ENCLOSURE 4

PROPOSED CHANGES

MARKED UP PAGES FOR PROPOSED CHANGES TO

APPENDIX A TECHNICAL SPECIFICATIONS OF

FACILITY OPERATING LICENSES

DPR-39 AND DPR-48

FOR LICENSE AMENDMENT REQUEST NO. 92-03

ZNLD/2110/26

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1.28 <u>OPERATING</u>

OPERATING is defined as performing the intended function in the intended manner.

1.29 <u>OPERATING CYCLE</u>

The OPERATING CYCLE shall be the interval between the end of one major refueling outage and the end of the next subsequent major refueling outage per unit.

1.30 <u>OPERATIONAL MODE - MODE</u>

An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1, when fuel assemblies are present in the reactor vessel.

1.31 <u>PHYSICS TESTS</u>

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

a

1.32 <u>PRESSURE BOUNDARY LEAKAGE</u>

PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in the Reactor Coolant System component body, pipe wall, or vessel wall.

Insert 1.31a

1.33 <u>PROCESS CONTROL PROGRAM (PCP)</u>

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes will be accomplished in such a way as to assure compliance with 10 CFR parts 20, 61 and 71, and Federal and State regulations and other requirements governing the shipment and disposal of radioactive waste.

1.34 <u>PROTECTION LOGIC CHANNEL</u>

A PROTECTION LOGIC CHANNEL shall be an arrangement of relays, contacts or other components which operate in response to INSTRUMENT CHANNEL outputs to produce a decision output. The decision output is the initiation of a protective action signal. At the system level, the decision output is the operation of a sufficient numer of ACTUATION DEVICES and the associated ACTUATED EQUIPMENT as required to place or restore the Nuclear Steam Supply System to a design safe state. The channel is deemed to include the ACTUATION DEVICES.

1.35 <u>PROTECTION SYSTEM</u>

The PROTECTION SYSTEM shall consist of both the Reactor Protection System and the Engineered Safeguards System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through ACTUATION DEVICES) which are required to operate in order to place or restore the Nuclear Steam Supply System to a design safe state.

MOVED TO PAGE 5a

Insert 1.31a

1.31a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period in accordance with Specification 6.6.1.G. Plant operation within these operating limits is addressed in Specification 3.3.2.A.

LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURIZATION AND SYSTEM INTEGRITY

A. Heatup and Cooldown

and heatup and cooldown rates

The Reactor Coolant System temperature and pressure (with the exception of the pressurizer) shall be ~~limited in accordance with the limit lines shown in Figures 3.3.2-1 and 3.3.2-2~~ during heatup, cooldown and inservice leak and hydrostatic testing. ~~with:~~

maintained within the limits specified in the Pressure and Temperature Limits Report (PTLR)

1. a. A maximum heatup rate of 20°F/hr applicable up to and including 180°F RCS indicated temperature. A maximum heatup rate of 60°F/hr applicable for RCS indicated temperatures greater than 180°F.

   b. A maximum cooldown of 100°F in any 1 hour period.

   c. A maximum temperature change of $\leq$10°F in any 1 hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

specified in the PTLR

ACTION: With any of the ~~above~~ limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least MODE 3 within the next 5 hours and reduce RCS T_{AVG} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURIZATION AND SYSTEM INTEGRITY

A. The reactor coolant temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

Specified in the PTLR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.2 (Continued)	4.3.2
B. The Reactor Coolant System pressure and temperature limits specified in the PTLR ~~limit lines shown in Figures 3.3.2-1 and 3.3.2-2~~ shall be recalculated periodically as required, based on results from the material surveillance program.	B. Not Applicable
C. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the primary and secondary coolant is below 70°F.	C. Not Applicable
D. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.	D. Not Applicable
E. Hydrostatic Testing 1. System inservice leak and hydrotests shall be performed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI and applicable addenda; except as stated in Specification 4.3.4.C.1.	E. Not Applicable

LIMITING CONDITION FOR OPERATION

3.3.2.G. Low Temperature Overpressure Protection (Continued)

b. The Reactor Coolant System (RCS) pressure shall be less than 100 psig, and the pressurizer level less than 25%, or

c. The RCS is depressurized and one PORV and it's isolation valve are open.

shall be capable of

2. a. A maximum of one* charging pump ~~or safety injection pump, aligned for~~ injection into the RCS, ~~and no accumulators shall be OPERABLE~~.

b. No safety injection pumps shall be capable of injection into the RCS.

c. No accumulators shall be capable of injection into the RCS.

* For short durations of time during pump switchover, two charging pumps may be ~~OPERABLE~~ for the purpose of maintaining seal injection flow to the reactor coolant pumps.

capable of injection into the RCS

SURVEILLANCE REQUIREMENT

4.3.2.G. Low Temperature Overpressure Protection (Continued)

4. Verifying each PORV's isolation valve is open at least once per shift when this method is being used for low temperature overpressure protection.

5. Testing pursuant to Specification 4.0.5.

b. The RCS pressure shall be verified to be less than 100 psig, and pressurizer level shall be verified to be less than 25% at least once per shift, when this method is being used for low temperature overpressure protection.

c. Verifying one PORV and it's isolation valve are open at least once per shift, when this method is being used for low temperature overpressure protection.

two / three charging

2. At least ~~four~~ of the ~~five~~ pumps ~~(charging pumps and safety injection pumps)~~, and all accumulators, shall be verified to be incapable of injecting into the RCS prior to entering a condition in which they are required to be ~~inoperable~~, and at least once per shift thereafter while they are required to be ~~inoperable~~. incapable of injection into the RCS.

incapable of injection into the RCS

safety injection pumps and

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.2.G. Low Temperature Overpressure Protection (Continued)

3. When starting a reactor coolant pump, when no reactor coolant pumps are running, the temperature in the secondary side of the steam generator in the loop in which the reactor coolant pump is to be started shall be less than 50°F higher than the RCS temperature.

APPLICABILITY: Mode 4 when the temperature of any RCS cold leg is less than or equal to 320°F ~~250°F~~, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days, or within the next 24 hours either;

- Depressurize the RCS to less than 100 psig and lower pressurizer level to less than 25%, or

- Depressurize the RCS and open at least one PORV and it's block valve.

4.3.2.G. Low Temperature Overpressure Protection (Continued)

3. Not applicable.

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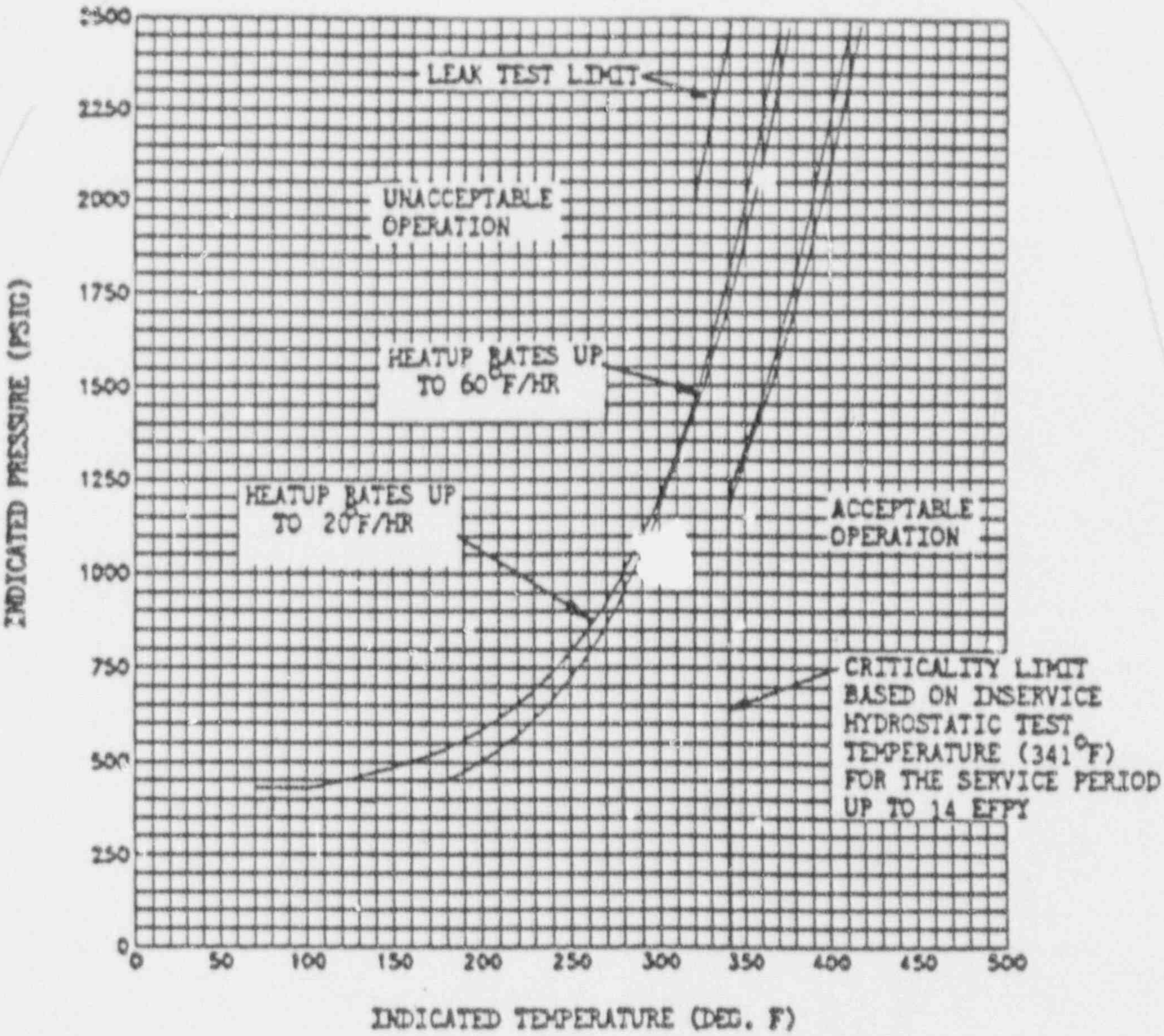

The 20°F/hr heatup rate is applicable for all RCS indicated temperatures. The 60°F/hr heatup rate is applicable for RCS indicated temperatures greater than 180°F.

Zion 1 and 2 Reactor Coolant System Heatup Limitations applicable for up to 14 EFPY and Heatup Rates up to 20°F/hr and 60°F/hr. Curves contain margins of 10°F and 60 psig for possible instrument errors.

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS

Figure 3.3.2-1

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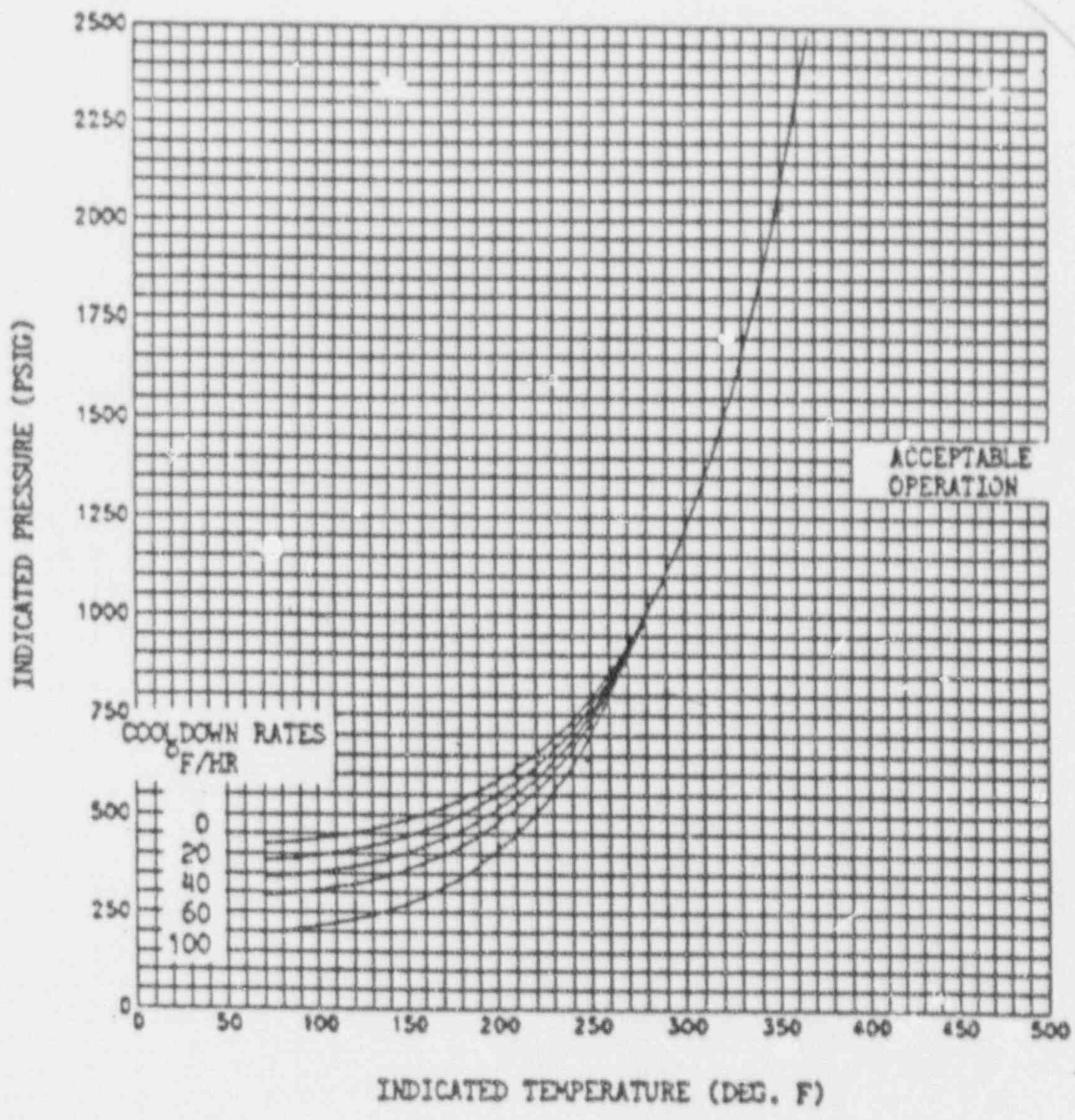

Zion 1 and 2 Reactor Coolant System Cooldown Limitations applicable for up to 14 EFPY and Cooldown Rates up to 100°F/hr. Curves contain margins of 10°F and 60 psig for possible instrument errors.

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS

Figure 3.3.2-2

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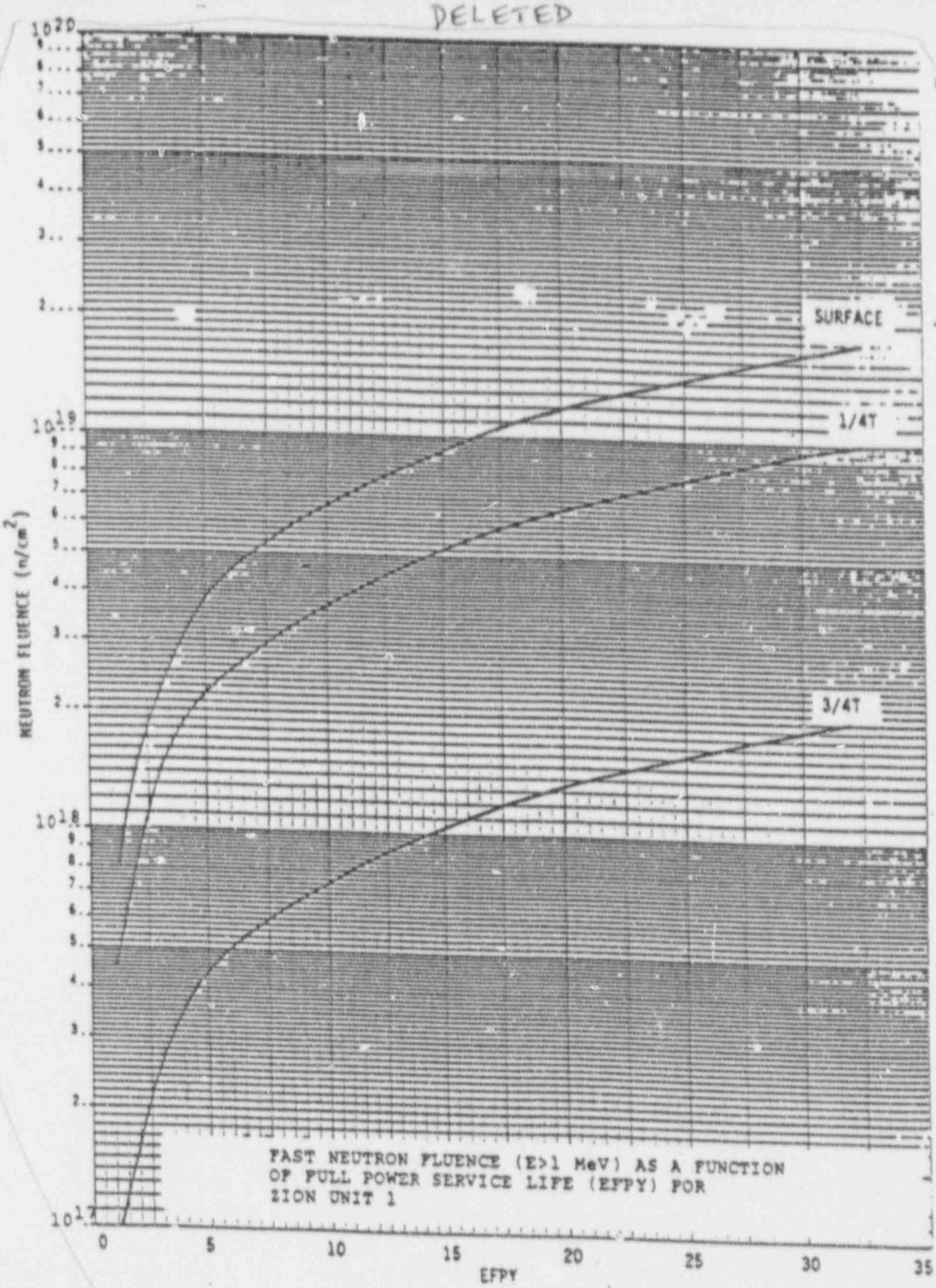

Figure 3.3.2-3

Amendment Nos. ~~101 & 91~~

~~10580/10590~~
~~0657A~~

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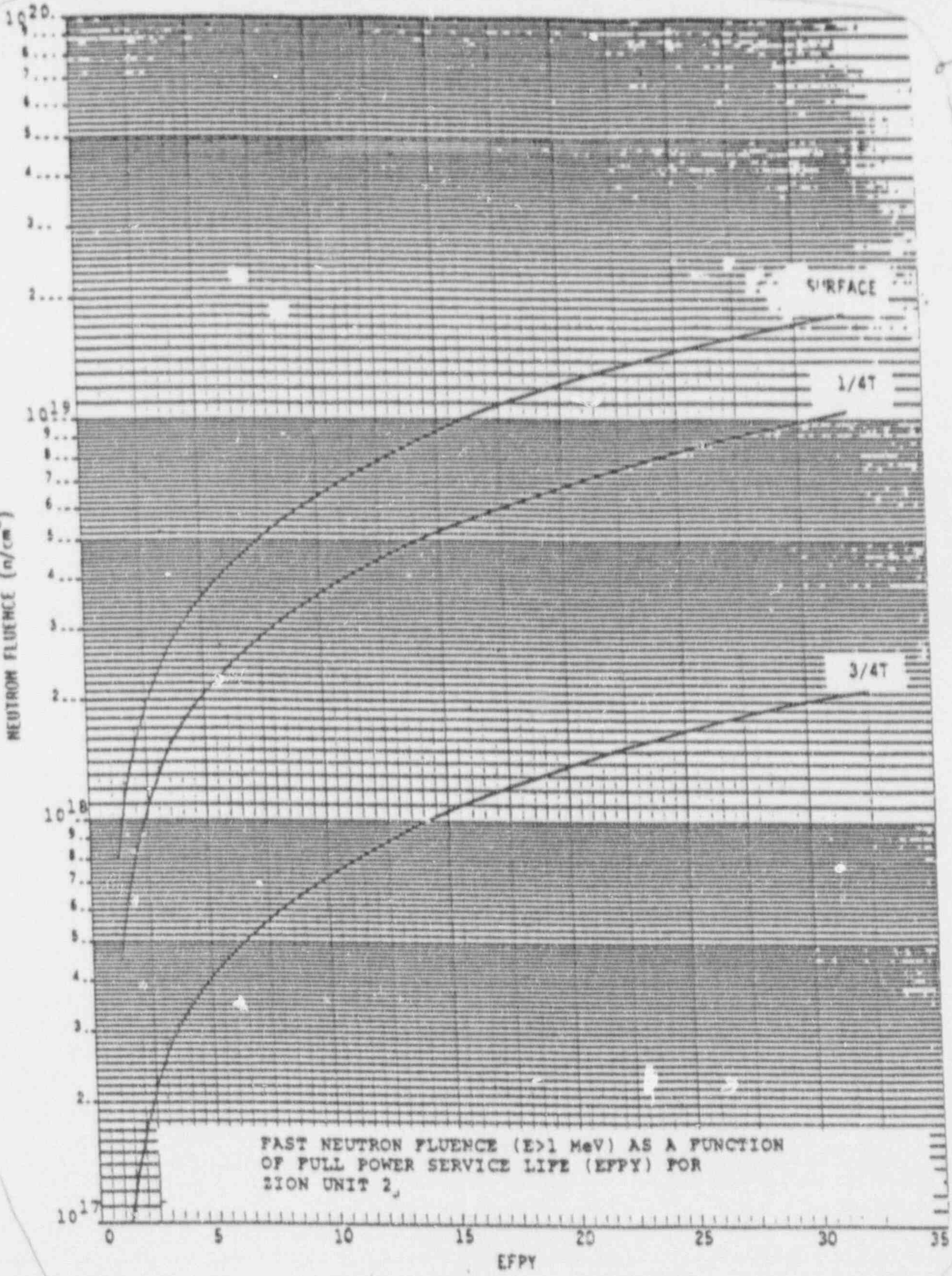

Figure 3.3.2-4

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COMPONENT	HEAT NO.	MATERIAL TYPE	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	50 FT-LB/35 MIL TEMP[a] (°F)	RT_{NDT} (°F)	UMS[a] USE (FT-LB)
Closure Head Dome	B9094-2	A533B, CL.1	.14	.55	.012	20	90	30	77
Closure Head Segment	C5086-1	" "	.09	.54	.014	10	32	10	103
Closure Head Segment	B8793-3	" "	.09	.52	.012	10	53	10	96
Closure Head Flange	123K323	A508, CL.2	-	.69	.010	55[a]	26	55	96
Vessel Flange	123V236	" "	.06	.68	.004	7[a]	-2	7	131
Inlet Nozzle	ZT3600-1	" "	.12	.68	.009	60[a]	27	60	79
Inlet Nozzle	ZT3600-2	" "	.11	.67	.009	60[a]	41	60	82
Inlet Nozzle	ZT3592-1	" "	.10	.66	.011	60[a]	103	60	77
Inlet Nozzle	ZT3592-2	" "	.11	.67	.010	60[a]	51	60	62
Outlet Nozzle	ZT3592-3	" "	.11	.68	.010	60[a]	60	60	86
Outlet Nozzle	ZT3592-4	" "	.11	.68	.009	46[a]	16	46	85
Outlet Nozzle	ZT3600-3	" "	.10	.67	.011	60[a]	52	60	82
Outlet Nozzle	ZT3600-4	" "	.11	.68	.011	60[a]	46	60	>63
Upper Nozzle Shell	123V426	" "	.06	.75	.005	10	43	10	115
Lower Nozzle Shell	ZV3300	" "	.06	.83	.008	20	72	20	87
Inter. Shell	C3795-2	A533B, CL.1	.12	.49	.010	-10	70	10	85
Inter. Shell	B7835-1	" "	.12	.49	.010	-20	65 (Actual)	5	115 (Actual)
Lower Shell	B7823-1	" "	.13	.48	.013	-20	56 (Actual)	-4	115.5 (Actua
Lower Shell	C3799-2	" "	.15	.50	.010	-20	80 (Actual)	20	116 (Actual)
Bottom Head Trans. Ring	ZV3779	A508, CL.2	.09	.71	.010	10	60	10	92
Bottom Head Dome	B7777-1	A533B, CL.1	-	.62	.015	-30	33	-27	84
Inter. to Lower Shell Girth Weld Seam	WF70[b]	SAW	.32	.56	.017	0[a]	--	0	--
Inter. Shell Long. Weld Seam	WF4[c]	SAW	.29	.55	.013	0[a]	--	0	--
Inter. Shell Long. Weld Seam	WF8[d]	SAW	.29	.55	.013	0[a]	--	0	--
Lower Shell Long. Weld Seam	WF8[d]	SAW	.29	.55	.013	0[a]	--	0	--
Nozzle to Inter. Shell	WF154[e]	SAW	.31	.59	.013	0[a]	--	0	--
Girth Weld Seam	SA1769[f]	SAW	.26	.60	.019	0[a]	--	0	--

(a) Estimated using Methods of U.S.NRC NUREG-0800, Branch Technical Position MTEB 5-2, July, 1981
(b) Weld Wire Heat No. 72105 and Linde 80 Flux Lot No. 8669
(c) Weld Wire Heat No. 8T1762 and Linde 80 Flux Lot No. 8597
(d) Weld Wire Heat No. 8T1762 and Linde 80 Flux Lot No. 8632
(e) Weld Wire Heat No. 406L44 and Linde 80 Flux Lot No. 8720
(f) Weld Wire Heat No. 71249 and Linde 80 Flux Lot No. 8738

ZION UNIT 1 REACTOR VESSEL TOUGHNESS DATA
TABLE 3.3.2-1

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COMPONENT	HEAT NO.	MATERIAL TYPE	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	50 FT-LB/35 MIL TEMP[a] (°F)	RT_{NDT} (°F)	ANS[a] USE (FT-LB)
Closure Head Dome	B9094-1	A533B, CL.1	.14	.55	.012	-20	71	11	72
Closure Head Segment	C4787-1A	" "	.13	.62	.008	0	30	0	88
Closure Head Segment	C5086-2	" "	.09	.54	.014	30	45	30	88
Closure Head Flange	124W609	A508, CL.2	.08	.70	.010	12[a]	-13	12	105
Vessel Flange	2V-965	" "	.12	.74	.010	60[a]	33	60	79
Inlet Nozzle	ZT4007-2	" "	.11	.70	.009	48[a]	32	48	>78
Inlet Nozzle	7T3885-1	" "	.11	.58	.012	60[a]	43	60	82
Inlet Nozzle	ZT3885	" "	.11	.56	.011	43[a]	31	43	78
Inlet Nozzle	ZT3885	" "	.11	.56	.012	60[a]	48	60	>84
Outlet Nozzle	ZV3930	" "	.12	.66	.010	58[a]	20	58	93
Outlet Nozzle	ZV3930	" "	.11	.65	.011	48[a]	15	48	>80
Outlet Nozzle	ZV3930	" "	.12	.67	.011	55[a]	28	55	84
Outlet Nozzle	ZT3885-4	" "	.11	.57	.013	60[a]	41	60	>61
Upper Nozzle Shell	ZD3940	A508, CL.2	.07	.62	.008	10	65	10	106
Lower Nozzle Shell	ZV3855	" "	.09	.66	.008	10	70	10	>80
Lower Shell	B8029-1	A533B, CL.1	.12	.51	.010	-10	82	22	81
Lower Shell	C4007-1	" "	.12	.53	.010	10	82 (Actual)	22	94 (Actual)
Inter. Shell	B8006-1	" "	.12	.54	.010	10	68	10	89
Inter. Shell	B8040-1	" "	.14	.52	.008	-10	62	2	92
Bottom Head Trans. Ring	3V-433	A508, CL.2	.09	.76	.010	0	43	0	87
Bottom Head Dome	C4007-2	A533B, CL.1	.12	.53	.010	-20	60	0	72
Inter. to Lower Shell Girth Weld Seam	SA1769[b]	SAW	.26	.60	.019	0[a]	--	0	--
Lower Shell Long. Weld Seam	WF29[c]	SAW	.23	.63	.019	0[a]	--	0	--
Inter. Shell Long. Weld Seam	WF70[d]	SAW	.32	.56	.017	0[a]	--	0	--
Nozzle to Inter. Shell Girth Weld Seam	WF200[e]	SAW	.24	.63	.010	0[a]	--	0	--

(a) Estimated using Methods of U.S.NRC NUREG-0800, Branch Technical Position MTEB 5-2, July, 1981
(b) Weld Wire Heat No. 71249 and Linde 80 Flux Lot No. 8738
(c) Weld Wire Heat No. 72102 and Linde 80 Flux Lot No. 8650
(d) Weld Wire Heat No. 72105 and Linde 80 Flux Lot No. 8669
(e) Weld Wire Heat No. 821T44 and Linde 80 Flux Lot No. 8773

ZION UNIT 2 REACTOR VESSEL TOUGHNESS DATA
TABLE 3.3.2-2

Bas "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 & 2"

3.3.2 & 4.3.2 FRACTURE TOUGHNESS PROPERTIES

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and 10CFR50 Appendix G.

1986 Edition of

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ~~ASTM E185-73, and in accordance with additional reactor vessel requirements~~. These properties are then evaluated in accordance with Appendix G of the ~~1976 Summer Addenda to~~ Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-~~7924-A~~ 13406, ~~"Basis for Heatup and Cooldown Limit Curves"~~, ~~April 1975~~ July 1992.

adjusted 14 adjusted This adjusted adjusted 14

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, (RT_{NDT}) at the end of ~~15~~ 14 effective full power years (EFPY) of service life. The ~~15~~ 14 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. ~~The selection of such a~~ This limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

WCAP-13406 "HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION FOR ZION UNITS 1 & 2" or based on credible surveillance data Calculated

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in ~~Tables 3.3.2-1 and 3.3.2-2~~. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. ~~Therefore, an~~ adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be ~~predicted~~ Calculated using Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves ~~of Figures 3.3.2-1 and 3.3.2-2~~ include ~~predicted~~ adjustments for this shift in RT_{NDT} at the end of ~~15~~ 14 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Specified in the Pressure and Temperature Limits Report (PTLR)

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-~~7924-A~~ 13406.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the adjusted nil-ductility reference temperature, (adjusted RT_{NDT}) is used, ~~and this~~ which includes the initial RT_{NDT}, the radiation-induced shift, (ΔRT_{NDT}) corresponding to the end of the period for which heatup and cooldown curves are generated, and a margin factor.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR} for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = ~~constant~~ reference stress intensity factor provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup and cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for adjusted RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

the PTLR ~~Figures 3.3.2-1 and 3.3.2-2~~ defines limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

The leak test limit curve shown on the heatup curves in the PTLR ~~(Fig. 3.3.2-1)~~ represent minimum temperature requirements at the leak test pressure specified by ASME Section III and the NRC Standard Review Plan NUREG-0800.

Allowable combinations of pressure and temperatures in the PTLR for specified temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, 1976 Summer Addenda.

2. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves", April 1975.

3. ASME Boiler and Pressure Vessel Code, Section III, N-331.

4. ASME Boiler and Pressure Vessel Code, Section III, N-415.

5. FSAR, Chapter 4.3.

6. WCAP-8724, "ASME III, Appendix G Analysis of the Commonwealth Edison Company Zion Unit 1 Reactor Vessel".

7. WCAP-8727, "ASME III, Appendix G Analysis of the Commonwealth Edison Company Zion Unit 2 Reactor Vessel".

8. WCAP-10677, "Adjoint Flux Program for Zion Units 1 and 2".

9. Regulatory Guide 1.99 Revision 2.

10. Code of Federal Regulations, 10CFR50 Appendix G, "Fracture Toughness Requirements."

11. WCAP-~~14247~~ 13406, "Heatup and Cooldown Limit Curves for ~~the Commonwealth Edison Company~~ Normal Operation for Zion Units 1 ~~and~~ & 2 ~~Reactor Vessel~~", July 1992.

12. WCAP-10962, Rev. 2, "Zion Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations", December 1990.

13. CWE-86-563, "Low Temperature Overpressure Protection System Setpoin~~g~~t Analysis", August [illegible], 1986.

14. CWE-865-588, Low Temperature Overpressure Protection System Setpoin~~g~~t Analysis Extensions" October 27, 1986.

Bases: Low Temperature Overpressure Protection

3.~~2~~3.2.G & 4.~~2~~3.2.G There are 3 means of protecting the RCS from overpressurization by a pressure transient at low temperatures (below ~~250°F~~ 320°F). The first type of protection is ensured by the operation and surveillance of the power operated relief valves with a lift setting of 435 psig. A single power operated relief valve (PORV) will relieve a pressure transient caused by 1) a mass addition into a solid RCS from a charging pump or 2) a heat input based on a reactor coolant pump being started in an idle RCS and circulating water into a steam generator whose temperature is 50°F greater than the RCS temperature. (1)

The second means of protection is ensured by a PORV being open. It will have the same relieving capabilities as mentioned above.

The third means of protection limits the pressurizer level to 25% and the pressurizer pressure to 100 psig. A pressure transient caused by the inadvertent mass addition from a charging pump running for 10 minutes will be relieved by the large gas volume and low pressure present in the pressurizer as mentioned above. Maintaining the pressurizer level below 25% will also make the high pressurizer level deviation alarm available to the operator during a mass addition accident.

In the event that a single PORV becomes inoperable, the repair period of 7 days is based on allowing sufficient time to effect repairs using safe and proper procedures and upon the operability of the redundant PORV. The 24 hour time period to reach the restrictive conditions in the pressurizer provides sufficient time to meet these conditions.

In the event that both PORV's become inoperable, the condition is more serious than for a single inoperable PORV, therefore every attempt should be made to depressurize the RCS in a controlled manner as rapidly as possible. The 16 hour time period to reach the restrictive conditions in the pressurizer represents a reasonable amount of time to meet these conditions under an expedited circumstance.

The Low Temperature Overpressure Protection System must be tested on a periodic bases consistent with the need for its use. A CHANNEL FUNCTIONAL TEST shall be performed prior to enabling the overpressure protection system during cooldown and startup.

The limitations and surveillance requirements on the ECCS equipment provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or the limiting conditions placed on the pressurizer.

The restrictions for startup of a RCP limits the heat input accident to within the relieving capabilities of a single PORV.

(1) Pressure Mitigating Systems Transient Analysis Results July 1977 Westinghouse Owners Group on RCS Overpressurization.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.4

Shall be removed and examined to determine changes in their material properties, as required by 10 CFR 50 Appendix H.

4.3.4.D. Materials Irradiation Surveillance Specimen Inspection. (per unit)

Specimen capsules ~~to be used in the reactor vessel material surveillance program shall be withdrawn during the refueling period either immediately preceding or following the Effective Full Power Years (EFPY) of unit life as follows:~~

~~CAPSULE WITHDRAWAL SCHEDULE~~

~~UNIT 1~~

CAPSULE DESIGNATION	CAPSULE REMOVAL TIME (EFPY)
T	REMOVED (1.16)
U	REMOVED (3.52)
X	REMOVED (5.17)
Y	8.5
Z	32
W,S,V	STANDBY

UNIT 2

CAPSULE DESIGNATION	CAPSULE REMOVAL TIME (EFPY)
U	REMOVED (1.27)
T	REMOVED (3.56)
Y	8.5
X	13
W,S,V,Z	STANDBY

Amendment Nos. ~~106 & 96~~

Basis

4.3.4 The surveillance inspection program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i). The design of the plant, state of non-destructive testing technology, and access to areas to be inspected require such relief.

The Reactor Vessel Material Surveillance Program is designed to evaluate the effects of radiation on the fracture toughness of reactor vessel steel based on the transition temperature approach and the fracture mechanics approach.

10CFR50, Appendix H, paragraph II B.1 requires that the reactor vessel material surveillance program shall meet the requirements of ASTM E185-82 such that the surveillance capsules represent end-of-life fluences at the reactor vessel surface and 1/4T wall thicknesses. ~~Previous capsules were removed under Amendment Nos. 62 and 59~~.

Insert for new page 316 b

6.6.1.G PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. The reactor vessel pressure and temperature limits and the heatup and cooldown rates are addressed in Specification 3.3.2.A. The analytical methods used to determine the pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and approved by the NRC as described in WCAP-13406, "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 & 2", dated July 1992 and approved by the NRC SER dated ________. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determinedso that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

ENCLOSURE 4

PROPOSED CHANGES

TYPED PAGES FOR PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
DPR-39 AND DPR-48
FOR LICENSE AMENDMENT REQUEST NO. 92-03

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- 1.4 Actuation Logic Test
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- 1.7 Channel Calibration, Instrument
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1.28 OPERATING

OPERATING is defined as performing the intended function in the intended manner.

1.29 OPERATING CYCLE

The OPERATING CYCLE shall be the interval between the end of one major refueling outage and the end of the next subsequent major refueling outage per unit.

1.30 OPERATIONAL MODE - MODE

An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1, when fuel assemblies are present in the reactor vessel.

1.31 PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

1.31a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period in accordance with Specification 6.6.1.G. Plant operation within these operating limits is addressed in Specification 3.3.2.A.

1.32 PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in the Reactor Coolant System component body, pipe wall, or vessel wall.

1.33 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes will be accomplished in such a way as to assure compliance with 10 CFR parts 20, 61 and 71, and Federal and State regulations and other requirements governing the shipment and disposal of radioactive waste.

1.34 PROTECTION LOGIC CHANNEL

A PROTECTION LOGIC CHANNEL shall be an arrangement of relays, contacts or other components which operate in response to INSTRUMENT CHANNEL outputs to produce a decision output. The decision output is the initiation of a protective action signal. At the system level, the decision output is the operation of a sufficient numer of ACTUATION DEVICES and the associated ACTUATED EQUIPMENT as required to place or restore the Nuclear Steam Supply System to a design safe state. The channel is deemed to include the ACTUATION DEVICES.

1.35 PROTECTION SYSTEM

The PROTECTION SYSTEM shall consist of both the Reactor Protection System and the Engineered Safeguarus System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through ACTUATION DEVICES) which are required to operate in order to place or restore the Nuclear Steam Supply System to a design safe state.

LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURIZATION AND SYSTEM INTEGRITY

A. Heatup and Cooldown

The Reactor Coolant System (with the exception of the pressurizer) temperature and pressure and heatup and cooldown rates shall be maintained within the limits specified in the Pressure and Temperature Limits Report (PTLR) during heatup, cooldown and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION: With any of the limits specified in the PTLR exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in a least MODE 3 within the next 6 hours and reduce RCS T_{AVG} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURIZATION AND SYSTEM INTEGRITY

A. The reactor coolant temperature and pressure shall be determined to be within the limits specified in the PTLR at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

LIMITING CONDITION FOR OPERATION

3.3.2 (Continued)

B. The Reactor Coolant System pressure and temperature limits specified in the PTLR shall be recalculated periodically as required, based on results from the material surveillance program.

C. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the primary and secondary coolant is below 70°F.

D. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

E. Hydrostatic Testing

1. System inservice leak and hydrotests shall be performed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI, and applicable addenda; except as stated in Specification 4.3.4.C.1.

SURVEILLANCE REQUIREMENT

4.3.2

B. Not Applicable

C. Not Applicable

D. Not Applicable

E. Not Applicable

LIMITING CONDITION FOR OPERATION

3.3.2.G. Low Temperature Overpressure Protection (Continued)

b. The Reactor Coolant System (RCS) pressure shall be less than 100 psig, and the pressurizer level less than 25%, or

c. The RCS is depressurized and one PORV and it's isolation valve are open.

2. a. A maximum of one* charging pump, shall be capable of injection into the RCS.

b. No safety injection pumps shall be capable of injection into the RCS.

c. No accumulators shall be capable of injection into the RCS.

* For short durations of time during pump switchover, two charging pumps may be capable of injection into the RCS for the purpose of maintaining seal injection flow to the reactor coolant pumps.

SURVEILLANCE REQUIREMENT

4.3.2.G. Low Temperature Overpressure Protection (Continued)

4. Verifying each PORV's isolation valve is open at least once per shift when this method is being used for low temperature overpressure protection.

5. Testing pursuant to Specification 4.0.5.

b. The RCS pressure shall be verified to be less than 100 psig, and pressurizer level shall be verified to be less than 25% at least once per shift, when this method is being used for low temperature overpressure protection.

c. Verifying one PORV and it's isolation valve are open at least once per shift, when this method is being used for low temperature overpressure protection.

2. At least two of the three charging pumps, and all safety injection pumps and accumulators, shall be verified to be incapable of injecting into the RCS prior to entering a condition in which they are required to be incapable of injection into the RCS, and at least once per shift thereafter while they are required to be incapable of injection into the RCS.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.2.G. Low Temperature Overpressure Protection (Continued)

3. When starting a reactor coolant pump, when no reactor coolant pumps are running, the temperature in the secondary side of the steam generator in the loop in which the reactor coolant pump is to be started shall be less than 50°F higher than the RCS temperature.

APPLICABILITY: Mode 4 when the temperature of any RCS cold leg is less than or equal to 320°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION: a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days, or within the next 24 hours either;

- Depressurize the RCS to less than 100 psig and lower pressurizer level to less than 25%, or
- Depressurize the RCS and open at least one PORV and it's block valve.

4.3.2.G. Low Temperature Overpressure Protection (Continued)

3. Not applicable.

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DELETED

3.3.2 & 4.3.2 FRACTURE TOUGHNESS PROPERTIES

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and 10CFR50 Appendix G.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan. These properties are then evaluated in accordance with Appendix G of the 1986 Edition of Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-13406, "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 & 2", July 1992.

Heatup and cooldown limit curves are calculated using the most limiting value of the adjusted nil-ductility reference temperature (adjusted RT_{NDT}) at the end of 14 effective full power years (EFPY) of service life. The 14 EFPY service life period is chosen such that the limiting adjusted RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This limiting adjusted RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in WCAP-13406 "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 & 2". Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Adjusted reference temperatures, based upon the fluence, copper content, and nickel content of the material in question or based on credible surveillance data, can be calculated using Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves specified in the Pressure and Temperature Limits Report (PTLR) include adjustments for this shift in RT_{NDT} at the end of 14 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-13406.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the adjusted nil-ductility reference temperature (adjusted RT_{NDT}) is used, which includes the initial RT_{NDT}, the radiation-induced shift (ΔRT_{NDT}) corresponding to the end of the period for which heatup and cooldown curves are generated, and a margin factor.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR} for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{IT} = the stress intensity factor caused by the thermal gradients,

K_{IR} = reference stress intensity factor provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup and cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for adjusted RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The PTLR defines limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

The leak test limit curve shown on the heatup curves in the PTLR represent minimum temperature requirements at the leak test pressure specified by ASME Section III and the NRC Standard Review Plan NUREG-0800.

Allowable combinations of pressure and temperature in the PTLR for specified temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, 1976 Summer Addenda.
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13. CWE-86-563, "Low Temperature Overpressure Protection System Setpoint Analysis", August 25, 1986.
14. CWE-865-588, "Low Temperature Overpressure Protection System Setpoint Analysis Extension", October 27, 1986.

Bases: Low Temperature Overpressure Protection

3.3.2.G & 4.3.2.G There are 3 means of protecting the RCS from overpressurization by a pressure transient at low temperatures (below 320°F). The first type of protection is ensured by the operation and surveillance of the power operated relief valves with a lift setting of 435 psig. A single power operated relief valve (PORV) will relieve a pressure transient caused by 1) a mass addition into a solid RCS from a charging pump or 2) a heat input based on a reactor coolant pump being started in an idle RCS and circulating water into a steam generator whose temperature is 50°F greater than the RCS temperature. (1)

The second means of protection is ensured by a PORV being open. It will have the same relieving capabilities as mentioned above.

The third means of protection limits the pressurizer level to 25% and the pressurizer pressure to 100 psig. A pressure transient caused by the inadvertent mass addition from a charging pump running for 10 minutes will be relieved by the large gas volume and low pressure present in the pressurizer as mentioned above. Maintaining the pressurizer level below 25% will also make the high pressurizer level deviation alarm available to the operator during a mass addition accident.

In the event that a single PORV becomes inoperable, the repair period of 7 days is based on allowing sufficient time to effect repairs using safe and proper procedures and upon the operability of the redundant PORV. The 24 hour time period to reach the restrictive conditions in the pressurizer provides sufficient time to meet these conditions.

In the event that both PORV's become inoperable, the condition is more serious than for a single inoperable PORV, therefore every attempt should be made to depressurize the RCS in a controlled manner as rapidly as possible. The 16 hour time period to reach the restrictive conditions in the pressurizer represents a reasonable amount of time to meet these conditions under an expedited circumstance.

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The limitations and surveillance requirements on the ECCS equipment provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or the limiting conditions placed on the pressurizer.

The restrictions for startup of a RCP limits the heat input accident to within the relieving capabilities of a single PORV.

(1) Pressure Mitigating Systems Transient Analysis Results July 1977 Westinghouse Owners Group on RCS Overpressurization.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.4	4.3.4.D. Materials Irradiation Surveillance Specimen Inspection. (<u>per unit</u>) Specimen capsules shall be removed and examined to determine changes in their material properties, as required by 10CFR50 Appendix H.

Basis

4.3.4 The surveillance inspection program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i). The design of the plant, state of non-destructive testing technology, and access to areas to be inspected require such relief.

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10CFR50, Appendix H, paragraph II B.1 requires that the reactor vessel material surveillance program shall meet the requirements of ASTM E185-82 such that the surveillance capsules represent end-of-life fluences at the reactor vessel surface and 1/4T wall thicknesses.

6.6.1.G PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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ENCLOSURE 5

SUPPORTING DOCUMENTATION

(1) WCAP-13406, "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 & 2"

(2) WCAP-10962, "Zion Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations", Revision 2

(3) LTOP Enable Temperature Methodology