Nebraska Public Power District
Nebraska's Energy Leader

NLS2001051
May 16, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

Subject: Additional Information Related to Potential Yellow Finding (EA 00-248)
NRC Inspection Report No. 50-298/00-07
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

References:

1. Letter to J. H. Swailes (NPPD) from K. E. Brockman (NRC) dated December 18, 2000, "Cooper Nuclear Station Special Inspection - NRC Inspection Report 50-298/00-07; Preliminary Yellow Finding."

2. Letter to NRC from J. H. Swailes (NPPD) dated March 22, 2001, "Information Related to Preliminary Finding, NRC Inspection Report 50-298/00-07, Cooper Nuclear Station, NRC Docket 50-298, DPR-46."

3. Letter to J. H. Swailes (NPPD) from C. S. Marschall (NRC) dated April 16, 2001, "Cooper Nuclear Station - Request for Additional Information Regarding a Potential Yellow Finding (EA 00-248)."

Nebraska Public Power District (NPPD) submits the attached additional information addressing a Nuclear Regulatory Commission (NRC) preliminary yellow significance determination concerning findings identified in NRC Inspection Report 50-298/00-07 (Reference 1). The NRC inspection evaluated the Cooper Nuclear Station (CNS) environmental qualification (EQ) program. In Reference 2, NPPD submitted updates of reports that had been provided earlier to the NRC Inspection Team, which are pertinent to the NRC final significance determination. The contents of these reports and NPPD's perspectives regarding the preliminary finding were discussed at a public regulatory conference conducted on March 29, 2001, at NRC Regional Headquarters in Arlington, Texas.

At the conference, NPPD committed to provide responses to several requests for additional information. The NRC formalized these requests in Reference 3. The subject of each request is identified in Attachment 1. The response to each request is provided herein as indicated in

IEP1

Attachment 1. NPPD appreciates the NRC's thorough and timely consideration of the information provided herein and the information previously provided in Reference 2 and the regulatory conference.

NPPD also reiterates that tests completed shortly before the regulatory conference (the results of which were discussed at the conference and are provided herein) provide clear evidence that the NPPD circuit evaluation associated with safety-relief valve (SRV) operation was highly conservative. Application of the results of that testing to the existing analysis demonstrates that significant margin exists with respect to assuring SRV operability. This new information provides important evidence related to the ultimate significance determination.

Further, in light of discussions at the regulatory conference, NPPD has performed a number of sensitivity analyses for the risk assessment. Those analyses, also discussed herein, demonstrate that there is a significant margin in the risk assessment in regard to the assumptions made concerning equipment performance and the ultimate contribution to changes in core damage frequency.

Based on the results of the testing identified above, the previous risk assessment, and the sensitivity analyses for the risk assessment, NPPD continues to conclude the change in risk associated with this issue does not exceed the green-to-white threshold value.

NPPD considers this letter to contain no commitments as identified in Attachment 2.

Enclosure C contains a report that is considered proprietary to Spectrum Technologies USA Inc. As such, NPPD hereby requests that this information be withheld from public disclosure in accordance with 10 CFR 9.17 and 10 CFR 2.790. An affidavit attesting to the proprietary status is provided in Attachment 3.

Should you have any questions concerning the information discussed in this letter, please contact Michael Boyce of the CNS Nuclear Licensing and Safety Department at (402) 825-5100.

Sincerely,

for

John H. Swailes
Vice President of Nuclear Energy

/erg
Attachments/Enclosures

cc: Regional Administrator
U.S. NRC - Region IV

Senior Project Manager
U.S. NRC - NRR Project Directorate IV-1

Senior Resident Inspector
U.S. NRC

NPG Distribution

Records

ATTACHMENT 1

The following table identifies the information requested by the NRC at the regulatory conference in Arlington, TX on March 29, 2001, regarding NRC Inspection Report 50-298/00-07.

These items are listed below (corresponding to the items in the April 16, 2001, request for additional information letter) for ease of reference and each is addressed following the listing in this attachment. Supporting documentation is provided in the indicated enclosures.

Item	Requested Information or Document	Enclosure
1	Environmental qualification of the SRV pilot solenoid valves.	Enclosure A
2	Risk sensitivity to the reliability of the SRVs.	Enclosure B
3	Copies of new test reports: Constellation Nuclear Summary Report Spectrum Technologies Test Report (re: fuses) Wyle Laboratories Test Report (re: lights & pressure switches)	Enclosure C
4	EQ of motor control centers (MCCs) @ 903' el.	Enclosure D
5	Ground detection circuit drawings and supporting data.	Enclosure E
6	Definitions and scope of "medium" harsh environment.	None
7	Components in a "medium" harsh environment.	Enclosure B
8	Sensitivity of a "medium" harsh environment.	Enclosure B
9	EQ of the Weidmuller terminal blocks.	Enclosure F
10	Distribution system level fuse/breaker coordination.	None
11	Higher level fuse/breaker coordination.	None

1. Provide copies of environmental qualification (EQ) records and documentation for the safety-relief valve pilot solenoid valves. Specifically, the profiles to which the solenoids were qualified.

   Enclosure A provides a copy of Environmental Qualification Data Package (EQDP)-253 which documents the basis for environmental qualification of the SRV solenoid valves to the peak drywell temperature of 340F. The EQDP concludes that the solenoid valves were tested to 335F and are qualified with margin to 340F. This qualification is based on testing performed by Target Rock Corporation (TRC) (Ref. Test Report 5074, Environmental Qualification Design Input (EQDI) 2580). This testing demonstrates SRV solenoid valve performance during a loss of coolant accident (LOCA) simulation with a peak temperature condition of 355F, which satisfies the Cooper Nuclear Station (CNS) temperature requirement of 340F, and the IEEE 323-74 recommended margin of 15F.

   Additionally, the following specific concerns are addressed in regard to the EQDP for the SRV solenoid valves (EQDP-253).

   1a. Confirm qualification of the conformal coating material which appears to be material different than identified in the test report.

   The similarity discussion in Section F of EQDP-253 describes that the terminal board and connections for the test specimens were coated with Chemtronics Konform silicone resin. The terminal boards and connections for the installed solenoid valves are coated with Patel Engineers Conformal Coating (PECC). The PECC is a silicone resin conformal coating (MIL-I-46058C, Type SR) that has been type tested in test report PEI-TR-850400-2 (EQDI 2536). The EQDP states that the test program described in PEI-TR-850400-2 meets or exceeds CNS environmental requirements with the exception of pressure. Section F of the EQDP provides a technical justification for this deviation.

   The following information is provided to substantiate that the PECC is a like-for-like replacement for the Chemtronics Konform Type C416 silicone conformal coating.

   TRC Test Report 5074 (EQDI 2580) documents that Chemtronics Konform silicone resin conformal coating was applied to the terminal board and connections of the SRV solenoid valve. The test specimens were coated with Konform Type C416 silicone conformal coating. TRC has performed an item equivalency evaluation (TERI-033, EQDI 3170) which establishes that the use of Konform SR 2000 in place of the Konform Type C416 silicone resin does not affect the environmental qualification of the SRV solenoids. In TERI-033, TRC states that the change in solvents, (e.g., volatile organic compounds) does not affect the protection provided by the silicone resin and concludes that the use of this replacement item does not affect environmental qualification.

   PECC and Chemtronics Konform SR 2000 are both silicone resin conformal coatings with a useful temperature range up to 199C/390F. Both products have the same material specific CAS registration number (68952-93-2) for the silicone resin. The primary difference being the weight percentage of the silicone resin (10-15% for Konform SR 2000 and 73% for PECC) due to the fact that Konform SR 2000 contains more volatile organic

compounds or solvents in the uncured state. Since PECC uses the same silicone resin as was originally tested, combined with the rationale in TERI-033, it is concluded that PECC is a like-for-like replacement for Konform Type C416 and is similar to the conformal coating used in TRC Test Report 5074.

1b. Identify the margin provided with regard to the temperature profile used in the operability evaluation (OE) (i.e., 340F vs. the test report 355F).

The testing to qualify the SRV solenoid valves was designed by TRC to qualify the valves to generic boiling water reactor (BWR) conditions including a peak temperature of 340F. The testing utilized the recommended temperature margin specified in IEEE 323-74 and Regulatory Guide (RG) 1.89, but no additional margin exists at the peak temperature. However, when compared to the CNS specific profile there is significant duration margin at the peak conditions. There is also significant temperature margin throughout the entire profile.

The design basis accident (DBA) composite profile in EQDP-46 reflects a maximum dwell of approximately 33 minutes at 340F. The tested temperature profile contains additional margin in that it exposed the test specimens to 355F for approximately three hours. This is significantly longer than the required profile for the small steam line break (SSLB). The time equipment is required or demonstrated to remain functional is the most significant in terms of additional confidence in qualification that is achieved by adding margin to service conditions. EQDP-253 establishes qualification of the SRV solenoid valves for the maximum 6-month post accident operating time period.

1c. Confirm qualification included the IEEE-323 required testing to a double peak (include resolution of problem identification report (PIR) on this topic).

Enclosure A provides the test report profile information which shows a double peak test was conducted. The test report used to establish qualification of this equipment (EQDI 2580) did expose the test specimens to two temperature transients which both had a 15F margin over the required peak temperature of 340F. Therefore, the testing in EQDI 2580 satisfies the recommended temperature margin in IEEE 323-74 and RG 1.89.

PIR 4-12918 was written to address the concern that the margin criteria in CNS Engineering Procedure (EP) 3.12.2 was based on the guidance in IEEE 323-83, which has not been formally endorsed by the NRC. IEEE 323-83 permits two methods of satisfying LOCA temperature margin (a double peak with no margin at the peak and a single peak with peak temperature margin). The NRC has endorsed/modified the margin criteria in IEEE 323-74 in NRC Regulatory Guide 1.89, which is satisfied by EQDP-253. Thus, the PIR issue is not pertinent to the SRV solenoid valves.

1d. Identify the temperature qualification margin prior to the circa 1995 change out of the SRVs.

Prior to the change out of the SRV solenoid valves under design change package DC 92-043, the SRV solenoid valves were qualified per 10 CFR 50.49 by applying the

NRC Division of Operating Reactors (DOR) Guidelines for original plant equipment [Ref. EQDP 51]. The DOR Guidelines do not require an additional temperature margin. Upon replacement of the SRV solenoids, their qualification was upgraded in accordance with 10 CFR 50.49(j). The upgrade included qualification employing temperature margin in accordance with IEEE 323-74.

Prior to their replacement and upgrade the EQ of the SRV solenoid valves was based on Test Report 2199 (EQDI 403). The minimum peak temperature for the first and second transients was 342F and 343F, respectively. These peak temperatures bound the newly defined SSLB temperature conditions of 340F. Consequently, these solenoid valves would have been qualified, per the DOR Guidelines, to the newly defined SSLB temperature conditions.

1e. Identify the pressure qualification margin prior to the circa 1995 change out of the SRVs.

Prior to the change out of the SRV solenoid valves under design change package DC 92-043, the SRV solenoid valves were qualified per 10 CFR 50.49 by applying the DOR Guidelines for original plant equipment (Ref. EQDP-51). The DOR Guidelines do not require an additional pressure margin. Upon replacement of the SRV solenoids, their qualification was upgraded in accordance with 10 CFR 50.49(j). The upgrade included qualification employing additional pressure margin in accordance with IEEE 323-74.

EQDP-51 reflected that the required drywell pressure condition was 46 psig. This is lower than the currently defined pressure condition of 58 psig for drywell as specified in EQDP 46. The testing performed in EQDI 403 simulated accident pressure conditions up to 65 psig. Therefore, the change in the pressure service conditions did not impact the prior qualification basis for the SRV solenoid valves evaluated in EQDP-51.

1f. Identify the vibration qualification of the SRVs.

The qualification program used to establish qualification of the SRV solenoid valves included the effects of vibration. The test sequence in Target Rock Test Report 5074 (EQDI 2580) included the following vibration-related simulations prior to the simulated DBA test:

1. Mechanical Cycling
2. Vibration Aging
3. SRV Induced Load Aging
4. Seismic Test (5 operating basis earthquakes and 1 safe shutdown earthquake)

2. Provide the results of a probabilistic risk assessment sensitivity study (risk achievement worth) for the safety-relief valves in the overall plant configuration presented for the EQ issue.

Enclosure B provides a copy of PSA-ES056, "Sensitivity Studies for PSA-ES051." This study indicates that the failure rate of the SRVs would have to increase significantly (a factor of 165) before the results of PSA-ES051 would be affected, i.e., would cross the GREEN/WHITE

threshold for core damage frequency. NPPD concludes from this study that PSA-ES051 is not particularly sensitive to the failure rate of the SRVs.

3. Provide copies of all new test reports discussed during the conference and the associated data. Include as a minimum, the safety-relief valve tailpiece pressure switches, the fuses, and the airlock door indicating light.

   As discussed at the regulatory conference, the SRV performance requirements are that one SRV remains functional for eight hours. Enclosure C provides the Constellation Nuclear Services "Summary Report of Fuse, Pressure Switch, and Light Assembly Testing Performed for Cooper Nuclear Station, NPPD by Spectrum Technologies and Wyle Laboratories." This summary report (EQDI 3173) provides independent evaluation of the Spectrum Technologies and Wyle Laboratories testing which were performed to validate assumptions and conclusions in the SRV circuit analysis that provides reasonable assurance that the eight SRVs remained functional for at least eight hours.

   Enclosure C also provides the Spectrum Technologies test report #TR01F0410, "Bussmann MIN-10 Fuses." This report (EQDI 3174) describes the results of the testing to demonstrate the fuses have acceptable current carrying capacity and clearing time. The "new" fuses met the test requirements, and in doing so confirmed the design of the test set-up. The set of fuses removed from the plant, the "old" set, met the test requirements for current carrying capacity sufficiently to validate the presence of margin above the 10 amp rated current carrying capability of the Bussmann MIN-10 fuses used in the SRV circuit analysis.

   Finally, Enclosure C provides the Wyle Labs test report 45670-01, "Nuclear Environmental Test Program on Two Pressure Switches and One Indicator Light Station." This report (EQDI 3167) also describes the results of the testing of mockups of portions of the SRV circuit and the drywell access hatch indication light circuit to validate the NPPD operability analysis assumptions for the circuits. This testing established that: (1) the insulation resistance to ground values used in the NPPD operability analysis for the SRV tailpipe pressure switches (i.e., zero ohms or a bolted ground fault) were excessively conservative and (2) that the assumed 100 ohm insulation resistance to ground value for the drywell access hatch indication light was reasonable. The SRV circuit analysis assumed a bolted fault to ground for the SRV tailpipe pressure switches. The testing, which duplicated the plant configuration of the SRV pressure switches and associated junction box and terminal blocks, demonstrated that leakage current through ground during LOCA conditions for the previously installed pressure switches would not be sufficient to blow the SRV circuit 10 amp fuses during postulated DBA conditions for the required mission time of at least eight hours. Based on these test results, the maximum total fault current would be reduced from the bounding circuit analysis value of more than six amps to approximately 0.008 amps. Since the other inside containment components of the SRV circuits are fully qualified for the LOCA conditions, SRV circuit ground fault leakage currents must flow through these pressure switches. Given these pressure switch results, the insulation to ground resistance assumptions of other inside containment DC components, such as the drywell airlock indicating light, are not significant since they cannot result in SRV circuit leakage current values in excess of approximately 0.008 amps.

Regarding the drywell airlock indicating light, the SRV circuit analysis assumed an insulation resistance to ground value of 100 ohms. After approximately 2.5 hours of accident simulation, the 0.1 amps circuit fuse for the drywell airlock light opened. The initial insulation resistance reading at the time the fuse opened was 257 ohms and the last reading taken after 24.5 hours of test was 695 ohms. Therefore, the testing confirmed that this assumption was reasonable.

4. Provide EQ records and documentation for the qualification of motor control centers on the reactor building 903-foot elevation, specifically, the profiles to which these centers were qualified.

   Enclosure D provides a copy of EQDP-230, Rev. 11 showing qualification of the motor control centers (MCCs).

   4a. Also identified was a specific concern related to the qualification of splices associated with the MCC; the affected component; and how NPPD considered the availability/qualification for use in the risk analysis. Were there any Scotch 33 or nonconforming Okonite splices, and if so, how did we address them?

   The only motor control centers on 903-foot elevation (903') that contained treatments that required repair were MCC-Q, the 250 VDC Division I Starter Rack, and the 250 VDC Division II Starter Rack.

   MCC-Q is located across the walkway from the Injection Valve Room on 903' NW. There were eleven nonconformances identified, which in turn, affected three components. Four of the nonconformances were on indication cables for RHR-MO-MO15A, and consisted of Raychem splices that needed to be re-shrunk. The other seven were Okonite tape splices on 120 VAC control cables. The components associated with these nonconformances were CS-MO-MO26A and RHR-MO-MO34A.

   The 250 VDC Starter Racks are located along the west side of 903'. There is no direct path from a high energy line break (HELB) onto these starter racks. Each of the two racks had two non-conforming splices; all four were Okonite taped splices. In the Division I rack, the affected equipment was the control power for RR-MO-MO53A. The Division II rack nonconformances affected the power leads for RHR-MO-MO25B.

   In PSA-ES051, none of these motor control centers nor the equipment with non-conforming splices were included as failures. This is because the temperature at the location of the nonconformances would not exceed 212F. (However, for HELBs in the Injection Valve Room, both LPCI Injection valves were already failed because of nonconformances in the room itself.) In PSA-ES056 (Enclosure B), sensitivity 2, the affected components are assumed to be failed:

   Case 1 - with a failure probability specified by the given run, or
   Case 2 - if the bulk temperature of the 903' exceeds 120F.

   PSA-ES056 contains a description of the components listed above.

5. Provide the controlled drawings and data used for your ground fault analysis and the 125 VDC fuse/breaker coordination review.

   Enclosure E provides a copy of Design Calculation No. 97-197, Rev. 1, "Low Voltage Drywell Penetration Short Circuit Withstand Calculation," which discusses the ground detection circuits. In addition, Enclosure E provides a copy of the simplified diagram of the SRV circuit used at the March 29, 2001, regulatory conference. The simplified diagram is annotated with references to the controlled drawings and documents which were the source material for creating the simplified diagram. These source drawings are listed below and a copy of each is provided in Enclosure E. These are: a) C&D drawing KBC-1225C, b) C&D drawing INV-30-70048, Sheet 2, c) Westinghouse one line drawing 152D009, d) B&R drawing 3018, Sheet 2, e) B&R drawing 3045, Sheet 14, f) B&R drawing 3048, Sheet 1, g) B&R drawing 3058, h) B&R drawing 3059, Sheets 4 and 5, i) GE drawing 791E253, Sheets 1, 2, and 3, and j) GE drawing 791E266, Sheet 10.

   Enclosure E also provides a copy of an Altran Corporation Technical Report, No. 01016-TR-001, which provides a third party review of the historical operability of the safety relief valves. This report concludes that, "in all likelihood, the ADS [automatic depressurization system] function would have been available for Reactor Coolant System small-to-medium size break mitigation." The Altran report reached this conclusion via an alternative approach (to that utilized by NPPD) that includes multiple, cumulative uncertainties, mainly in the area of margin application. Although the Altran report yielded different numerical results, it supports the NPPD conclusion that the SRVs would have remained functional. In addition, the report confirms that the assumptions made by CNS in the development of the NPPD SRV circuits evaluation are reasonable.

6. Provide the definitions and scope of your category "medium" harsh environment EQ profile.

   The NPPD EQ Program does not utilize a "medium" harsh environment for purposes of qualification pursuant to 10 CFR 50.49. However, in the context of the probabilistic safety assessment (PSA), the NPPD evaluation did not automatically assume failure for every potentially nonconforming EQ component (i.e., EQ splices associated with EQ devices) in an EQ (10 CFR 50.49) harsh environment. PSA-ES051 (provided in Reference 2) assumed failure of nonconforming components in areas away from the break only if the temperature exceeded 212F at the location of the splice. These "medium" harsh or "less than harsh" environment propagation assumptions were used in the PSA evaluation of nonconforming EQ treatments only.

   In PSA-ES051, one of the assumptions concerning splice failure in the reactor building was that a splice needed to be heated above 212F before degradation would occur. This was based on two facts. First, the non-conforming EQ splices had splice material covering the conductors. This was termed "no exposed metallics." In order for moisture to come into contact with the conductors in the splice, the material properties of the splice material would need to be challenged. The second fact was derived from various test reports of EQ splices. In these reports, the activation temperature for the splice material was listed in the 235F to 310F range. PSA-ES051 assumed that 212F was low enough that the splice material properties

would not be challenged, and consequently, performance of those splices would not be degraded in the presence of high humidity.

7. Provide a list of components that are considered to be in a "medium" harsh environment.

   Enclosure B provides the list of components that were considered to be in a "medium" harsh environment in PSA-ES051.

   The answer to this question is fairly complex because of the differences in the way the EQ program determines which equipment is in a harsh environment and how the PSA determined which components would be assumed to fail for a given break size and location.

   The EQ program qualifies equipment to a most limiting or worst case temperature profile. Limited consideration is given to the propagation of event conditions, particularly in locations when the effects of the particular break is less than the worst case.

   In PSA-ES051, it was assumed that potential nonconformances in the area of the break would fail, while those that were in other areas of the Reactor Building would perform with nominal reliability. This was based, in part, on the temperature at the location of the splice not exceeding 212F, a temperature below the point at which degradation of the splice material would begin to occur (235F - 310F range). In sensitivity 2, case 2 of PSA-ES056, the threshold was also applied to the bulk temperature of the area, rather than the specific location of the treatment. Below this threshold, there is operational data demonstrating that non-conforming and non-EQ treatments will not fail.

   The table in Enclosure B list those components that were credited in PSA-ES051 with nominal reliability but were in an area that exceeded the threshold in PSA-ES056. There is a separate table for each break location. Each table lists the component affected, the location of the non-conformance, and whether the new threshold is exceeded for a given break. An "X" on the table indicates that the non-conformance falls within the "more than mild but less than harsh" category for the given break size and location.

   The breaks are divided into isolated and unisolated scenarios. The scenarios are further divided by size of the break. The sizes follow the LOCA sizes from the PRA. For the isolated scenarios, medium breaks are included in the small group. For the unisolated cases, the small group was divided into very small (less than 2 inches equivalent diameter) and small. Large breaks were also divided, providing for a very large group (greater than 6 inches equivalent diameter). These groupings were done based on the propagation effects of the break. The basis for these groups is provided in PSA-ES056.

8. Provide the results of a sensitivity study evaluating the impact of the assumptions used in "medium" harsh environments.

   Enclosure B provides a copy of PSA-ES056, "Sensitivity Studies for PSA-ES051." This study concludes that the failure probability of systems and/or trains of equipment associated with non-conforming EQ treatments away from the HELB would all need to exceed 20% before the

results of PSA-ES051 would change, i.e., would cross the GREEN/WHITE threshold for core damage frequency.

This study also evaluates the sensitivity of the failure rate of a splice away from the HELB to the temperature of the environment. For this sensitivity, the temperature at which the splice is assumed to degrade was lowered (from 212F at the location of the splice) to a temperature where operational data has shown Cooper's splices have performed reliably (120F or 140F, depending on operating experience, in the room containing the splice). The study indicates the results of PSA-ES051 would not change, i.e., would not cross the GREEN/WHITE threshold for core damage frequency. NPPD concludes from this study that the results of PSA-ES051 are not sensitive to the HELB environment propagation assumptions used and that the evaluation in PSA-ES051 provides a realistic estimate of the significance of the as found configuration.

9. Provide qualification data and records for the Weidmuller terminal blocks in the drywell that were associated with the safety-relief valves.

   Enclosure F provides a copy of EQDP-204, Rev. 8, addressing qualification of the Weidmuller SAK-6N phenolic terminal blocks, and a copy of the CNS Operability Evaluation (OE) which addresses the Weidmuller SAK-6N melamine terminal blocks. This information is historical since the Weidmuller terminal blocks in EQ applications inside the drywell were replaced during the 2001 mid-cycle outage via design change package CED 2001-0011. Revision 8 of EQDP-204 incorporated Wyle Report 48259-01 (EQDI 3157) as the principal basis for qualification of the vertically and horizontally oriented Weidmuller SAK-6N terminal blocks made from phenolic with top entry conduits. The similarity evaluation in the EQDP provides information on how this report is used in conjunction with other test reports to establish qualification of the terminal blocks associated with the SRV solenoids and SRV tailpipe pressure switches.

   Following the revision to the EQDP for Weidmuller terminal blocks, the operability evaluation for PIR 4-12831 was revised to specifically address the SAK 6N terminal blocks made from melamine that were not covered by the revision to EQDP-204. These melamine terminal blocks were associated with the main steam isolation valve (MSIV) solenoids, the MSIV limit switches and the power circuit for RCIC-MO-MO15. These melamine blocks were not used on the SRV circuitry. These melamine terminal blocks were treated differently than the phenolic blocks qualified in EQDP-204 for the following reasons (none of which are relevant to the blocks in the SRV circuits):

   - The testing in Wyle Report 48259-01 only included Weidmuller SAK series terminal blocks made from phenolic. The terminal blocks associated with the MSIV solenoids, the MSIV limit switches and RCIC-MO-MO15 were made from melamine.

   - The revision to the OE for PIR 4-12831 also addressed the material issue raised in PIR 4-14334. This PIR was initiated to document the issues associated with the procurement and installation of melamine terminal blocks in EQ applications without the EQDP being revised to reflect the qualification of the change in material.

- The terminal blocks associated with the main steam limit switches utilized the bottom pole of a vertically oriented block. The results of the testing described in test report OHT 96-5457-1 indicated that the leakage currents to ground at the bottom terminal point can be higher than other points on the same terminal block. The testing performed in Wyle 48259-01 did not monitor leakage currents at the bottom terminal point at voltage levels that are representative of the 125 VDC system. This was not an issue for the terminal blocks qualified by revision 8 of EQDP-204, since none of them utilized the bottom terminal point.

Wyle Report 48259-01 was used in revision 2 of the OE to substantiate that the estimated leakage currents that were previously extrapolated from the results of the test programs described in Wyle 42542-1 and OHT 96-5457-1 were conservative.

10. Provide the likelihood and risk impact of distribution system level fuses (20 amp) in the 125 VDC system opening as a result of the postulated ground faults caused by the unqualified EQ treatments.

    The NPPD assessment is that the likelihood of distribution system level fuses (20 amp) in the 125 VDC system opening as a result of the postulated ground faults caused by the unqualified EQ treatments is negligible. The analysis indicates the fuse loading is well below the nominal rating of these fuses.

    PSA-ES054 (submitted with Reference 2) indicated that the risk impact of distribution system level fuses (20 amp) in the 125 VDC system opening is negligible. Case 4 in that study evaluated the failure of division 1 power to all of the SRVs. Success relied upon the transfer to the other division. The resulting core damage frequency increase was shown to be negligible.

11. Provide your assessment of the likelihood that either the Distribution Panel A fuse (100 amp) or the Switchgear 1A fuse (450 amp) would open as a result of the postulated ground faults.

    The NPPD assessment is that the likelihood of either the Distribution Panel A fuse (100 amp) or the Switchgear 1A fuse (450 amp) opening as a result of the postulated ground faults is negligible. The accident loads for these circuits are sufficiently below the nominal rating of these fuses that the addition from the postulated ground faults remains below the fuse rating.

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS

Correspondence Number: NLS2001051

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described for information only and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	

SPECTRUM TECHNOLOGIES®
UTILITIES SERVICES U.S.A., Inc.

May 14, 2001

Nuclear Licensing & Safety
Cooper Nuclear Station
Brownville, NE 68321

Tel: (402) 825-5815
Fax: (402) 825-5827

Subject: Your Purchase Order 4500014779 - Proprietary Information in Test Report #TR01F0410

Dear Mr. Grant:

Under the subject purchase order, Spectrum Technologies Utilities Services USA, Inc. conducted functional testing on Bussmann MIN-10 fuses. The results of this testing have been documented in Test Report #TR01F0410, Rev. 0, dated March 27, 2001, which has been identified as "Spectrum Proprietary" by Spectrum, stating: "The contents of this procedure are exclusive property of Spectrum Technologies USA, Inc. and shall not be used by or divulged to anyone without prior written consent of Spectrum Technologies USA, Inc." (Spectrum Technologies USA is synonymous with Spectrum Technologies Utilities Services USA)

Spectrum understands that it is necessary for you to share the contents of the Test Report #TR01F0410 with the Nuclear Regulatory Commission (NRC) in support of their review and evaluation of the testing performed and the results therefrom. To this end, you are hereby authorized to release Test Report #TR01F0410 to the NRC. This release applies solely to providing the subject information to the NRC. If any other release of this information to any other third party, specific written consent must be obtained from Spectrum Technologies Utilities Services USA, Inc.

If you have any questions, please don't hesitate to call, fax or e-mail us.

Thanks, and have a great day.

Very truly yours,

William R. Willis
Vice President, QA

Spectrum Sets Standards Above The Standards

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ENCLOSURE A

Environmental Qualification of the SRV Pilot Solenoid Valves

(Related discussion is provided in Attachment 1)

Includes:

Environmental Qualification Data Package
EQDP-253, Rev. 5
(A1) - (28 pages)

Selected Pages from
Test Report No. 5074 (EQDI 2580)
Cover and Transient Profile Data
(A2) - (2 pages)

and

Selected Information re:
Conformal Coating Evaluation
(A3) - (20 pages total)

TERI-033 (EQDI 3170) (3 pages)
Patel/EGS Conformal Coating (PECC) Fact Sheet (1 page)
EGS PECC Material Safety Data Sheet (6 pages)
CHEMTRONICS Technical Data Sheet (2 pages)
CHEMTRONICS KONFORM SR Material Safety Data Sheet (2 pages)
Information on use of CAS Registry for substance identification (6 pages)

A1

EQUIPMENT QUALIFICATION DATA PACKAGE

PACKAGE NUMBER: 253 EQUIPMENT TYPE: Solenoid Valves

MODEL NUMBER: ½-SMS-S-02-1 MANUFACTURER: Target Rock

PREPARED BY: ______ DATE: ______

CHECKED BY: ______ DATE: ______

APPROVED BY: ______ DATE: ______

Revision 1: 2-21-92

Revision 2: 5-15-97

Revision 3: 06-30-98

Revision 4: 11-05-99

Revision 5: 12-14-99

RECORD OF REVISION

PACKAGE NUMBER: 253 EQUIPMENT TYPE: Solenoid Valves

MODEL NUMBER: ½-SMS-S-02-1 MANUFACTURER: Target Rock

REVISION NUMBER	DATE	REVIEWED BY	CHECKED	APPROVED	EFFECTIVE PAGES	DESCRIPTION OF CHANGES.
1	2/21/95				8, 18	Revised pressure requirement.
2	5-15-97				5 22	Added Reference 21. Added Reference 21 summary.
3	6-30-98				5 27 General	Retitled Reference 5, Added Reference 22. Correct table header. Reformatted, corrected typos.
4	11-05-99				26-28 26-27	Moved page 28 to page 29 Addressed IEN 82-41, 83-22, 86-12, 86-57, 88-30, 88-30S1, and 90-11
5	12-14-99				27	Addressed Part 21 Log No. 1997-43 and 1995-049

EQUIPMENT QUALIFICATION DATA PACKAGE

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I. Equipment Qualification Maintenance Requirements

A. Purpose

The purpose of this evaluation is to determine if the qualification documentation available for the Class 1E equipment in the Cooper Nuclear Station provides reasonable assurance that it will perform its required function in the specified accident environment.

B. Procedure

The evaluation will be performed in accordance with the requirements set forth in 10CFR50.49 and IEEE 323-1974.

C. References

1. Accident Environments, EQDP 46.

2. Master List Response to Plant Operators Comments, EQDI 2139.

3. Target Rock Technical Manual on Safety Relief Valve Model 7567F, dated November 1993, D.I. 2363.

4. Qualification Test Report for The Environmental Qualification Of The Target Rock Corporation Three Way Valve, Solenoid Operated P/N ½ SMS-S-02-1, Test Report 5074, dated 1-12-90 , D.I. 2580.

5. CNS Master Equipment List, D.I. 2139.

6. IEEE 323-74, Standard for Environmental Qualification of Class 1E Equipment, D.I. 1409.

7. Environmental Qualification Evaluation of The Target Rock 7567F 2-Stage SRV With Improved Actuator Assembly; GE AR 404-LIL08-HPI-92; Dated 11/15/93, DI 2579.

8. Beta Shielding Evaluation, EQDP 72.

9. Design Change 92-043, SRV Solenoid Valve Design Change, Dated 12/5/94, DI 2596.

10. NPPD Letter to B.D. Maytum (Impell) From J.R. Hackney, Dated 2/8/84, DI 1307.

11. Target Rock Letter to S. Jobe (NPPD/CNS) from K. Wenzel, dated 10-12-83, NPPD P.O. 216553, D.I. 1314.

12. SWEC Radiation Calculations, SWEC Report 13095.31 Rev. 1, 8-15-85, D.I. 2082.

13. Accident Profiles for EQ, 8-17-85, D.I. 2097.

14. CNS Engineering Procedure 3.12.2 Rev. 9 dated 8/4/94, D.I. 2200.

15. CNS DC 86-140, "EQ Flooding Modifications", dated 12-26-86, D.I. 2300.

16. 10CFR50.49, "Environmental Qualification Of Electrical Equipment Important To Safety For Nuclear Power Plants," Final Rule, Dated 1/21/1983, DI 1407.

17. SAND-80-17466, "Occurrence And Implication Of Radiation Dose Rate Effects For Material Aging Studies," K.T. Gillen and R.L. Clough, Sandia National Laboratories, DI 2388.

18. "Interim Staff Position On Environmental Qualification Of Safety Related Electrical Equipment," NUREG 0588, DI 1408.

19. SWEC Calculation 15798.17-UR-011, Summary Dose Data for MEL, Rev.8, DI 2423.

20. Final Test Report On Patel Engineers Conformal Coating (PECC), Technical Report No. PEI-TR-850400-2, Dated 12/21/84, DI 2536.

21. Wyle Test Report 40951-00, "Viton Silicons, and EPR O-Rings Assessment Report," dated 2/15/90, D.I. 2829.

22. Target Rock Test Report 2199A, 1-9-79, D.I. 0403

D. Equipment Description

1. Manufacturer:

   Target Rock

2. Component:

   Solenoid Valves

3. Model Number(s):

   ½-SMS-S-02-1

4. System(s):

   Main Steam

5. Plant Identification Number(s):

MS-SOV-SPV71A	MS-SOV-SPV71B	[5]
MS-SOV-SPV71C	MS-SOV-SPV71D	
MS-SOV-SPV71E	MS-SOV-SPV71F	
MS-SOV-SPV71G	MS-SOV-SPV71H	

6. Location:

   Containment - Drywell 921' [5]

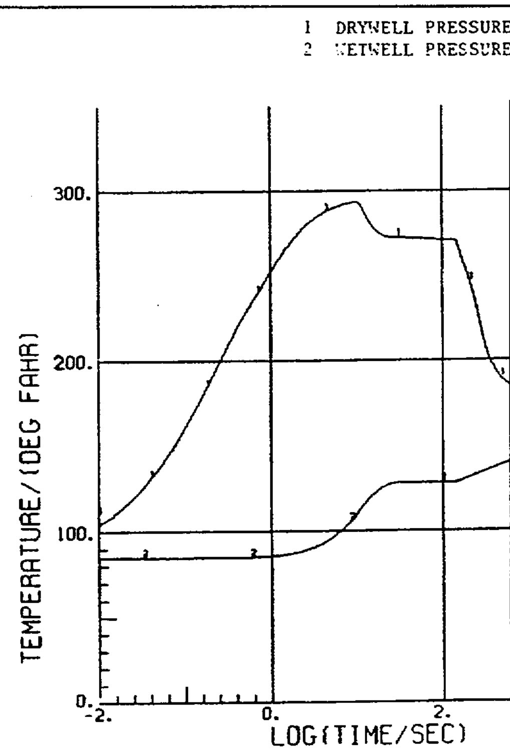
E. Environmental And Functional Parameters

1. Normal Environment

Parameter		Reference
a.	Installed Life:	
	10.0 years	[4]
b.	Temperature:	
	150°F	[1]
c.	Pressure:	
	-0.5 + 2.0 psig	[1]
d.	Relative Humidity:	
	90% maximum	[1]
e.	Radiation:	
	8.8×10^6 R Gamma	[12]

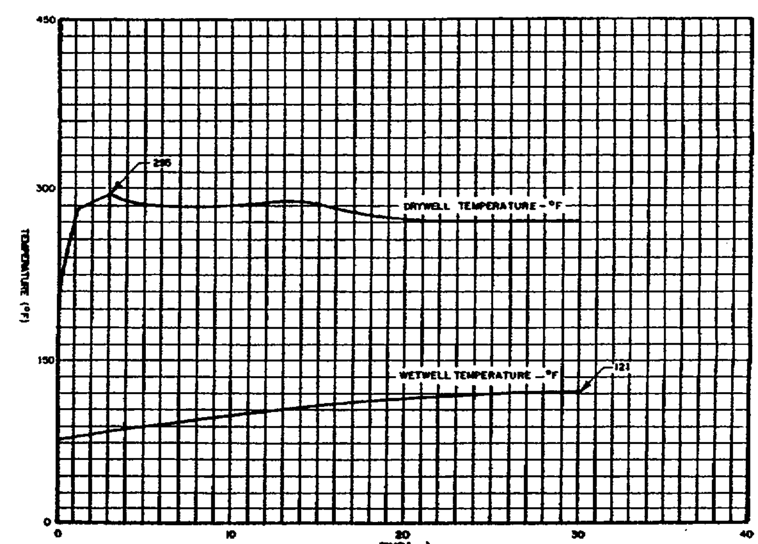
2. Accident Environment

	Parameter	Reference
a.	Limiting Accident:	
	LOCA.	[1]
b.	Operating Time:	
	6 months.	[10]
c.	Temperature:	
	310°F (Includes margin, see Fig. 1)	[1]
d.	Pressure:	
	63.8 psig (Includes margin, see Fig. 2)	[1]
e.	Relative Humidity:	
	100%	[1]
f.	Chemical Spray:	
	D.I. Water.	[1]
g.	Radiation:*	
	1.30×10^7 R Gamma	[19]
	2.0×10^8 R Beta.	[1]
	*Radiation values include normal dose, plus accident, plus 10 percent margin.	
h.	Submergence:	
	N/A	[15]



Nebraska Public Power District
COOPER NUCLEAR STATION
UPDATED SAFETY ANALYSIS REPORT (USAR)

Original Short-Term Primary Containment Pressure and Temperature Response Following a Loss of Coolant Accident
Figure XIV-6-20 07/22/94



Nebraska Public Power District
COOPER NUCLEAR STATION
UPDATED SAFETY ANALYSIS REPORT (USAR)

DBA Containment Temperature Response
Figure XIV-6-19
7/22/83

Figure 1

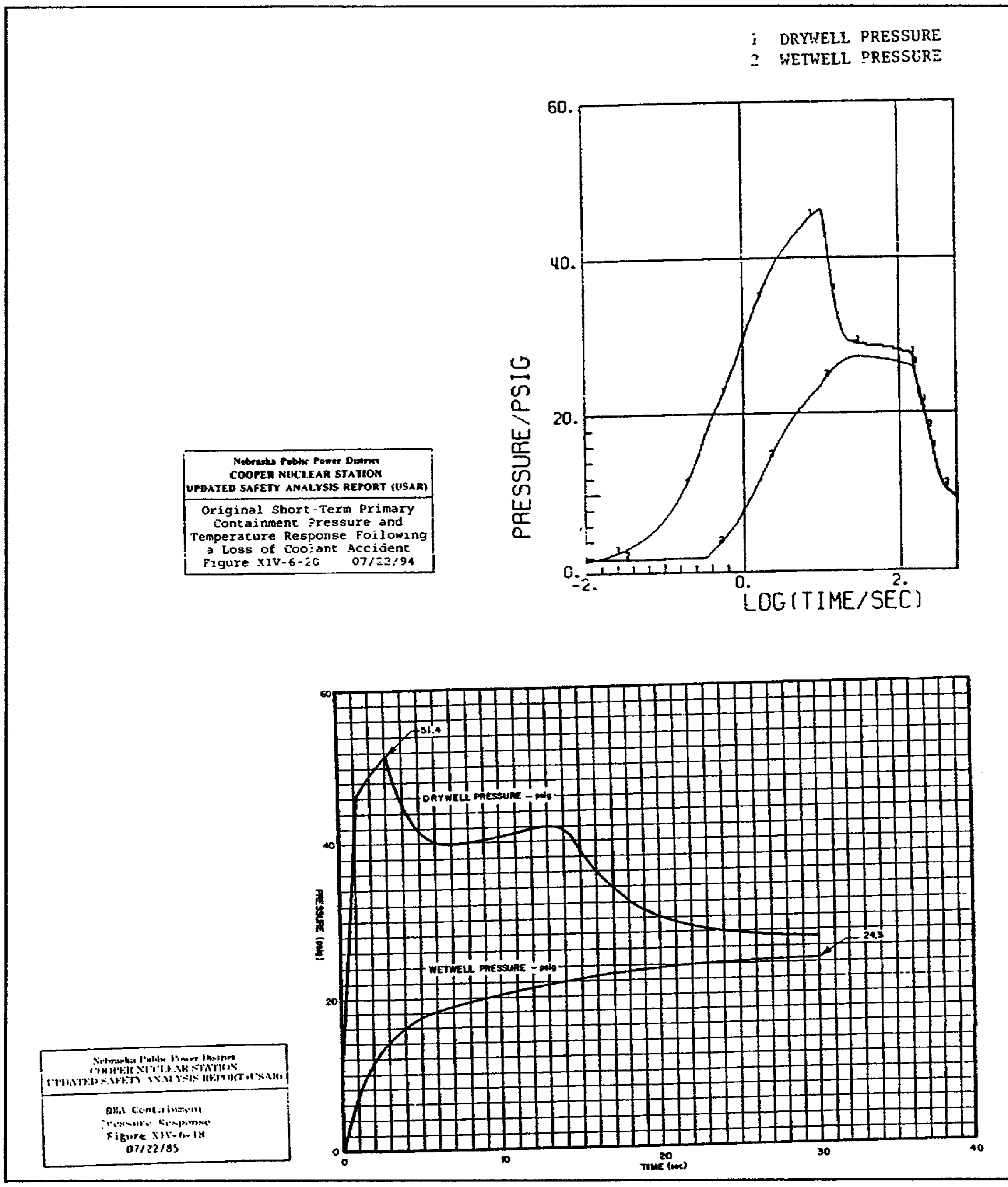

1 DRYWELL PRESSURE
2 WETWELL PRESSURE
PRESSURE/PSIG
LOG(TIME/SEC)
Nebraska Public Power District
COOPER NUCLEAR STATION
UPDATED SAFETY ANALYSIS REPORT (USAR)
Original Short-Term Primary Containment Pressure and Temperature Response Following a Loss of Coolant Accident
Figure XIV-6-20 07/22/94
51.4
DRYWELL PRESSURE – psig
WETWELL PRESSURE – psig
24.3
PRESSURE (psig)
TIME (sec)
DBA Containment
Figure XIV-6-18
07/22/85

Figure 2

3. Normal Functional Requirements:

   Pilot for Safety Relief Valves - main steam lines A-D.

4. Accident Functional Requirements:

   Pilot for Safety Relief Valves - main steam lines A-D.

F. Equipment Evaluation

This evaluation will be performed in accordance with the requirements set forth in 10CFR50.49 and IEEE 323-1974. [16,6]

Reports:

Qualification Test Report for The Environmental Qualification Of The Target Rock Corporation Three Way Valve, Solenoid Operated P/N ½ SMS-S-02-1, Test Report 5074, dated 1-12-90. [4]

1. Test Specimen Evaluation:

The tested solenoid assembly was model ½ SMS-S-02-1. This solenoid valve is designed to operate at 125 VDC and up to 300 psig. Two three way valves were bolted to a plugless manifold using Viton O-ring face seals. The manifold was then bolted to the air operator. The tested SRVs utilize two ½ SMS-S-02-1 solenoids. [4]

2. Similarity of Tested Equipment to Installed Equipment:

The equipment tested is a Target Rock Corp. ½-SMS-02-1 solenoid valve assembly and associated manifold assembly as shown in Reference 4. Each CNS SRV uses one ½-SMS-02-1 solenoid valve mounted on a single solenoid manifold. The test program was performed on two Target Rock ½-SMS-02-1 solenoid assemblies mounted on a dual solenoid manifold as depicted in Reference 4, Pages 335 to 337. The air operator was not qualified in the test program.

Three SRVs, complete with air operator, air manifold and solenoid valve, were purchased through General Electric (CNS Agreement No. 86A-MS2, Task Authorization 292). These valves had been previously supplied to the Shoreham plant however, these valves had not seen plant service. These three valves are identical to those supplied directly to CNS by Target Rock Corp. and are described in Ref. 7.

The remaining nine SRVs were purchased under Contract 94-62 with a Certificate of Conformance/Compliance to Target Rock Test Report 5074.

A single solenoid is basically ½ as heavy as two solenoids. Therefore, the CNS configuration incurs lower stresses from dynamic loads. The single and dual manifold assemblies use the same seal (Viton and PEEK) and coil materials. They are therefore equal in their capability. The single vs. dual solenoid difference between the CNS assembly and the test assembly does not impact the qualification.

Conduit seals are not required at the 3/4" NPT entrance to have a qualified configuration. The test program used an open raceway at the 3/4" NPT entrance port.

Appendix IV to Reference 4 requires the wires, terminals and terminal boards to be sprayed with a conformal coating to deposit a protective covering of silicone resin on the electrical connection. Appendix VIII to Reference 4, states that Chemtronics "Konform" Silicone resin be applied to the terminal board and connections. The solenoids installed at CNS use PECC Conformal coating (Ref. 20). Reference 20 describes the qualification program for PECC. The qualification program meets or exceeds all CNS environmental requirements except for accident pressure. The PECC was tested at a maximum pressure of 30 psig while Figure 2 shows a peak pressure of 51.4 psig. During accident conditions the higher pressure is not considered to be a potential failure mechanism for PECC, since the PECC adheres to the surface of the components no pressure differential is experienced across the conformal coating. Therefore, the use of PECC on the terminal boards and connections for the solenoids is considered acceptable.

3. Test Sequence Evaluation:

- Pre-Test Functional Test
- Radiation Exposure
- Functional Test
- Thermal Aging Test
- Thermal Transient Test
- Functional Test
- Pressurization Aging Test
- Mechanical Aging Test
- Functional Test
- Vibration Aging Test
- Cyclic Load Aging Test
- Design Basis Event (DBE)
- Seismic test
- Post Seismic Functional Test
- DBE Radiation Exposure
- DBE Environmental Simulation Test
- Post-Test-Final Functional Test

The test sequence follows the methods outlined in IEEE 323 (radiation exposure was performed prior to thermal aging to account for any synergistic effects) and is, therefore, considered satisfactory. [6]

4. Service Conditions.

a. Operating Time.

1) Required:

6 months (180 days) [10]

2) Reported:

100 days.

3) Discussion:

A comparison of the Test Profile (Figure 3) and the CNS Postulated Accident Profile (Figure 1) shows that the peak temperatures are enveloped by the test. Figure 3 shows that the test profile envelopes the required profile (See Ref. 4, page 185 and Ref. 7, page A2-3) for the 100 days duration of the test with extensive margin however, the post-accident phase is not enveloped for 180 days by the test conditions. Since the materials follow an Arrhenius relationship, the requirements at one time and temperature can be transferred to another set of time-temperature coordinates using the relationship:

$$t_1 = \sum_{x=2}^{n+1} t_x/\exp\left((Ea/k_B)(1/T_x - 1/T_1)\right)$$

where,

t_1 = equivalent time at T_1

T_1 = reference temperature

t_x = time at temperature T_x

T_x = accident temperature above T_1

exp = exponent to base e

Ea = activation energy (eV)

k_B = Boltzmann's Constant (8.617×10^{-5}eV/K)

A time-temperature equivalency can be derived to show that the test conditions exceed the plant postulated accident conditions, including the post-accident phase.

Using the above equation the entire set of parameters given for the plant accident (Figure 1) can be transferred to an equivalent time at 150°F.

Using the Arrhenius parameters in the above equation and the accident test profile of Figure 3, the total equivalent time at 150°F for the material with the lowest activation energy was calculated.

Using an activation energy of 1.11 eV for Viton and the Temperature Profile of Fig. 1, the Postulated Plant Accident Profile Equivalent Time at 150°F was calculated as follows:

Temperature	Time (Hours)	Equivalent Time At 150°F (Hours)
296°F (419.8 K)	.007	10.9
275°F (408.2 K)	.1	64.9
250°F (394.3 K)	.1	21.3
180°F (355.4 K)	27.8	166.0
170°F (349.8 K)	60	200.6
150°F (338.7 K)	4656 (Includes Margin)	4656

This yields 213.3 days at 150°F.

Using an activation energy of 1.11 eV for Viton and the test profile as shown in Fig. 3, the Test Profile Equivalent Time at 150°F was calculated as follows:

Temperature	Time (Hours)	Equivalent Time At 150°F (Hours)
355°F (452.6 K)	3	43,031.6
335°F (441.5 K)	3	21,039.0
265°F (402.6 K)	18	7,531.0
215°F (374.8 K)	99	3859.6

This yields 3,144.2 days at 150°F.

Material	Test Profile Equivalent Time at 150°	Postulated Plant Accident Profile Equivalent at 150°
Viton (Ea = 1.11 eV)	3,144.2 days	213.3 days

Thus, using the lowest activation energy method, it is demonstrated that the accident tests exposed the nonmetallic

materials to a greater thermal degradation than the postulated plant accident, plus margin.

b. Temperature and Duration.

1) Required:

310°F (Includes 15°F margin) See Figure 1 [1]

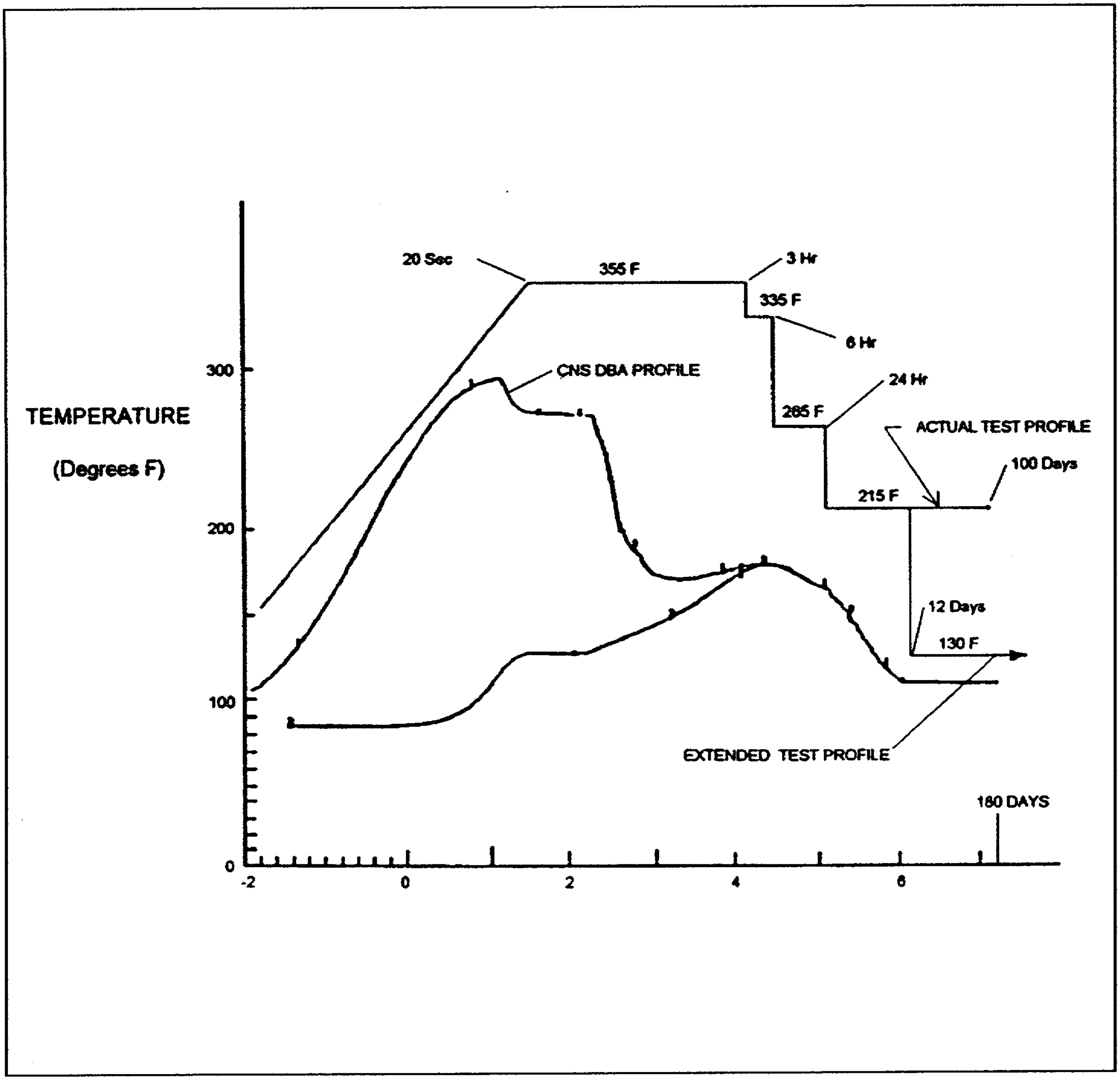
2) Reported:

355°F (See Figure 3) [4]

3) Discussion:

The test temperatures are 45°F higher than the required peak temperature plus margin.

Therefore, temperature requirements are satisfied.



Log Time (sec)

Figure 3

c. Pressure and Durations.

1) Required:

63.8 psig (Includes margin, see Figure 2) [1]

2) Reported:

68.2 psig (See Figure 4) [4]

3) Discussion:

The test pressures exceed the postulated plant accident pressures. Therefore, the pressure condition is considered satisfactory.

d. Relative Humidity.

1) Required:

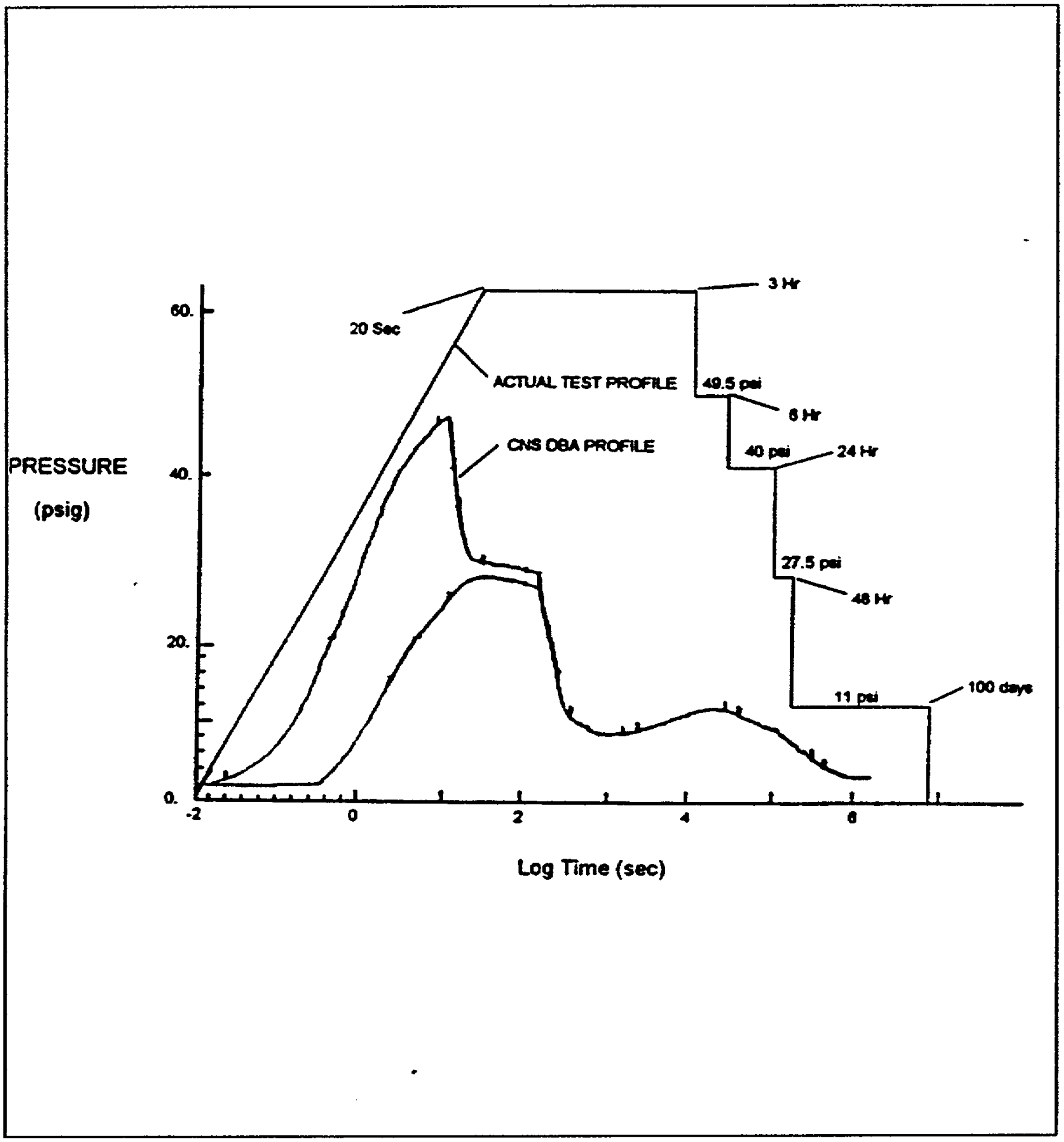
100% [1]

2) Reported:

100% [4]

3) Discussion:

Therefore, the RH is satisfactory.



Log Time (Sec)

Figure 4

e. Chemical Spray.

1) Required:

Demineralized water [1]

2) Reported:

Demineralized water [4]

3) Discussion:

Simultaneous testing, as described in Reference 4, satisfies the chemical spray requirements.

f. Radiation.

1) Required:

1.30×10^7 R Gamma T.I.D. [12]
2.0×10^8 R Beta [1]

2) Reported:

1.0×10^7 R Gamma T.I.D. [4]

3) Discussion:

The DBA radiation dose of 1.3×10^7 R Gamma contains the 40 year normal dose of 8.8×10^6 R. Since the actuator is only qualified to 10 years, 30 years of the normal dose (6.6×10^6 R) may be subtracted from the total. This leaves a 10 year normal dose plus 110% of the Accident dose (3.86×10^6 R) for a total dose of 6.45×10^6 R Gamma. [12]

Beta radiation is not a concern for the actuator, since all nonmetallics are within metallic enclosures capable of shielding the susceptible parts from Beta radiation. [4]

Therefore, the Gamma and Beta radiation requirements are satisfied.

g. Aging.

1. Mechanical Wear Aging.

a) Required:

The valves are only cycled during SRV testing (conducted during reactor startup) and during SCRAM operation. A conservative cycling requirement of 8 cycles/year is postulated.

b) Reported:

900 cycles [4]

c) Discussion:

The SOVs are required (by this data package) to be rebuilt every 10 years. If the entire 10 years is considered, a total of 80 cycles between rebuilds results. During rebuilding, any signs of wear are noted and corrected. The result is that the rebuilt valve is equivalent to a brand new valve. Therefore, since the cycling requirement is less than 9% of the tested value, the mechanical aging requirements are satisfied.

2. Radiation Aging.

   See Section 6.4.6 for discussion of Radiation.

3. Thermal Aging.

   a) Required:

      See Discussion.

   b) Reported:

      10 years at 150°F ambient. [4]

   c) Discussion:

      All of the age sensitive non-metallic materials are identified in Table III-1 of Appendix V of Reference 4. The most limiting non-metallic (lowest activation energy) is Viton (Ea. = 1.11 eV). The Arrhenius method was used to compare materials test data and specified conditions at installed locations. The equivalent accident aging was subtracted from the service life to yield a qualified life. All aging calculations are based on 150°F normal ambient temperature, and thermal aging of the test specimens for 1356 hours at 342°F. As reported in Reference 4 the heating effects of steam in contact with the SRV and the ambient temperature of 150°F produce a temperature of 235°F at the solenoid. The calculations performed in Reference 4 use 240°F as the temperature at the solenoid, thus including a 5°F margin.

      Wyle Test Report 40951-00 verified the suitability of Silicone Rubber and EPR O-Rings as replacements for Viton. O-Rings made from Silicone Rubber are qualified for greater than the 10 years specified and for a total integrated radiation dose (TID) of greater than 5.0E7 rads. EPR O-Rings are qualified for 10 years when the service temperature can be reproduced to less than 205°F for TID of greater than 5.0E7 rads. [21]

      Therefore, the Target Rock SOVs, non-metallics, are qualified for 10 years of service at CNS. Target Rock SOV metallics are qualified for 40 years of service at CNS.

h. Synergistic Effects.

a) Required:

Applicable

b) Reported:

See below

c) Discussion:

Radiation aging was performed prior to thermal aging as discussed in Reference 4. This sequence is generally preferred to accommodate unknown synergistic effects. Additionally, the radiation dose which the SOV was subjected to exceeded the required dose, including DBA+10%, by 55%. Therefore, the test program has accounted for possible synergistic effects.

i. Submergence.

a) Required:

Not applicable

b) Reported:

None

c) Discussion:

Submergence is not postulated for the SOV at CNS.

5. Functional Testing:

The valves were cycled 900 times through part of the aging simulation at 6 cycles/min and cycled 120 times during the accident simulation test.

Therefore, the specimen was successfully tested for a total of 900 + 120 cycles. [4]

The valves were operated during all phases of the test to demonstrate the integrity of the test specimens. Therefore, functional testing requirements are satisfied.

This component is a "primary" component in that it actually performs a required safety-related function, as opposed to a "support" component which merely supports the operation of this primary component. Support components makeup the balance of the circuit loop of the primary component. The performance of this component as installed using its support components have been analyzed (D.I. 2407). The results of this analysis demonstrates that this component as installed in its circuit loop at CNS will perform its safety-related function. The support components have been analyzed to determine their effect on the circuit under accident conditions. The circuit has been show to be capable of supporting the performance of the primary component's safety-related function.

During the production tests, the valves assemblies were actuated at 60 VDC at room temperature to simulate derated conditions at high temperatures.

During the thermal transient part of the thermal aging test the valves were each actuated 32, 32, and 36 times at 92.5, 125, and 154 VDC respectively while the ambient temperature varied from 220°F to 265°F.

During the accident simulation test, each valve was actuated at least 20 times at voltages ranging from 92 to 154 VDC.

During the accident simulation the electropneumatic actuator assembly was supplied with air at pressures up to 260 psig.

G. Conclusion.

The preceding qualification evaluation has shown that the equipment described in Section 4 of this report is qualified for its intended use. This evaluation was performed in accordance with the requirements set forth in 10CFR50.49 and IEEE 323-1974.

The SOVs are qualified for 40 years of service with scheduled rebuilds at 10 year intervals. All non-metallics shall be replaced during rebuild.

It is recommended that Surveillance Tests and Inspections specified by the manufacturer in Reference 4 be performed at the specified time intervals.

It is recommended that the manufacturer's specified surveillance and maintenance inspection intervals/tests outlined in Reference 4 be followed (especially leak testing).

H. Response to NRC IE Notices/Bulletins.

Applicable X Not Applicable ____ IE Notice No. 84-68,80-25

Disposition:

84-68 - Degradation of SOV pigtail leads.

The Target Rock SOVs at CNS are supplied with potted, Kapton insulated pigtails. Kapton is one of the most heat and combustion resistive polymers in the industry. It is not subject to thermal degradation induced failure.

80-25 - GN_2 Instrument supply regulation for MSRV controls.

Three safeguards have been taken to assure that excessive pressure (135 psig) does not accumulate in the GN_2 instrument control lines:

1) The nitrogen supply pressure regulator has been set at 110 psig.

2) Two relief valves on the GN_2 supply (downstream of the pressure regulator) have been set at 120 psig.

3) The backup pneumatic instrument air supply relief valves at the instrument air dryers have been set at 130 psig. These three safeguards insure that the instrument supply GN_2 pressure is within safe limits (135 psig).

Applicable ____ Not Applicable X IE Notice No. 82-41

Disposition:

Valves affected by this notice are no longer used at CNS.

Applicable ____ Not Applicable X IE Notice No. 83-22

Disposition:

The model affected by this notice is not part of the CNS EQ program.

Applicable ____ Not Applicable X IE Notice No. 86-12

Disposition:

The model directly affected by this notice is not used at CNS. There is surveillance in place for similar valves.

Applicable X Not Applicable ____ IE Notice No. 86-57

Disposition:

This issue was addressed and the valves with the affected elastomers were replaced.

Applicable ____ Not Applicable X IE Notice No. 88-30 & 88-30S1

Disposition:

The SOVs affected by this notice are not used at CNS.

H. Response to NRC IE Notices/Bulletins (continued).

Applicable ____ Not Applicable X IE Notice No. 90-11

Disposition:

There are proper maintenance procedures in place for CNS SOVs.

Applicable ____ Not Applicable X Part 21 Log No. 1997-43

Disposition:

A Target Rock MSLSRV failed testing at Wyle Laboratories. The failure was caused by a screw jamming the valve. The presence of the screw was determined to be due to insufficient FME by Wyle during handling of the valve. Corrective action has been taken to monitor Wyle's testing procedure. This does not affect CNS as valves are tested and set individually, which ensures proper installation.

Applicable X Not Applicable X Part 21 Log No. 1995-049

Disposition:

The ADS system failed during startup in February of 1995. Corrosion was found to be binding the valves. After consulting with the vendor it was determined that the valves may not have been appropriately dried. CNS now has procedures in place for refurbishing SOV/SRV that have precautions incorporated to preclude this issue.

I. Equipment Qualification Maintenance Requirements

Maintenance Item	Source Document	CNS PM Number	Comments
Rebuild/Replace			
MS-SOV-SPV71A	EQDP 253	00335	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71B	EQDP 253	00336	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71C	EQDP 253	00337	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71D	EQDP 253	00338	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71E	EQDP 253	00339	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71F	EQDP 253	00340	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71G	EQDP 253	00341	M.P. 7.2.22/7.2.22.1
MS-SOV-SPV71H	EQDP 253	00342	M.P. 7.2.22/7.2.22.1

Replace after 10 years of service:

Electrical Components including solenoid coil and terminal boards.

Elastomers seals and P.E.E.K. Valve Seat Inserts.