-12'WRL	120'	RA
89	50	11'SR2-3'ERG	123'(2)	RA
90	50	5'NR3-13'ERF	125'	RA
91	50	5'SR3-8'ERF	123 1/2'(2)	RA
92	10	8'NR3-3'ERH	123'	RA
93	10	8'NR3-3'ERH	123'	RA
55	10	8'SR1-25'WRL	98'	RA
		RCIC SYSTEM		
RCSEH-2	10	2'SR11-15'ERA	101'	RA
20	3	1'NR11-14'ERA	98'	RA
21	3	5'ERA-7'SR9	120'	DA
23	10	14'SR7-9'ERA	122'(2)	DA
		RHR SYSTEM		
RHRH-184	20	17'ERH-3'SR3	90'	RA
185	10	17'ERH-3'SR3	91'	RA
186	3	12'ERH-0'SR3	88'(2)	RA
187	10	8'ERH-4'NR3	89'(2)	RA
188	10	5'WRL-5'SR5	90'	RA
189	10	8'WRL-2'NR5	90'	RA
192	3	18'ERH-9'NR3	123'	RA
193	3	17'ERH-9'NR3	123'	RA
195	3	18'WRL-6'SR3	125'	RA
196	3	18'WRL-6'SR3	120'	RA
199	3	3'WRL-10'SR3	123'	RA
202	3	7'ERH-6'SR7	110'	RA
207	3	15'ERH-10'SR7	151'(2)	RA
209	3	16'WRL-6'NR3	105'(2)	RA
210	3,10	16'WRL-9'NR3	108'(2)	RA
211	10	10'WRL-7'NR3	100'	RA
212	10	12'WRL-6'NR3	104'	RA
213	3	12'WRL-3'SR2	124'	RA
214	20	2'WRL-3'SR2	119'	RA
215	3	12'WRL-3'SR2	118'(2)	RA
216	10	16'WRL-11'NR3	122'	RA
217	10	19'WRL-11'NR3	121'	RA
218	3	13'ERH-9'SR2	116'	RA

SNUBBER #	SIZE (kips)	LOCATION	ELEVATION	*SHUTDOWN ACCESSIBLITY CODE
RHRH 220	10	12'WRL-4'SR7	126'	RA
221	10	10'WRL-2'SR7	122'	RA
222	10	10'WRL-2'SR7	120',122'(2)	RA
223	10	4'ERH-1/2'NR11	125'	RA
224	10	7'ERH-3'SR11	123'	RA
225	3	13'ERH-9'NR13	117"	RA
226	10	19'WRL-12'SR11	121'	RA
227	10	17'WRL-12'SR11	122'	RA
228	3	12'WRL-3'NR13	124'	RA
229	10	12'WRL-3'NR13	120'	RA
230	3	12'WRL-3'NR13	119'116' (2)	RA
231	3	17'WRL-9'SR11	104'(2)	RA
232	10	17'WRL-9'SR11	107'	RA
233	20	10'WRL-7'SR11	97'	RA
234	10	12'WRL-6'SR11	104'	RA
237	10	2'NR9-3'ERH	155'(2)	RA
238	3	2'NR9-4'WRH	159'(2)	RA
239	20	2'NR9--5'WRH	160'	RA
240	3	6'NR5-25'WRL	147'(2)	RA
242	10	3'SR4-4'WRH	150'	RA
244	10	10'SR11-4'ERF	121'(2)	RA
245	20	10'SR11-3'ERF	121'	RA
246	10	10'SR11-3'ERF	122'	RA
249	3	11'SR11-10'ERF	118'	RA
250	10	10'NR3-4'ERF	121'(2)	RA
251	20	10'NR3-3'ERF	119'	RA
252	10	10'NR3-3'ERF	121'	RA
254	3	11'NR3-10'ERF	118'	RA
279	10	3'SR5-3'WRL	116'(2)	RA
282	10	1/2'SR4-7'ERG	121'	RA
286	3	3'SR2-17'WRL	116'(2)	RA
288	10	3'NR9-3'WRL	116'(2)	RA
292	10	4'SR2-6ERH	124'	RA
299	10	4'NR13-6ERH	122'	RA
305	10	5'NR3-4'ERH	123'	RA
306	10	3'NR3-7'ERH	121'(2)	RA
307	10	10'SR5-17'ERH	132'	RA
309	3	9'SR11-18'ERH	115'	RA
310	3	9'SR11-18'ERH	124'	RA
312	3	6'NR11-18'WRL	124'	RA
313	3	6'NR11-18'WRL	125'	RA
316	3	10'NR11-4'WRL	123'	RA
319	10	9'NR9-17'ERH	136'(2)	RA
320	20	4'NR11-17'ERH	90'	RA
321	10	3'NR11-15'ERH	91'	RA
322	3	3'SR11-12'ERH	89'(2)	RA
323	10	5'SR11-9'ERH	89'(2)	RA
324	10	5'NR9-5'WRL	91'	RA
325	10	2'SR9-9'WRL	90'	RA
332	3	2'SR6-2'WRH	190'	RA
344	10	1/2'NR10-1'WRH	122'	RA
348	3	3'NR13-17'WRL	117'(2)	RA
399	3	6'NR5-6'ERH	148'	RA
400	10	6'NR5-1'ERH	148'	RA

*SHUTDOWN ACCESSIBLITY CODE RA-Readily Accessible
DA-Accessible with difficulty
N-Not Accessible

3.6.L and 4.6.L Shock Suppressors (Snubbers)

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 hydraulic snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold 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 3.6.L.4 prohibits startup with inoperable snubbers.

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 inpsection. 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 material are available, primarily ethylene propylene

3.6.L. and 4.6.L. Shock Suppressors (cont'd)

compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

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. Snubbers in high radiation areas 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 impracticality of testing such large units.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-321

Introduction and Discussion

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.

In order to provide a long term solution to eliminate recurring snubber failures, the NRC staff developed model technical specifications which provide additional assurance for satisfactory snubber operation. These model technical specifications reflect the accumulated industry-wide experience with respect to hydraulic snubber performance and reliability.

By letter dated July 8, 1975, we sent Georgia Power Company (GPC) a copy of the model technical specifications for hydraulic snubbers and requested that they adapt the model to Edwin I. Hatch Nuclear Plant Unit No. 1 (HNP-1). On September 4, 1975, GPC submitted an application for a license amendment to incorporate the model technical specifications into Appendix A of Facility Operating License No. DPR-57 for HNP-1. Subsequently, as a result of industry comments and further consideration by the NRC staff, the model technical specifications for shock suppressors were revised to provide clarification of the operability and surveillance requirements. By letter dated December 24, 1975, we requested GPC to submit an application for license amendment which would reflect the changes to the model technical specifications. GPC's submittals of March 10, 1976 and August 2, 1976 were responsive to our request.

Evaluation

Hydraulic shock suppressors (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 incident 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 PWRs and BWRs 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 requirements should be incorporated into the Technical Specifications. We have further concluded that these requirements should be applied to all safety related hydraulic snubbers, regardless of manufacturer, in all light water cooled reactor facilities.

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

The proposed Technical Specifications and Bases provide additional assurance of satisfactory snubbers 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 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 and 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 the 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 Technical Specifications require functional tests once each refueling cycle. The tests will verify proper piston movement, lock up and bleed.

Summary

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

Environmental Consideration

We have determined that the amendment does 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 amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes 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 changes 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 15, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 37 to Facility Operating License No. DPR-57 issued to Georgia Power Company and Oglethorpe Electric Membership Corporation, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance. The amendment to the Technical Specifications adds a section which will incorporate a program for the surveillance of shock suppressors (snubbers) on safety related systems.

The application for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated September 4, 1975, as supplemented by letters dated March 10, 1976 and August 2, 1976, (2) Amendment No. 37 to License No. DPR-57 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 Appling County Public Library, Parker Street, Baxley, Georgia 31513.

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 15 day of October 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors