Distribution:

AEC PDR	VStello
Local PDR	MJinks (4)
Docket (2)	JKnight
PWR-1 Reading	DSullivan
PWR-1 File	RZavadoski
RO (3)	ADromerick
RCDeYoung	DVassallo
JMHendrie	JLee
DSkovholt	ACRS (16)
RKlecker	WMiller

AUG 1 7 1973

Docket Nos. 50-280
and 50-281

License Nos. DPR-32
and DPR-37

Change No. 10

Mr. Stanley Ragone
Vice President
Virginia Electric and
Power Company
P. O. Box 26666
Richmond, Virginia 23261

Dear Mr. Ragone:

Your letter dated July 17, 1973 enclosed proposed changes in Section 2.1.C, "Safety Limit, Reactor Core", and Section 4.4, "Containment Tests" of the Technical Specifications for Facility Operating Licenses Nos. DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. You proposed deletion of Section 2.1.C which limits the reactor thermal power to 1220 megawatts thermal until the results of the environmental qualification tests performed on the recirculation spray pump motors inside containment have been evaluated and approved by the AEC. In our letter of August 21, 1972, we stated that the requirements of Technical Specification TS 2.1.C were considered satisfied for a one year period. You also proposed that Type A periodic containment leak tests will be performed in accordance with the peak pressure test program as defined in Appendix J of 10 CFR Part 50.

We have reviewed the second addendum to the "Topical Report on G. E. Vertical Induction Motors Inside Containment Recirculation Spray Motors for Surry Units 1 and 2", dated June 12, 1973, which you submitted in support of your proposed change to delete Section 2.1.C of the Technical Specifications. The qualification program included exposure to nuclear radiation to a dose exceeding that anticipated from normal and post-accident service, a vibration test simulating the response of the Surry Power Station to the design basis earthquake, and a steam/chemical-spray exposure simulating the in-containment environment following a loss-of-coolant accident. On the basis of our review of the test results, we have determined that the prototype recirculation spray pump motor has satisfactorily performed throughout all phases of a qualification test program for service inside the containments of Surry Power Station Units 1 and 2. We, therefore, conclude that the tested prototype motor has been demonstrated to be qualified in all respects for the intended service at Surry Power Station Units 1 and 2. With this approval, the limitation of Section 2.1.C is no longer required and, as you proposed, is hereby deleted. The subdivisions following this provision are appropriately renumbered.

We have also reviewed the report "Reactor Containment Building Integrated Leak Rate Test - Surry Power Station Unit 2," which supports your proposed change that Type A periodic containment leak tests be performed as defined in Appendix J of 10 CFR Part 50. This change deletes the alternative method of performing containment testing at pressures less than 39.2 psig. On the basis of our review, we have determined that the proposed change complies with Appendix J of 10 CFR Part 50 and is acceptable.

We have concluded that the proposed changes do not involve significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specification changes outlined above are hereby authorized. To effect these changes, replace pages TS 2.1-2, TS 2.1-6, and TS 4.4-2 of the Technical Specifications of Facility Operating Licenses Nos. DPR-32 and DPR-37 with the revised pages (designated as Change No. 10 on the bottom of the page) TS 2.1-2, TS 2.1-6, and TS 4.4-2 enclosed.

Sincerely,

Original signed by R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

FOR CONCURRENCES SEE PREVIOUS YELLOW

cc: George D. Gibson, Esq.
Hunton, Williams, Gay,
and Gibson
P. O. Box 1535
Richmond, Virginia 23213

bcc: J. R. Buchanan, ORNL
Thomas B. Abernathy, DTIE

	AD/PWRs	RO
	RCDeYoung	
	8/ /73	8/ /73

OFFICE ▶	PWR-1	PWR-1	TR	TR	TR	OGC
SURNAME ▶	ADromerick:cls	DBVassallo	DSullivan	JKnight	RZavadoski	
DATE ▶	8/ /73	8/ /73	8/ /73	8/ /73	8/ /73	8/17/73

Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43—16—81465-1 445-678

We have also reviewed the report "Reactor Containment Building Integrated Leak Rate Test - Surry Power Station Unit 2" which supports your proprosed change that Type A periodic containment leak tests be performed as defined in Appendix J of 10 CFR Part 50. On the basis of our review, we have determined that the proposed change complies with Appendix J of 10 CFR Part 50 and is acceptable.

We have concluded that the proposed changes do not involve significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specification changes outlined above are hereby authorized. To effect these changes, replace pages TS 2.1-2, TS 2.1-6, and TS 4.4-2 of the Technical Specifications of Facility Operating Licenses Nos. DPR-32 and DPR-37 with the revised pages (designated as Change No. 11 on the bottom of the page) TS 2.1-2, TS 2.1-6, and TS 4.4-2 enclosed.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

cc: George D. Gibson, Esq.
Hunton, Williams, Gay,
and Gibson
P. O. Box 1535
Richmond, Virginia 23213

bcc: J. R. Buchanan, ORNL
Thomas B. Abernathy, DTIE

CONCURRENCE BY TELEPHONE - SNIEZAK ACTING FOR KEPPLER
GKD - 8-16/73

	AD/PWRs	RO
	RCDeYoung	
	8/17/73	8/ /73

OFFICE ▶	PWR-1	PWR-1	TR	TR	TR	OGC
SURNAME ▶	ADromerick:cls	DBVassallo	DSullivan	JKnight	RZavadoski	
DATE ▶	8/15/73	8/16/73	8/15/73	8/16/73	8/15/73	8/ /73

10,000 effective
full prime lawns

8/17/73
D.H.

B. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds the limit on TS Figure 2.1-4 9

C. The fuel residence time shall be presently limited to 10,000 effective full power hours (EFPH) under design operating conditions provided the primary system pressure is reduced to 2000 psia by 3500 EFPH. 6 7 10

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio (DNBR) during steady state operation, normal operational transients and anticipated transients, is limited to 1.30. A DNBR

The curve of TS Figure 2.1-4 represents the fuel overpower design limit as a function of burnup. This limit is the fuel melting temperature or a linear heat rate of 21.1 kw/ft, whichever is more restrictive. Additional peaking factors to account for local peaking due to fuel rod axial 9 gaps and reduction in fuel pellet stack length have been included in the calculation of this curve.

The fuel residence time for Cycle 1 is limited to 10,000 EFPH to assure no fuel clad flattening without prior review by the Regulatory staff. If residence time of the present core will exceed 10,000 hours under design 6 9 operating conditions, the assumption of clad flattening is presently required. Prior to 10,000 hours, the licensee may provide the additional analyses required for operation beyond 10,000 EFPH.

References

(1) FSAR Section 3.4

(2) FSAR Section 3.3

(3) FSAR Section 14.2

b. The leakage rate test will be performed at a pressure of 39.2 psig (P_p). 3 10

c. The measured leakage rate L_{pm} shall not exceed the design basis accident leakage rate (La) of 0.1 weight percent per 24 hours at pressure P_p.

10

2. Type B and C tests will be performed at a pressure of 39.2 psig (P_p) in accordance with the provisions of Appendix J, section III. B. and C.

C. Acceptance Criteria

Type A, B, and C tests will be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Sections III.A.5, III.A.7, III.B.3, and III.C.3 are met.

D. Retest Schedule

The retest schedules for Type A, B, and C tests will be in accordance with Section III-D of Appendix J.

E. Inspection and Reporting of Tests

Inspection and reporting of tests will be in accordance with Section V of Appendix J.