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WASHINGTON,S.L.	Washington Public Power Supply System
POWERS,C.M.	Washington Public Power Supply System
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 88-024-01:on 880703,single primary containment monitor intermittently exceeded 150 F temp & time limit.
W/8 ltr.

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	H ST LOBBY WARD	1	1	LPDR	1	1
	NRC PDR	1	1	NSIC HARRIS,J	1	1
	NSIC MAYS,G	1	1			

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Special Report - Reactor Containment Temperature Greater Than 150°F For More Than Eight Hours

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 7	0 3	8 8	8 8	— 0 2 4 —	0 1	1 2	1 9	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 5

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Special Report
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
S. L. Washington, Compliance Engineer	5 0 9	3 7 7 - 2 0 8 0

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES *(If yes, complete EXPECTED SUBMISSION DATE)*	X NO	EXPECTED SUBMISSION DATE (15) MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

Technical Specification Section 3/4.7.8 requires preparation and submittal of a Special Report anytime an area temperature monitor listed in Table 3.7.8-1 exceeds the specified limit for more than 8 hours.

For the period from July 1, 1988 to July 20, 1988, a single Primary Containment Monitor, CMS-TE-39, intermittently exceeded the 150°F temperature and time limit. The temperature exceeded the 150°F limit for approximately 275 hours, with the maximum temperature experienced being 154°F.

The abnormally high temperatures experienced were due to a 30 to 40 percent increase in the Primary Containment heat load after the April-June, 1988 Refueling and Maintenance Outage. The cause of the heat load increase is not known, but, leaking Main Steamline Relief Valves (MSRV) and leaking MSRV Tailpipe Vacuum Breaker valves are thought to be the major contributors. The electrical, mechanical, and structural equipment in the vacinity of CMS-TE-39 has been reviewed and there are no immediate problems or conditions. All equipment will remain qualified for both normal (including operation above 150°F) and accident conditions until the 1989 Spring outage.

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Plant Conditions

a) Power Level - 95% - 98.8%
b) Plant Mode - 1

Event Description

For the time period between July 1, 1988 through July 20, 1988, a single Primary Containment Monitor, CMS-TE-39, intermittently exceeded the 150°F limit for more than 8 hours as specified in Technical Specification Section 3/4.7.8, Table 3.7.8-1. The temperature record for temperature monitor CMS-TE-39 for the period of July 1, 1988 through July 20, 1988 is included in Table 1. Temperature measurements were initially being recorded once each shift. In support of this report, tabulation of measurements every two hours was initiated. The temperature exceeded the 150°F limit for approximately 275 hours, with the maximum temperature experienced being 154°F.

The abnormally high temperatures experienced were due to an increased heat load in the Primary Containment after the third (R3) Refueling and Maintenance Outage April through June, 1988. This increased heat load was determined from the increased delta (change) temperature between the Reactor Closed Cooling water primary containment inlet and outlet water temperatures. Prior to the R3 Outage the delta T was 6-7°F and after the delta T is 10-11°F which represents a 30%-40% increase in primary containment heat load and a 15% increase in RCC System heat load. The cause of the heat load increase is not known but the major contributor to the increased heat load is thought to be leaking Main Steamline Safety Relief Valves (MSRVs) and leaking MSRV tailpipe vacuum breaker valves.

Immediate Corrective Action

In an effort to increase cooling to the drywell, Reactor Closed Cooling Water System (RCC) valve lineup and flow paths were verified, non-critical RCC loads were shed and a third RCC pump was started. There was no RCC system performance increase as a result of running the third RCC pump.

On July 15, 1988 in an effort to determine if the shell side of the RCC heat exchangers was fouled, one of the three heat exchangers, RCC-HX-1A, was taken out of service and drained. Based on visual observations with a video probe, no significant shellside fouling was found. However, while in the two heat exchanger lineup RCC heat exchanger outlet temperatures began to decrease due to improved thermal performance. Plant data showed that in the two heat exchanger lineup the calculated coefficient of heat transfer more than doubled from approximately 125 BTU/hr-ft^2°F to greater than 300 BTU/hr-ft^2°F. The discovery of the more efficient two heat exchanger lineup and the subsequent reduced RCC water temperatures more than any other Plant action contributed to decrease in incidents when CMS-TE-39 exceeded 150°F. In a further effort to decrease drywell heat loads, seven MSRVs were cycled in a effort reseat the valves to reduce MSRV leakage. Steam flow through an MSRV is piped through the drywell to the suppression pool in uninsulated pipe. No appreciable reduction in MSRV leakage resulted from this effort.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Further Evaluation

Equipment

In the general vicinity of the area temperature monitor, safety-related equipment includes: Electrical penetrations, acoustical monitor sensors and charge converters, main steam relief valves and their associated air operators, air valves and solenoid valves.

On July 20, 1988, while lifting MSRV's to reseat them the acoustical monitors associated with relief valves 1B and 1C failed to operate properly. Corrective action compensated for the degraded signal from 1C and it was returned to operable status. Alternative means exist for relief valve position or leakage indication. Compliance with the associated Technical Specification is being documented independently. (See Supply System Letter G02-88-161 dated July 25, 1988 and Technical Specification Amendment 61 dated July 27, 1988.)

The failure of Acoustic Monitor Channel 1C was due to the reduced capacity of circuit charge amplifier which is located in an environmentally (moisture) protected shield box in the areas where the high temperatures were observed. Acoustic Monitor Channel 1B was failed due to a broken conductor at the accelerometer connector. This failure, of 1B, was not temperature related. During troubleshooting of Acoustic Monitor Channel 4D a bad softline cable and a reduced capacity charge amplifier were found. At the present time there is insufficient data to correlate the 1C and 4D charge amplifier failures or the 4D softline cable failure to the higher than normal temperatures experienced during this event period as a charge amplifier failure may also be attributed to a broken or intermittent signal path conductor.

Accident Conditions

Following a loss of coolant accident condition, the drywell region of the containment is postulated to experience a temperature of 340°F for 3 hours and 320°F for 3 hours, with decreasing but elevated temperatures thereafter. The above equipment is qualified to function under such conditions.

Environmental Qualification of Electrical Equipment

All safety-related electrical equipment in the vicinity of temperature element CMS-TE-39 has been reviewed. With respect to thermal aging, the most sensitive components are the Solenoid Pilot Valves (SPVs) on the Main Steam Relief Valves. All other safety-related equipment in the vicinity of the area temperature monitor was determined to be less sensitive to ambient temperature.

The solenoid valves on the Main Steam Relief Valves (MSRV) are Crosby Valve Co. [model IMF-2] which were installed in the plant prior to obtaining the initial Operating License. The solenoid valves have a qualified life of 6.4 years at 150°F. Thermal aging was the limiting factor in establishing the solenoid valves' qualified life. In establishing qualified life of equipment in containment, the design base abnormal temperature of 150°F is used; however, actual operating temperature measurements on the MSRV Solenoid Valves have not been made.

A calculation has been performed which evaluated the effects on the SPVs of termperatures higher than the maximum in-containment (design base) abnormal temperature [150°F]. The calculation determined that the 275 hours, during which the solenoid valves experienced termperatures up to 154°F, would reduce the calculated thermal qualified life by 52 hours. This 52 hours is based upon a conservative assumption of 275 hours at a constant 154°F (i.e., 150°F + 4°F). The 52 hours is not significant when compared to the initial 6.4 years of qualified life (see Figure 1).

Figure 1 illustrates the dependence of qualified life on equipment temperature and percent of time the plant is critical (for this system, time critical is indicative of time at temperatures). Percent of time critical, to date, has been approximately 67%. As can be seen from the graph, a qualified life of approximately 5.3 years can be demonstrated for 163°F with a plant time critical of 70%. This would still place the end of qualified life at R-4, and our original replacement schedule for change out at R-4 is still valid.

As part of the investigation of high area temperature, relief valve tail pipe temperatures were investigated. Tail pipe temperatures of up to 240°F were measured using temperature elements installed to monitor for relief valve leakage. The acoustical monitor transducers and associated cable assemblies were evaluated for qualified life at 150°F, similar to other equipment in containment. The transducers are mounted directly on the tailpipes, the element sensitive to temperature is the cable assembly.

The acoustical monitor sensor and mounting acts as large thermal resistance to limit the temperature exposure from the tailpipe to the degradable material of the sensor. To demonstrate this, an actual acoustic monitor assembly was instrumented with thermocouples and tested by the Supply System. These tests were performed to establish the relationship between the temperature of the pipe (simulated to be 240°F) on which the acoustic monitor is mounted and the temperature at the location of degradable materials within the acoustic monitor assembly. Critical materials are the cable insulation and the sealant for the connector. These tests demonstrated a significant temperature gradient between the tailpipe temperature and the location of these materials. The temperature gradient was on the order of 75°F, dependent upon the surrounding air temperature (simulated to be approximately 150°F during the test) and the tailpipe temperature.

These temperature test results have been factored into new calculations, calculating the qualified life of the equipment. The new calculation of qualified life also took into account actual tailpipe temperature data for the past year, from November 1987 to November 1988. Considering this year as a representative, conservative year of operation exposure, includes the effect on qualified life of tailpipe temperatures up to 240°F. These calculations conclude that the age sensitive materials are qualified for the following periods:

Material	Qualified Life (years)
Brand Rex Coaxial Cable	37
Dow-Corning Sealant	4.8
RayChem Shrink Sleeving	>40

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
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Two connector configurations exist in the Plant. Original equipment used an Amphenol connector, this connector is sealed with the Dow-Corning sealant. The sealant was installed during November 1986. Based on the calculations described above, these devices require repair or replacement in 1991. The Amphenol connector is no longer available. Installation of the replacement connector uses a Raychem shrink sleeving assembly to seal the connector. The replacement connector itself has no ageable parts. The Raychem shrink sleeving will require replacement when the cable has aged to the end of its qualified life.

The results of these tests and calculations will be included in Qualification documentation and in required maintenance schedules.

Additionally, 4 cable assemblies were selected for inspection and removed from containment during the recent November 30 through December 8, 1988 unscheduled outage. The selected assemblies were associated with the highest tailpipe temperatures. The assemblies were tested for critical electrical characteristics and inspected. The assemblies exhibited characteristics virtually unchanged from those exhibited by new cable. Visual inspection indicated that the outer cable jacket was just starting to exhibit some of the initial symptoms of aging (discolorization).

In summary, the operable electrical equipment in the region of the temperature monitor which exceeded the temperature limit has the capability to remain operable until at least the next scheduled refueling outage, even if similar temperature excursions are experienced in the future.

Environmental Qualification of Mechanical Equipment

The materials in safety-related mechanical equipment in the vicinity of the area monitor which exceeded the 150°F limit would not be adversely effected by the temperature exceeding 150°F for a significant period of time and remained operable throughout the period. Qualified life would also not be effected significantly.

Structural Materials

The temperature effects on structural materials in general have been considered for concrete and steel inside containment. Potentially the two primary effects would be loss of strength and increased stresses due to differential movement of structural members. Secondary effects would include degradation of the shielding properties of concrete. For temperatures in the range of 150°F the effect on concrete and steel (including mild steel and high strength bolting) is minimal, less than 10% reduction in strength. Therefore, a change in temperature of 10 to 15°F in this temperature range would have a negligible effect on strength. In the same manner, increased stresses due to possible increase in differential movement would be negligible compared to the magnitude of the allowable stresses. Effects on shielding properties of concrete would be negligible.

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Further Corrective Action

1. During a Plant Shutdown, August 24 to September 5, 1988 two actions were taken to reduce drywell heat loads. First, three MSRV valves were reworked to reduce their leakage. The tailpipe temperatures of two of these valves were reduced by an average of 72°F. The third reworked valve was cycled following the rework to test the associated acoustic monitor and apparently did not properly reseat and no substantial improvement in the tailpipe temperature has been observed. Also, another problem thought to be contributing to the drywell heat load was leaking MSRV vacuum breaker valves. These valves are sealed by the use of O-rings which are often damaged when an MSRV is lifted. All vacuum breaker valve O-rings were replaced by new 5% smaller O-rings which must be stretched slightly to install. It is hoped that this will reduce O-ring damage and reduce steam leaking into the drywell.

2. During the "R4" Refueling and Maintenance Outage schedule for April/May 1989 several (current estimate is seven) MSRV valves will be either replaced or reworked to try to reduce the number of leaking SRVs.

3. At least two techniques are available for improving the basis for establishing equipment service temperatures, surveys when the containment can be entered and the reactor system is at temperature and pressure and monitoring while the plant is at power. Such techniques will be applied in the future to improve the basis for service temperature of equipment in containment.

4. Repeated periods when the area temperature limit for containment will be exceeded may occur in the coming period of high makeup water temperature and high ambient air temperatures. Subsequent Special Reports will be issued at 30 day intervals, if required, in compliance with the Technical Specification requirement. Since this report there have been no other high temperature periods that exceeded eight hours.

5. It is necessary to complete documentation modifications to the qualification file for acoustical monitors and schedule acoustical monitor cable assemblies for replacement at the required interval based on the revised evaluation.

6. During the 1989 refueling outage, further inspections will be performed to identify other potential sources of increased heat loads. Following the outage, a Drywell temperature monitoring program will be set up to determine the effectiveness of the repairs made during the outage.

Safety Significance

The equipment in the vicinity of CMS-TE-39 has the capability to remain operable until at least the next scheduled refueling outage, even if similar temperature excursions are experienced in the future. Additionally, the equipment can experience future temperature excursions above 150°F, but not exceeding 160°F, for extended periods and remain within the bounds of this technical evaluation.

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Simliar Events

LERs 84-034 and 85-018 documentated events in which area temperature Technical Specification limits were exceeded.

EIIS Information

Text Reference	EIIS Reference System	Component
Primary Containment Monitor (CMS-TE-39)	IK	TE
Main Steamline Relief Valve (MSRV)	MS	RV
Main Steamline Relief Valve Tailpipe Vacuum Breakers	MS	PDCV
Reactor Closed Cooling System	WBA	- - - - -
Main Steam Relief Valve Solenoid Pilot Valves (SPV)	MS	PSV
Primary Containment	NH	- - - - -
RCC Pump	WBA	P
RCC Heat Exchanger (RCC-HX-1A)	WBA	HA
Acoustic Monitor Sensor	MS	VE
Acoustic Monitor Charge Convertor (Amplifier)	MS	AMP
Main Steamline Relief Valve Tailpipes	MS	PSP

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

TABLE 1

CMS-TE-39 AREA TEMPERATURE MONITOR

TEMPERATURE, DEGREES FAHRENHEIT

TIME/DATE	7/01	7/02	7/03	7/04	7/05	7/06	7/07	7/08	7/09	7/10
0000		152	148	148	150	150	150	151	152	153
0200						150	150	151	151	153
0400						149	150	151	151	153
0600							150	151	151	152
0800		151	148	147	149	150	151	151	151	152
1000							151	151	151	152
1200							151	152	151	152
1400							151	153	152	152
1600	150	152	148	148	150	149	151	152	152	152
1800							151	152	152	153
2000					150	150	151	152	151	153
2200					150		151	152	152	153

TIME/DATE	7/11	7/12	7/13	7/14	7/15	7/16	7/17	7/18	7/19	7/20
0000	153	153	152	152	150	149	149	149	148	151
0200	153	153	152	152	150	149	149	149		150
0400	153	152	152	152	150	149	149	149		150
0600	153	152	151	152	150	148	149	148		150
0800	153	152	151	151	150	148	148	148	148	147
1000	153	153	151	151	149	147	147			146
1200	153	153	151	150	148	148	147			145
1400	153	153	151	149	148	148	147			146
1600	154	153	151	149	148	148	148	148	150	147
1800	154	152	151	149	149	-	148			
2000	153	152	151	150	149	148	149			
2200	153	153	151	150	149	148	149			

NOTE: Blanks indicate temperature readings not taken.

*U.S. GPO: 1988-520-589/00070

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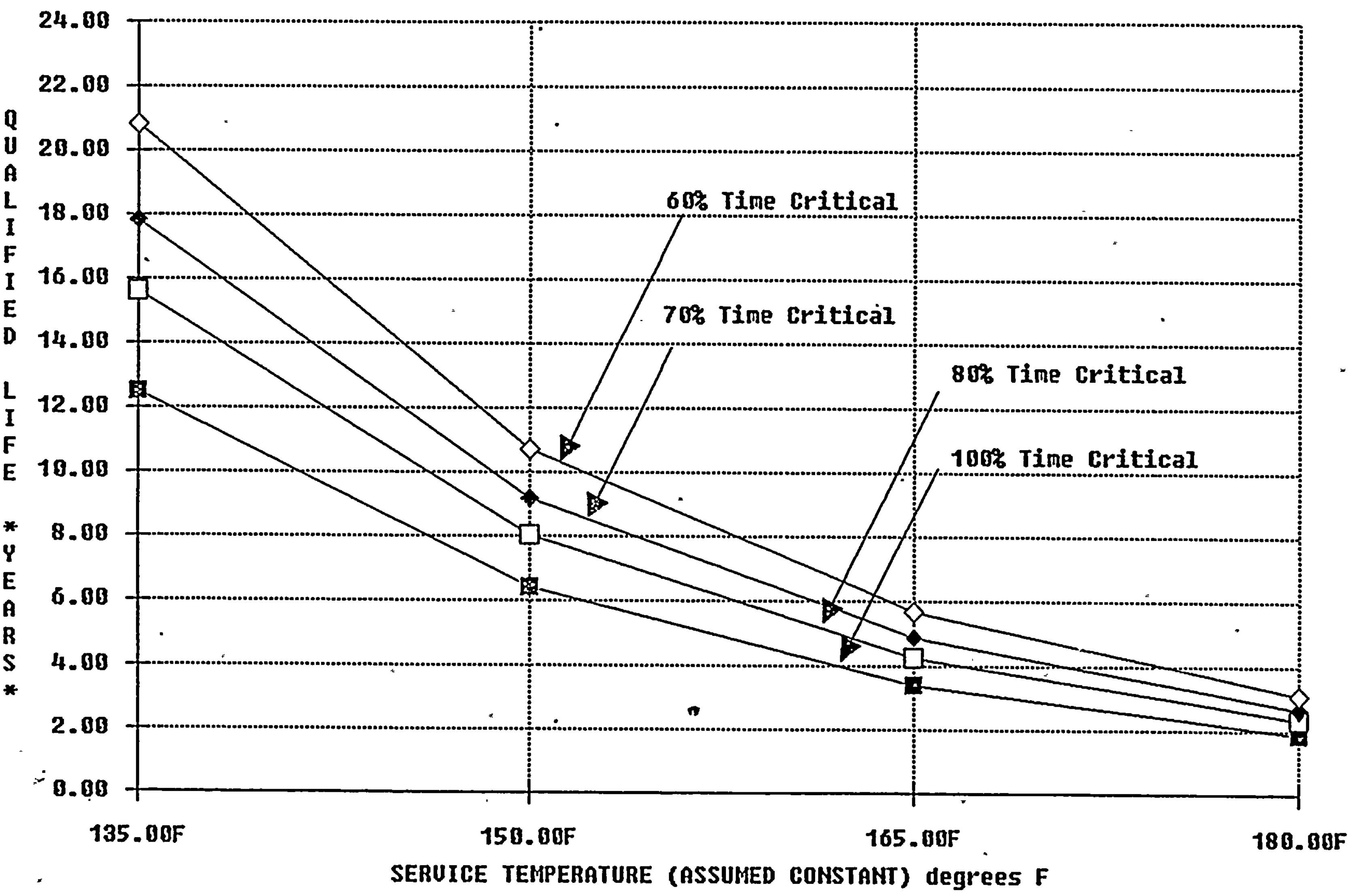
FIGURE 1

QUALIFIED LIFE, ADJUSTED BY DIFFERENT % OF PLANT TIME CRITICAL FACTORS, AND PLOTTED AGAINST VARIOUS TEMPERATURES.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

Docket No. 50-397

December 19, 1988

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 88-024-01

Dear Sir:

Transmitted herewith is Licensee Event Report No. 88-024-01 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Very truly yours,

JW Baker

C.M. Powers (M/D 927M)
WNP-2 Plant Manager

CMP:lg

Enclosure:
Licensee Event Report No. 88-024-01

cc: Mr. John B. Martin, NRC - Region V
Mr. C.J. Bosted, NRC Site (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D.L. Williams, BPA (M/D 399)

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