Request for Additional Information on
Probabilistic Risk Assessment
Extended Power Uprate License Amendment Application
Arkansas Nuclear One, Unit 2

1. Totaling up the individual impacts for the fire analysis, ANO-2 shows a change in core damage frequency (CDF) of 1.6E-5, with a base CDF value of just over 1E-4. This is in Region I on the chart in Regulatory Guide (RG) 1.174, where "applications ... would not normally be considered." Please provide additional discussion and any additional analyses to justify why these high resulting values are acceptable and/or describe any mitigative or compensatory features that would reduce the major risk contributors (i.e., Cable Spreading Room, Diesel Corridor, Lower South Electrical/Piping Penetration Room, North and South Switchgear Rooms, MCC2B63 Room, etc.). Many of these impacts seem to be due to operator recovery actions available times, which were determined using the CENTS code by calculating the time to core uncovery as opposed to the time to core melt. Thus, the resulting human error probabilities (HEPs) have high, conservative values. What other conservatisms in the modeling may account for the resulting high fire CDFs? How would the results be affected by using the time to core melt and removing these other conservatisms from the CENTS code? If possible, the licensee should recalculate the pre and post extended power uprate (EPU) fire analysis CDF removing the over-conservatisms and may take credit for the additional mitigative and/or compensatory actions. If the licensee does recalculate these values, the analysis, including any assumptions and actions credited, should be described.

2. The licensee indicated that the potential for creating an initiating event due to a spurious main steam isolation signal (MSIS) or containment spray actuation signal (CSAS) is compensated by trip hardening their signals. Though this modification is argued to compensate for the potential increases in spurious signals, it is stated that it is not quantified. How are these signals addressed in the ANO-2 probabilistic risk assessment (PRA) models? Does the ANO-2 EPU PRA explicitly model these signals as designed (and to be installed), considering their benefits (i.e., reduced frequency due to trip hardening) and potential adverse impacts (e.g., spurious operation) on initiating event frequency and following an initiating event? If not, will these signals be incorporated as part of a future update of the model and is this planned update prior to entering EPU operations?

3. The information states in a couple places that the uprate could cause components to wear out more quickly or involve more often preventive maintenance. How did the licensee address these conditions within the EPU PRA model? Were failure rates and/or maintenance outage rates increased for selected equipment that would be affected by the EPU? If so, please identify the equipment affected and provide the old and new failure rate/maintenance outage values (or if multiple components were increased by a proportional amount, provide the percentage increase). If not, please briefly explain why not and the basis for the acceptability of the potential increases in equipment being unavailable due to maintenance without modeling them in the EPU PRA (e.g., maintenance times used in model bound EPU projected maintenance times).

4. For shutdown operations, what is the shortest "time to boiling" calculated during a typical outage and when in the outage does this occur (e.g., mid-loop operations)? Describe other typical shutdown operations in which the containment cannot be closed within the estimated "time to boiling." For these shutdown operations and any other times of extremely short "time to boiling" duration, does ANO-2 take any additional precautionary/mitigative actions other than those cited in their response of June 28, 2001?

5. Is all equipment operated within its rated design capacity (e.g., transformers, switchgear, pumps, etc.)? If not, please identify the equipment operated beyond its rating and state why the equipment operations are acceptable (e.g., operator actions required to manually load shed overloaded transformer within a set time). If there are operator actions involved in the actions to protect the equipment, what are these actions and have they been assessed and incorporated into the EPU PRA?

6. The operator action available times affected by the EPU are expected to change inversely proportionally with the increase in decay heat resulting from the EPU. However, many of the available times for operator actions listed by the licensee decrease by a larger percentage (17-23%) than expected, considering the EPU is only a 7.5% increase. What is the reason for these larger than expected decreases in available times? If this is related to the conservatisms identified in Item 1 above, how would the results be affected by using the time to core melt instead of time to core uncovery in the CENTS code? If possible, new HEPs should be calculated for these operator actions and their impacts assessed, similar to Item 1 above.

Post-it® Fax Note 7671	9/17/01 pages 2
To TOM ALEXION	From DANA MILLAR
Co./Dept.	Co. ENTERGY
Phone #	Phone #
Fax #	Fax #

Response to NRC RAI on Eliminating Movem

Question:

If CEA #43 is assumed to remain fully withdrawn when a reactor trip occurs, is Shutdown Margin maintained?

Response:

The Technical Specification (TS) definition of Shutdown Margin (SDM) is:

> Shutdown Margin shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

TSs 3.1.1.1 (Modes 1 through 4) and 3.1.1.2 (Mode 5) specify that SDM shall be greater than or equal to that specified in the Core Operating Limits Report (COLR). The SDM operating limit, which is included in the COLR for Cycle 15 is 5.0 % Δk/k in Modes 1 through 5.

If a reactivity balance was performed for the Cycle 15 core with a known stuck CEA (e.g., a pair of stuck CEAs) in the current operating conditions (~250 EFPD), the TS required SDM would not be met. The SDM is 4.385 % Δk/k with the worst case two stuck CEAs. However, there is conservatism built into the calculation of the SDM. The following discussion is provided to demonstrate some of that conservatism. It should be noted that the following discussion is invisible to the operator in the control room. Emergency operating procedures require emergency boration of the RCS at a minimum rate of 40 gpm with 2500 ppmB if one or more CEA fails to fully insert to the core. Reactivity control is of primary importance in accident mitigation.

The only term in the SDM calculation that would change due to the proposed situation is the worth of the CEAs that do insert to the core. The worth of the fuel, boron, xenon and net samarium is 2.282% Δk/k. At 250 EFPD, the worth of the "worst stuck pair" of CEAs (the combination of the highest worth CEA with another CEA whose total reactivity impact is the greatest) is 3.419 % Δk/k. The total pattern worth of all the CEAs is 10.086 % Δk/k. Therefore, the SDM assuming the "worst stuck pair" would be 4.385 % Δk/k.

CEA #43 is not the single CEA of highest reactivity worth in the Cycle 15 core, nor does it make up part of the "worst stuck pair". If CEA #43 plus one more rod were assumed stuck out, then the worst possible resulting pair would be worth 2.880 % Δk/k. Substituting this value for the worth of the worst stuck pair discussed previously the resulting SDM would be 4.924 % Δk/k.

The reactor physics methods used to establish the guidelines for the development of the reactivity balance calculation are contained in Topical Report ENEAD-01-P-A, Revision 0,

"*Qualification of Reactor Physics Methods for Application to Pressurized Water Reactor of the Entergy System.*" The guidance calls for the addition of - 0.155 % Δk/k as margin. For conservatism, Entergy has applied a multiplication factor of 1/1.05 to the total CEA pattern worth instead of subtracting 0.155 % Δk/k. If the 5% conservatism is removed from the Total Pattern Worth and then 0.155 % Δk/k is subtracted from the raw Total Pattern Worth as called for in the Topical Report's 95/95 Reliability Factor, then the total pattern worth would be 10.435 % Δk/k. This results in an "extra margin" of 0.349 % Δk/k in the pattern worth. If this is accounted for when the worst possible pair including CEA # 43 are stuck out, then SDM would be 5.273 % Δk/k, which would satisfy the COLR requirement.

Furthermore, between 250 and 500 EFPD (end of cycle), the additional margin in the worst stuck pair that includes CEA #43 versus the two used in the reactivity balance calculation will increase from 0.539 to 0.826 % Δk/k and the margin in the Total Pattern Worth will increase from 0.349 0.389 % Δk/k. Therefore, it can be shown analytically that SDM in excess of the COLR limit will be present at all times for the remaining portion of Cycle 15 if CEA #43 fails to insert into the core.

Request for Additional Information on
Fuel Cladding Corrosion
Extended Power Uprate License Amendment Application
Arkansas Nuclear One, Unit 2 (ANO-2)

1. The recent experience from Calvert Cliffs has shown that the cladding corrosion is worse in the high-burnup regime and is consistently underestimated by the CENP corrosion model. Provide updated information of corrosion during power uprate and assess the potential impact for fuel operation at ANO-2.

(1)

NEBRASKA PUBLIC POWER DISTRICT

Cooper Nuclear Station

Nuclear Licensing and Safety Department
P.O. Box 98
Brownville, NE 68321-0098

FAX
(402) 825-5827

Verification
(402) 825-5117

TO: M. Thadani (x1476) FAX: 301-415-3061

COMPANY: NRC

FROM: Eddie Grant (402) 825-5815

DATE: 9-10-01 # OF PAGES (including cover) 8

Information for discussion with Tech. Spec. Branch.

(2)

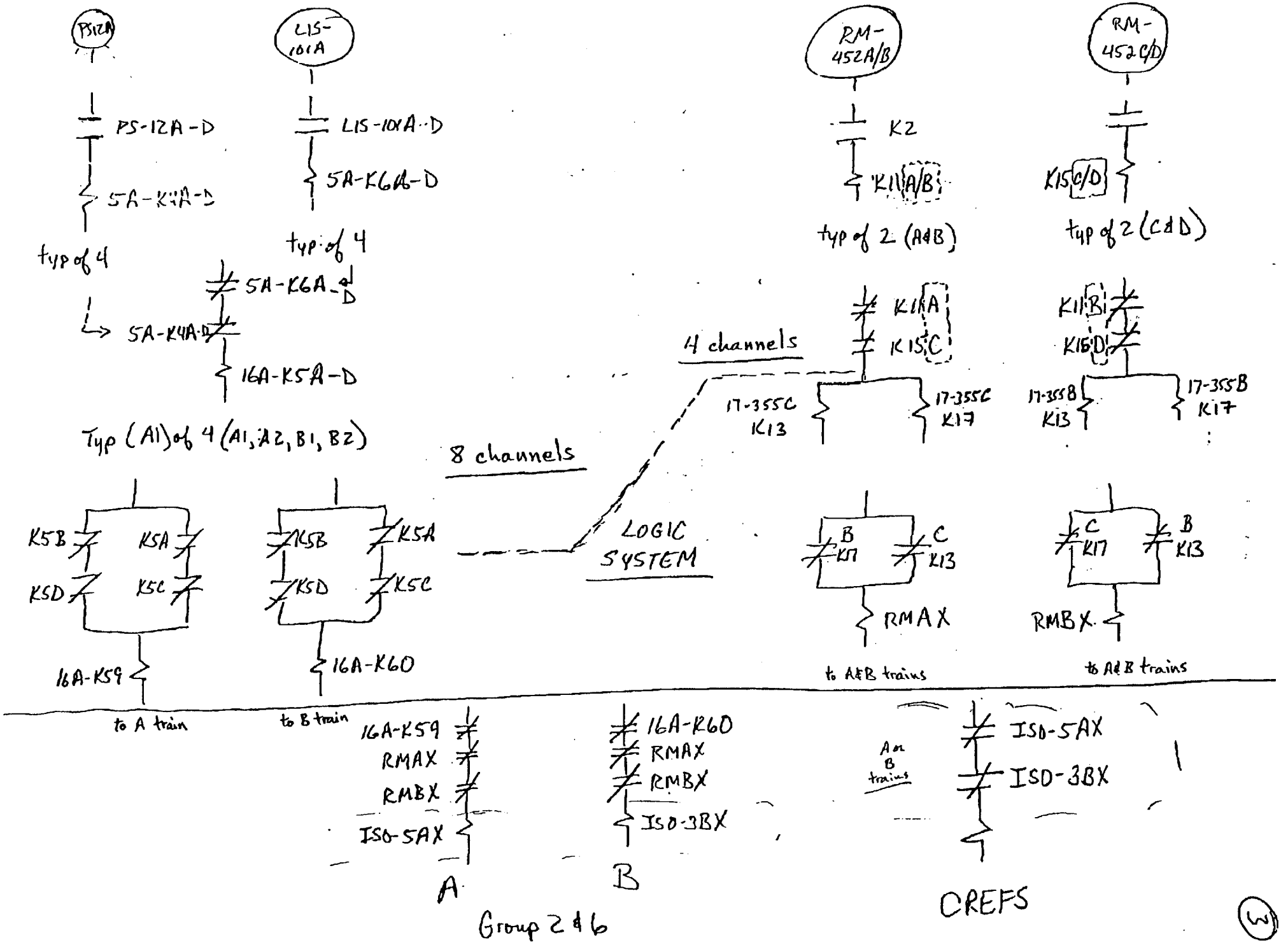
I. Number of Channels of Reactor Building Ventilation Exhaust Plenum Radiation-High Function

The LAR indicates the control room emergency filtration (CREF) System instrumentation "design will consist of two trip systems. Each trip system includes the sensors, relays, and switches necessary to cause initiation of the CREF System. Each trip system receives input from four sensors from each of the functions listed above (each sensor sends a signal to both trip systems). The Reactor Vessel Water Level-Low, Level 3, Drywell Pressure-High, and Reactor Building Ventilation Exhaust Plenum Radiation-High functions are each arranged in a one-out-of-two taken twice logic for each trip system."

The instrumentation used for these three functions is common for CREF System initiation, secondary containment isolation, primary containment isolation, and the reactor protection system. However, the instrumentation is not common to emergency core cooling system (ECCS) initiation.

The LAR further indicates that there are eight channels of the Reactor Vessel Water Level–Low (Level 3) function and eight channels of the Drywell Pressure–High function, but indicates there are four channels of Reactor Building Ventilation Exhaust Plenum Radiation–High function. The explanation for this difference in the number of channels lies in the specifics of the logic design. The logic design is basically the same for each of the three functions, i.e., the one-out-of-two taken twice logic for each trip system described above. However, the radiation instrumentation output is combined slightly differently such that the inputs to the trip systems are considered to be four channels.

For the level and pressure instrumentation, the output of the transmitter is fed into each trip system logic (i.e., a single relay opens a contact in each trip system), thus retaining its channel identity as two separate inputs to the logic system. This results in eight identifiable inputs to the logic (i.e, traceable back to the specific transmitter), and thus eight channels. For the radiation monitors, the output of the transmitter loses its channel identity earlier in that either of two (one out of two) separate transmitters will result in actuation of two relays in the trip system logic. Thus, further logic application (taken twice for each trip system) is within the logic system, not as a channel. Since there are only four identifiable inputs (i.e, traceable back to the specific transmitter) to the logic, there are four channels of the Reactor Building Ventilation Exhaust Plenum Radiation-High function.


PS12A
PS-12A-D
5A-K4A-D
typ of 4
LIS-101A
LIS-101A-D
5A-K6A-D
typ of 4
5A-K6A-D
5A-K4A-D
16A-K5A-D
Typ (A1) of 4 (A1, A2, B1, B2)
RM-452A/B
K2
K11 A/B
typ of 2 (A&B)
RM-452C/D
K15 C/D
typ of 2 (C&D)
K11 A
K15 C
K11 B
K15 D
4 channels
17-355C K13
17-355C K17
17-355B K13
17-355B K17
8 channels
LOGIC SYSTEM
K5B
K5A
K5D
K5C
16A-K59
to A train
K5B
K5A
K5D
K5C
16A-K60
to B train
B K17
C K13
RMAX
to A&B trains
C K17
B K13
RMBX
to A&B trains
16A-K59
RMAX
RMBX
ISO-5AX
A
16A-K60
RMAX
RMBX
ISO-3BX
B
Group 2 & 6
A or B trains
ISO-5AX
ISO-3BX
CREFS
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(4)

II. Applicability of Topical Report GENE-0770-6-1P-A

The following information is provided in the LAR:

> The 12 and 24 hour Completion Times are consistent with the times allowed by TS 3.3.6.2 for the same channels. Furthermore, the CREF initiation logic design is similar to previously analyzed systems for which the associated unavailabilities were found acceptable in the NRC Safety Evaluation Report (SER) (dated July 21, 1992) for General Electric (GE) Topical GENE-770-06-01, "Basis for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications." Therefore, this generic analysis is applicable for justifying these Completion Times.
>
> AND
>
> There are no changes proposed to the Surveillance Requirements or their associated Frequencies. The proposed application of those Surveillance Requirements for each new function conforms to their corresponding applications for these same functions in TS 3.3.6.2. The CNS CREF System initiation channels and Allowable Values are the same functions currently addressed in TS 3.3.6.2 for secondary containment isolation. Additionally, the associated channels are to be arranged in similar logic trip systems as that already addressed in TS 3.3.6.2 for secondary containment isolation. Furthermore, the setpoint methodology properly accounts for instrument drift and the associated channel functional test and channel calibration frequencies of 92 days and 18 months, respectively. The CREF System initiation logic design is similar to previously analyzed systems for which the associated surveillance frequencies were found acceptable in the NRC SER (dated July 21, 1992) for GE Topical GENE-770-06-01. Therefore, this generic analysis is applicable for justifying these surveillance frequencies. Furthermore, because the CNS logic design is similar to the previously analyzed systems, the 6 hour allowance for performing surveillance testing that is also approved within this generic review, is also applicable to the CNS design.

The NRC SER dated July 21, 1992, accepted GE Topical GENE-770-06-01, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications." As indicated in the cover letter for the SER, the NRC indicated that GENE-770-06-01 provides an acceptable basis for extending surveillance test intervals (STIs) and allowed out-of-service times (AOTs) for the BWR/4 Main Control Room Environmental Control System (MCRECS) and other systems. The SER further notes that applicants for proposed TS changes for individual plants must:

1. Confirm the applicability of the generic analyses of GENE-770-06-01 to the plant, and

2. Confirm that any increase in instrument drift due to the extended STIs is properly accounted for in the setpoint calculation methodology.

Section IV, Evaluation of Bases for STI and AOT Changes, of the SER associated Technical Evaluation Report (TER) specifically addresses the BWR/4 Main Control Room Environmental Control System (MCRECS) Actuation Instrumentation. This discussion addresses the acceptability of quarterly channel functional testing, a 24 hour AOT for repair, and a 6 hour AOT for surveillance testing. In the TER, the acceptance was based on showing each subsystem to be similar to a

previously analyzed system, thus the change in unavailabilities previously calculated for that system would also apply to the MCRECS. For MCRECS, the logic was shown to be similar to previously reviewed actuation instrumentation. The previous review was based, in part, on NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," dated July 1990.

With regard to item 1) above, the Nebraska Public Power District (NPPD) confirms the applicability of the generic analyses of GENE-770-06-01 to CNS for the CREF System. As indicated in the Bases for CNS TS 3.3.6.2, the AOTs and STIs for the Secondary Containment Isolation Instrumentation were based on NEDC-31677P-A and NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," dated March 1989.

The AOT and STI extensions for CNS TS 3.3.6.2 were approved in Cooper Nuclear Station (CNS) operating license amendment No. 178 which converted the TS to the improved NUREG-1433 format. The submittal for the conversion included the following three discussions of the changes for the Secondary Containment Isolation Instrumentation.

LB.1 The Channel Functional Test Frequency in CTS Table 4.1.1 for the Drywell Pressure High Function and in CTS Table 4.2.A, including Note 1, for the Reactor Low Water Level Function has been changed from once per month to once per 92 days, consistent with that provided in NEDC-30851P-A, Supplement 2, March 1989, and NUREG-1433. The NRC Safety Evaluation Report accepting this generic reliability analysis (dated January 6, 1989) requires each applicant for proposed Technical Specification changes to: 1) confirm the applicability of the generic analyses for NEDC-30851P-A, Supplement 2 to its plant; and 2) demonstrate by use of current drift information provided by the equipment vendor or plant-specific data, that the drift characteristics for instrumentation used in the Secondary Containment Isolation Instrumentation channels in the plant are bounded by the assumptions used in NEDC-30851P-A, Supplement 2 when the functional test interval is extended from monthly to quarterly. CNS has confirmed that the logic design of the instrumentation is bounded by that analyzed in the reliability analysis and the conclusions of the analysis are applicable to the CNS design. In addition, CNS has confirmed that the instrument drift due to the extended Surveillance Frequency is already properly accounted for in the setpoint calculation methodology.

LB.2 The time allowed for placing a channel in trip when one or more channels are inoperable and untripped (CTS Table 3.2.A, Note 2 and CTS Table 3.2.0, Note 1.8) is extended to 12 hours for those channels common to RPS and 24 hours for all other channels. While no finite completion time is currently provided, CNS administrative practices perform this action in a time shorter than the proposed 12 hour and 24 hour limit. These times have been shown to maintain an acceptable level of risk in accordance with previously conducted reliability analyses (NEDC 30851-P- A, Supplement 2, March 1989 and NEDC-31677-P-A, July 1990). As required by the NRC Safety Evaluation Reports accepting these generic reliability analyses (dated January 6, 1989 and June 18, 1990), CNS has confirmed that the logic design of the instrumentation is bounded by that analyzed in the reliability analyses and the conclusions of the analyses are applicable to the CNS design.

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LB.3 A time of 6 hours to perform Surveillances without requiring the associated Conditions and Required Actions to be taken, provided the associated Function maintains isolation capability, is proposed to be provided as a Note to the Surveillance Requirements. This short time has been shown to provide an acceptable assurance of Operability in accordance with previously conducted reliability analyses (NEDC 30851-P-A, Supplement 2, March 1989 and NEDC-31677P-A, July 1990). As required by the NRC Safety Evaluation Reports accepting these generic reliability analyses (date January 6, 1989 and June 18, 1990), CNS has confirmed that the logic design of the instrumentation is bounded by that analyzed in the reliability analyses and the conclusions of the analyses are applicable to the CNS design.

Similar LB discussions were provided in the NUREG-1433 conversion submittal for the TS 3.3.1.1 and TS 3.3.6.1 AOTs and STIs. These discussions were found acceptable for Amendment No. 178 as indicated in the associated SER Table L for Section 3.3, "Instrumentation."

The CNS instrumentation to be used for the CREF System initiation is the same instrumentation currently used for Secondary Containment Isolation Instrumentation as shown in CNS TS 3.3.6.2. The CNS instrumentation to be used for the CREF System initiation is also the same instrumentation currently used for Primary Containment Isolation Instrumentation and Reactor Protection System (RPS) actuation as shown in CNS TS 3.3.6.1 and TS 3.3.1.1 (with the exception that high radiation is not used for RPS actuation). Therefore, the system is confirmed to be similar to a previously analyzed and approved system.

With regard to item 2) above, NPPD confirms the instrument drift due to the STIs is properly accounted for in the setpoint calculation methodology. As indicated in the Bases for CNS TS 3.3.6.2, the setpoint calculations are performed using methodology described in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology" dated September 1996.

The Allowable Values for CNS TS 3.3.6.2 functions were approved in CNS operating license amendment no. 178 which converted the TS to the improved NUREG-1433 format. The submittal for the conversion included the following discussion of the Allowable Value changes for the Secondary Containment Isolation Instrumentation.

L.A.V This change revises the Technical Specification setting limits in CTS Tables 3.2.A and 3.2.D for proposed Section 3.3 instrumentation to reflect Allowable Values consistent with the philosophy of NUREG-1433. These Allowable Values (to be included in Technical Specifications) and the Trip Setpoints (to be included in plant procedures) have been established consistent with the General Electric (GE) Instrument Setpoint Methodology; the CNS specific safety analysis limits and the uncertainties associated with the CNS instrumentation. The setpoint evaluation used actual CNS physical data and operating practices to ensure the validity of the resulting Allowable Values and Trip Setpoints. All changes to safety analysis limits, applied in the methodology, were evaluated and confirmed as ensuring safety analysis licensing acceptance limits are maintained. All design limits, applied in the methodology were confirmed as ensuring that applicable design requirements of the associated systems are maintained. The methodology used to derive the Allowable Values and Trip Setpoints are based on combining the uncertainties of the associated channels.. In the methodology, the Trip Setpoints take into consideration drift uncertainties which were specifically determined in the CNS setpoint calculations. Plant calibration procedures will ensure the

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assumptions regarding calibration accuracy are maintained. The proposed Allowable Values have been established from each design or safety analysis limit by accounting for instrument accuracy and calibration uncertainties, as well as process measurement accuracy and primary element accuracy using the GE Instrument Setpoint Methodology. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions.

The CNS instrumentation to be used for the CREF System initiation is the same instrumentation currently used for Secondary Containment Isolation Instrumentation, Primary Containment Isolation Instrumentation, and Reactor Protection System (RPS) actuation as indicated above. Therefore, the setpoints are the same and the methodology used to determine the setpoints is as indicated in the Bases for CNS TS 3.3.6.2, TS 3.3.6.1, TS 3.3.1.1, and in the LAR provided Bases for CNS proposed TS 3.3.7.1. Similar LAV discussions were provided in the NUREG-1433 conversion submittal for the TS 3.3.1.1 and TS 3.3.6.1 Allowable Values.

Thus, GENE-770-06-01 provides an acceptable basis for the proposed STIs and AOTs for the CREF System.

Cross reference information:

Function	Level 3	Drywell Pressure	Radiation
Instruments	NBI-LIS-101A-D	PC-PS-12A-D	RMP-RM-452A-D
TS Allowable Value	≥3 inches	≤1.84 psig	≤19 mr/hr
TS Table 3.3.7.1-1*	Function 1	Function 2	Function 3
TS Table 3.3.6.2-1*	Function 1	Function 2	Function 3
TS Table 3.3.6.1-1*	Functions 2a, 5d, 6b	Function 2b	Function 2c
TS Table 3.3.1.1-1*	Function 4	Function 6	NA - not used
TS Table 3.3.5.1-1*	NA - **	NA - ***	NA - not used

* TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation
* TS 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation
* TS 3.3.6.1, Primary Containment Isolation Instrumentation
* TS 3.3.6.2, Secondary Containment Isolation Instrumentation
* TS 3.3.7.1, Control Room Emergency Filter (CREF) System Instrumentation
** ECCS Function 4c uses NBI-LIS-83A & ECCS Function 5c uses NBI-LIS-83B.
*** ECCS Functions 1b, 2b & 3b use PC-PS-101A-D.

(8)

3. Comparison to 50.36(c)(2)(ii) Screening Criteria for Removal:

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The Control Room Air Inlet Radiation - High function is not used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Other instrumentation is provided which provides adequate detection, and indication in the control room, of significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (see TS 3.4.5, Reactor Coolant System Leakage Detection Instrumentation).

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The Control Room Air Inlet Radiation - High function is not a process variable, design feature, or operating restriction that represents an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. High radiation in the control room air inlet is not assumed as an initial condition of any design basis accident or transient analysis.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The Control Room Air Inlet Radiation - High function is not a structure, system, or component that is part of the primary success path which functions or actuates to mitigate a DBA or transient since this function is no longer credited for protection of the control room personnel. Control room personnel protection is provided by actuation of the CREF System on Reactor Vessel Water Level–Low (Level 3), Drywell Pressure–High, and Reactor Building Ventilation Exhaust Plenum Radiation–High.

Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The Control Room Air Inlet Radiation - High function is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Recent information, i.e., operating experience, indicates that automatic control room isolation on high radiation in the control room air intake is not significant with more realistic source terms. Additionally, the proposed change does not result in a degraded control room environment following an accident or transient and therefore does not effect any risk significant prevention or mitigation activity in the control room. Finally, since there is no direct impact on the automatic operation or recovery of front-line core or containment cooling systems, the removal of control room isolation at high radiation levels has no impact on probabilistic risk significance.

Conclusion: Since the screening criteria have not been satisfied, the Control Room Air Inlet Radiation - High function may be removed from the CNS TS.

NEBRASKA PUBLIC POWER DISTRICT

Cooper Nuclear Station

Nuclear Licensing and Safety Department
P.O. Box 98
Brownville, NE 68321-0098

FAX
(402) 825-5827

Verification
(402) 825-5117

TO: M. Thadani (X1476) FAX: 301-415-3061

COMPANY: NRC

FROM: Eddie Grant (402) 825-5815

DATE: 9-10-01 # OF PAGES (including cover) 10

Following items from Thursday's telecon re: Dose Calcs:

1. FHA calc page that identifies changes, including the damper closure time change from 11 sec. to 10 sec.
2. March 20, 2000 letter responses to Q17 & Q18.
3. Bases for B3.3.6.2 which references approved GE Topical for setpoint methodology.

Nebraska Public Power District

DESIGN CALCULATIONS SHEET

NEDC: 99-032 Preparer: Robert L. Beilke Reviewer: J.J. Drasler

Rev. No: 2 Date: 9/4/01 Date: 9/4/01

in its entirety. The body of the calculation is now performed by CNS under Procedure 3.4.7 rather than by a consultant. Revision 1 has been prepared as a Status 2 calculation for NRC review and will be as-built upon receipt of a License Amendment which incorporates this calculation.

Revision 2 of NEDC 99-032 is being made as a result of data input format errors identified in the vendor supplied calculation Scientech Calculation 17080-M-01 Revision 2 (NEDC 99-031, Rev 3). Scientech Calculation 17080-M-01 provides the Control Room receptor X/Q data for NEDC 99-032. The data format errors were corrected in Scientech Calculation17080-M-01 and Revision 3 of 17080-M-1 was issued. The X/Q data in the corrected calculation resulted in higher X/Q values for the Reactor Building exhaust point and lower X/Q values for the elevated release point. This revision of NEDC 99-032 corrects the reactor building exhaust Control Room X/Q data. The elevated release point data was not updated as it was already conservative and contributes only a small amount to the overall Control Room personnel dose. In order to remain within GDC 19 dose limitations with the higher X/Q data the following additional concurrent changes were required throughout NEDC 99-032: reduced radial peaking factor from 1.9 to 1.8 which essentially changes the source term for all runs in Section A.0, reduced CREFS initiation from 11 to 10 seconds, revised time steps in the AXIDENT runs to better match the CREFS 10 second initiation, updated average Reactor Building flows for a given time period to better match the 10 second CREFS initiation, deleted average Reactor Building flow data not used, updated data files for all 7 AXIDENT runs based on the changes discussed above, and reran all 7 AXIDENT runs with the updated data.

9/4/01 2

2.0 INTENDED USE OF ANALYSIS RESULTS

The results of this analysis are to be used to reevaluate the design basis of CNS by comparing the calculated doses for the control room operator, a person at the EAB, and a person at the LPZ to the regulatory dose limits.

3.0 TECHNICAL APPROACH

3.1. General Description

The radiological consequences of a FHA are based on the fuel failure due to the drop of a fuel assembly onto the core in conjunction with a conservative transport methodology based on Standard Review Plan (SRP) 15.7.4 (Reference 1). Doses are calculated for control room operators and individuals at the site boundary beginning 67 hours after fuel irradiation.

The radiological consequences for the control room operators are assessed using the SCIENTECH-NUS computer code AXIDENT. This code calculates individual whole body (beta and gamma) and thyroid doses resulting from any postulated accident which releases radioactivity within the containment. AXIDENT models the transport of radioactivity to the environment and to the control room. This code includes the time dependent effects of containment sprays, recirculation, purge and intake filters, atmospheric dispersion and natural decay. The AXIDENT code is discussed in Reference 2. The principal application of the AXIDENT code is to determine the control room, LPZ and EAB dose due to a loss of coolant accident (LOCA).

The original AXIDENT code used very conservative dose conversion factors (DCFs) that were in effect and used for the design basis 10 CFR 100 type reactor siting analyses (i.e., TID 14844,

2

Nebraska Public Power District
Nebraska's Energy Leader

NLS2000029
March 20, 2000

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

Gentlemen:

Subject: Design Basis Accident Radiological Assessment Calculational Methodology - Response to Request for Additional Information
Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

References:
1. Letter to Mr. J. H. Swailes (Nebraska Public Power District) from Lawrence J. Burkhart [signed by Robert A. Gramm] (U.S. Nuclear Regulatory Commission) dated March 6, 2000, Cooper Nuclear Station - Request for Additional Information (TAC No. MA7758).

2. Letter to U.S. Nuclear Regulatory Commission (NLS990122) from John H. Swailes (Nebraska Public Power District) dated December 22, 1999, Design Basis Accident Radiological Assessment Calculational Methodology Revision.

By letter dated March 6, 2000 (Reference 1), the Nuclear Regulatory Commission (NRC) requested the Nebraska Public Power District (District) to provide additional information on the Design Basis Accident Radiological Assessment Calculational Methodology Revision submitted by the District on December 22, 1999 (Reference 2). Attachment 1 provides the additional information requested.

Reference 2 included six calculations. Based on the information provided in Attachment 1, two of the calculations are not impacted [Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Meteorological Dispersion-Accident Analysis (Nuclear Engineering Design Calculation (NEDC) 99-036) and Dose Calculation for Control Room, EAB, and LPZ for a Main Steam Line Break (NEDC 99-035)]. It was agreed in discussions held with the NRC staff during the week of March 13, 2000, that three calculations will be revised and provided under separate letter by March 24, 2000 [Control Room, EAB and LPZ Doses Following a Control Rod Drop Accident (NEDC 99-034), Control Room, EAB and LPZ Doses Following a Loss of Coolant Accident (NEDC 99-033), and X/Q Values for Control Room Intake Using ARCON96 (NEDC 99-031)]. The status of the remaining calculation [Control Room Habitability and Offsite Dose for a Fuel Handling Accident (NEDC 99-032)] will be also be addressed in the March 24th letter.

Cooper Nuclear Station
P.O. Box 98 / Brownville, NE 68321-0098
Telephone: (402) 825-3811 / Fax: (402) 825-5211
http://www.nppd.com

Reference 1, Question 6 requests justification for crediting iodine removal in the main turbine condenser. While the District believes that crediting iodine removal in the existing main turbine condenser design is already a part of the CNS licensing basis for radiological assessment calculation accident mitigation, the District also believes that it is appropriate to evaluate the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser and the main turbine condenser to confirm that these components will remain structurally intact (e.g., will not suffer gross structural failure) following a Safe Shutdown Earthquake.

The District will submit a letter, by March 24, 2000, describing the structural robustness of the existing main turbine condenser and the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser. The District will also address the low probability of needing the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser, and the main turbine condenser, for accident mitigation. In addition, this letter will provide a proposed license condition addressing when additional information will be provided to the NRC regarding the ability of the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser, and the main turbine condenser, to remain functional during and after a Safe Shutdown Earthquake.

Should you have any questions concerning this matter, please contact Sharon Mahler at (402) 825-5236.

Sincerely,

John H. Swalley
Vice President of Nuclear Energy

/rlb

Attachment

cc: Regional Administrator w/attachment
USNRC - Region IV

Senior Project Manager w/attachment
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachment
USNRC

NPG Distribution w/o attachment

Meteorological instruments are calibrated and maintained per approved procedures. Meteorological data is reviewed, validated, and summarized for analysis each year to support submittal of the CNS Annual Operating Report- Radioactive Effluents to the NRC. Self assessments and/or Quality Assurance Program audits are periodically conducted to ensure the meteorological program is conducted per procedures and regulatory commitments. Such assessments and/or audits provide reasonable assurance that the quality of the meteorological data adequately meets the guidelines of Regulatory Guide 1.23 and regulatory commitments. Lastly, NRC inspections are conducted to ensure meteorological program compliance with NRC rules and regulations, and with the conditions of the CNS Operating License. During the most recent inspection conducted in September 1999 no findings were noted in the meteorological program. The inspection included reviews of calibration procedures and calibration records for meteorological monitoring instrumentation and meteorological instrument operability, reliability, and annual meteorological data recovery.

Question 17. Was delta-T data recovery during 1995 and 1996 below 90%? Throughout the 5-year period, were there occurrences of very unstable conditions, as defined by the delta-T measurements, during night time hours? If so, to what is this attributed?

District Response:

As reported in the Cooper Nuclear Station Annual Operating Report – Radioactive Effluents (letters dated April 29, 1996 and April 19, 1997), the delta-T data recovery for 1995 and 1996 is given below:

	1995	1996
100m - 10m Delta T	83.4%	84.8%
100m - 60m Delta T	95.0%	95.2%
60m - 10m Delta T	84.2%	84.2%

There were occurrences of very unstable conditions, as defined by delta-T, recorded occasionally during nighttime hours. These occurrences were attributed to weather conditions such as wind shifts and minor temperature fluctuations.

CNS contracted a vendor to conduct a meteorological assessment in April 1999. The following excerpts are taken from the vendor report:

- [Vendor] has reviewed the stability data contained in Section 3.0 of the Updated Safety Analysis Report (USAR). The stability data are presented for two twelve-month periods; March 1970 through February 1971, and March 1971 through February 1972. Stability classifications are based on the 100- meter wind speed

and the direction and are divided into four classifications, which include a combination of Pasquill-Gifford standard A-G classes; Unstable (A, B, and C combined), Neutral (D), Moderately Unstable (E), and Very Stable (F and G combined).

- Joint frequency distributions of wind speed and direction versus stability class were compared between the USAR data and the 1998-1999 data period. Primary and secondary wind peaks of wind direction by stability class for the 1998-1999 period matched up well with the USAR data.

- [Vendor] compared stability class distributions for the previous 14 years of CNS meteorological data to determine if any differences in the data could be contributed to calibration of the data or were likely year to year climatological fluctuations.

- The 100-meter wind and associated delta-t distributions for the 14 data years [1984-1998] were very consistent with only minor year-to-year differences. The largest differences occurred in 1995 where a large occurrence of A (unstable) stability was offset by a corresponding decrease in D (neutral) stability. A similar pattern to a lesser degree occurred in 1996 and is common in warm and dry years. Based on this review, there is no indication from the stability distributions that the delta-t data were biased by the calibration procedures for the 100- or 10-meter temperature.

- As with the upper level wind and delta-t, the 10-meter wind and associated delta-t distributions for the 14 data years were very consistent with only minor year-to-year differences.

- A final comparison was made of the hourly stability classes and the joint frequency distributions using the 1997 and 1998 CNS meteorological data. The hourly stability classes were divided into three groups; day, night, and transition. The transition period is a the period one hour before and after sunrise/sunset where delta-t's and corresponding stabilities are changing rapidly. Although seasonally dependent, the typical hourly pattern expected would be stable (F and G) near sunrise moving rapidly to unstable (B and A) by late morning. The afternoon is typically unstable but begins to move toward neutral (D) near sunset, and finally back to stable (F and G) by midnight. Rainy, cloudy and windy days are characterized by neutral (D) to somewhat stable (E) conditions throughout the day or night. With some minor exceptions, this pattern was observed in the 1997 and 1998 data. Occasional occurrences of unstable conditions at night were evident in the summer months of both years but were associated with wind shifts and minor temperature fluctuations common during short summer nights.

-- [Vendor] performed a thorough review of the CNS meteorological data, stability class distributions, and USAR data to determine if the CNS temperature calibration practices have had an affect on the meteorological data quality. Based on this review, there appears to be no effect on the long term meteorological data quality. Further, the delta-t data and associated stabilities are representative of the CNS site based on a comparison with the USAR data and the additional 14-year data record.

Question 18. With respect to control room X/Qs, what is the basis for assuming a diffuse release from the turbine building? From where would releases be most likely to occur (vents, doors, and other potential openings to the environment)?

District Response:

During Turbine Building Ventilation System operation, the Turbine Building exhaust is directed to a common exhaust plenum (located east of the Turbine Building) by the Turbine Building exhaust fans (which would be de-energized on a loss of offsite power). The discharge of the plenum is approximately elevation 938 feet and is approximately 290 feet (88 meters) horizontally from the Control Room intake.

Following a postulated accident resulting in a radioactive release to the Turbine Building, in which there is no loss of offsite power (no LOOP), the Turbine Building exhaust fans continue to run, and the Turbine Building exhaust flow is directed vertically upward. In order for the release to reach the Control Room intake, the flow would have to rise above the turbine building roof (approximately elevation 1007 feet) and come down to the Control Room intake (approximately elevation 957 feet). Due to the vertical velocity, elevations, intervening building, and large horizontal distance to the Control Room intake, it is judged that this case would not yield bounding results for Control Room dose when compared to a postulated accident resulting in a radioactive release concurrent with a LOOP.

For the LOOP case, the Turbine Building exhaust fans would coast down and come to a stop, leaving no forced mode of ventilation to direct the Turbine Building atmosphere to the Turbine Building Ventilation system exhaust point. Leakage to the environment could be from any number of possible locations including the opening and closing of various doors, openings around door seals, duct penetrations, conduit penetrations, piping penetrations, and the turbine building siding itself. Leakage would therefore be expected to come from many locations along the perimeter of the turbine building external walls, not from any single location, as in the case of forced ventilation directed to a specific point. All faces have many potential leakage paths and each wall could be likely candidates for leakage.

Calculation NEDC 99-031 developed Control Room X/Q for turbine building release by conservatively selecting the wall closest to the Control Room intake as the leaky wall.

The calculation uses the entire surface area of this wall as the area source, in accordance with Section 2.5.7 of NUREG/CR-6331, Rev. 1. The NUREG gives the example for using the Diffuse Source Release model for the case where there are releases from many openings on a face of a building. Since this model was felt to most accurately describe the as-built layout of the plant and leakage openings, it was the one used in the calculation to model turbine building releases.

Based on followup discussions with the NRC Staff during the week of March 13, 2000, the initial horizontal and vertical diffusion coefficients used in determining the Turbine Building diffuse release X/Qs will be recalculated by dividing the assumed release area width, and the assumed release area height, by 6.

Question 19. With respect to the main steam line break assessment, **(a)** what is the basis for assuming a puff release to the environment? **(b)** Provide further details in the comparison between assuming a uniform and Gaussian distribution within the puff resulting in essentially the same integrated X/Q. **(c)** What assurance is there that the effluents will all pass relatively quickly as a puff?

District Response: *<"a, b, c" references added to the Question>*

(a) The main steam line break scenario is simulated as a puff release because it is a short-duration event for which the assumptions and theory of a continuous plume are not valid; i.e., the duration of the release for the main steam line break (~10 seconds) is shorter than the transport time between the source and receptor (~37 seconds). Standard Review Plan Section 15.6.4, paragraph II.3 and Standard Review Plan Section 2.3.4, paragraph III.1. provide additional details. Specifically, Standard Review Plan 2.3.4, paragraph III.1 states "Most accidental releases can be considered as continuous releases (i.e., on the order of several minutes or more). However, some releases such as from steam line breaks or of hazardous chemicals may be considered as instantaneous (puffs)." Additionally Section XIV, paragraph 6.5.8.3, of the Cooper Nuclear Station Updated Safety Analysis Report states "Since all of the activity is released to the environs in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated."

(b) Conservatively assuming that the entire release plume passes the Control Room intake prior to isolation of the intake (isolation occurs in 60 seconds), the integral of X/Q with respect to time (i.e., the dose) would be the same for the Gaussian distribution as for the uniform distribution.

(c) The assurance of relatively quick passage of the puff can be obtained from the comparatively small separation distance between the point of origin and the receptor (37 meters) and the wind climatology at that location. In the analysis presented (NEDC 99-035), 1 m/sec wind speeds were used as conservative low wind speeds

3

BASES

BACKGROUND (continued)

isolates the secondary containment and provides for the necessary filtration of fission products.

The exception to this arrangement is the Reactor Building Ventilation Exhaust Plenum Radiation-High Function, in which actuation of either trip system will close both valves on each penetration and start both SGT subsystems.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves and start the SGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The setpoint calculations are performed using methodology described in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal setpoint, but within its Allowable Value, is acceptable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis or appropriate documents. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when SCIVs and the SGT System are required.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

I. Reactor Vessel Water Level - Low (Level 3)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The Reactor Vessel Water Level—Low (Level 3) Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation systems on Reactor Vessel Water Level—Low (Level 3) support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Reactor Vessel Water Level—Low (Level 3) signals are initiated from level switches that sense the difference between the pressure due to a constant column of water

(continued)

Request for Additional Information on
Plant Systems
Extended Power Uprate License Amendment Application
Arkansas Nuclear One, Unit 2

1. Verify that for the 7.5% power uprate, the 160,000 gallon minimum volume for the operable condensate storage tank (CST) specified in Technical Specification 3.7.1.3 is adequate to meet the design criteria discussed in Section 9.2.6.1 of the Safety Analysis Report (SAR). The design criteria in the SAR indicates that the operable CST needs to have sufficient water for one hour of hot standby operations and the cooldown to hot shutdown. The design cooldown rate to hot shutdown is 75 degrees per hour. Include the volume required to meet the design criteria in your response.

Second Request for Additional Information on
Reactor Systems
Extended Power Uprate License Amendment Application
Arkansas Nuclear One, Unit 2 (ANO-2)

1. The proposed changes to Technical Specification (TS) 3.5.4 will only specify the refueling water tank (RWT) volumes assumed in the accident analysis and move the required RWT indicated water level to plant procedures. This proposal will not provide sufficient information in the TSs for operators control and Nuclear Regulatory Commission (NRC) enforcement of this safety requirement. Please modify your proposed TSs to keep the indicated RWT water level of 91.7% to 100% in TS 3.5.4 as that in the current TSs.

2. Section 2.4.6.1 of the application evaluates the emergency feedwater system. Please describe the affect of power uprate on the condensate storage capacity required to meet the requirement of Branch Technical Position RSB 5-1, using safety grade equipment to achieve cold shutdown, and coping of a station blackout.

3. In Table 3-1, the differences between the minimum TS values and the minimum analytical values of the reactor coolant system (RCS) flow rate, core inlet temperature, and pressurizer pressure are very small. Please discuss the method used to determine the uncertainties of these parameters.

4. In Section 6.4.5, the licensee stated that Table 6-6 presents transient lifetime occurrences for test conditions. Leak testing is covered under Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Section XI permits leak tests in lieu of hydrostatic tests. Consequently, the hydrostatic tests are no longer required to be analyzed for fatigue requirements. The licensee also stated that since leak testing at nominal operating pressure is done in conjunction with normal plant operation, there is no requirement to analyze leak testing with respect to fatigue considerations, except for the special secondary side tests associated with the steam generator. There is no discussion of how the results and measurements of these tests will be acceptable for the proposed power uprate. Provide such a discussion with regard to the fatigue usage and leak considerations.

5. In Section 6, there are tables which list number of cycles for various plant transients during the life the plant for the purpose of mechanical design. Please compare these data with the current design basis associated with the original steam generators and discuss the reason of the changes.

6. Please expand Section 7.3 to address all changes of reactor protection system (RPS) trip delays, including the reasons of the changes. Confirm that the changes of RPS trip delay have been factored in all the re-analyses of affected events with acceptable consequences.

7. Please confirm that all computer codes (CENTS, HERMITE, etc.) used in the re-analyses have been reviewed and approved by NRC for their applicability at ANO-2. Provide a discussion that explains how all limitations have been satisfied.

8. Please address the following areas regarding the reactor coolant pump (RCP) shaft seizure accident described in Section 7.3.5:

   a) Explain why a concurrent loss of off-site power is not assumed with a RCP shaft seizure.

   b) Describe the method used to determine the amount of failed fuel and state the number of failed fuel in this event.

9. Provide the methods used in determining the allowable power level with inoperable main steam safety valves.

10. Please address the following areas regarding the feedwater line break accident analysis described in Section 7.3.11.2:

    a) Explain the need for the change in methodology for determining the most limiting break size. Provide discussion on why the feedwater line break analysis submitted by your letter dated November 29, 1999 (Enclosure 4, Page 40 of 172) is no longer valid.

    b) Explain why the proposed method would able to determine a most limiting break size which could bound the spectrum of potential break sizes including a double ended main feedwater line break.

    c) Is the proposed method of determining the most limiting feedwater line break size consistent with that used in the Combustion Engineering (CE) System 80+ design? Has the proposed method been applied in any other CE-designed pressurized water reactors? Provide the citation for staff approval of the revised methodology and its applicability to ANO-2.

    d) Discuss the instrument used in the RPS to initiate a reactor trip on low water level (with 40,000 lbs of water remaining) in the failed steam generator. Is this level measurement reliable during the dynamic transient conditions of a steam generator?

    e) Discuss the single failure assumed in the feedwater line break analysis.

11. Please expand Section 7.3.13 to discuss the following:

    a) The most limiting single failure assumed in the steam generator tube rupture (SGTR) analysis.

    b) Confirm the operator actions assumed in the SGTR analysis are consistent with emergency operating procedures at ANO-2.

    c) Describe operator actions relative to steam generator overfill during a SGTR event.

12. To show that the referenced generically approved loss-of-coolant accident (LOCA) analysis methodologies apply specifically to ANO-2, provide a statement that ANO-2

and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

13. The ANO-2 power uprate submittal references CENPD-137, Supplement 2-P-A, April 1998, as the generically approved small-break LOCA (SBLOCA) methodology as the one which will become the methodology to be included in licensing documentation and which was used to perform the ANO-2 SBLOCA licensing analyses for the uprated power. The NRC approved CENPD-137, Supplement 2-P-A invoking unique criteria for the specific methodology and the then-existing or then-proposed plant conditions. Show that this methodology continues to be applicable to ANO-2 at the uprated power.

14. Page 1 of the cover letter, last paragraph: Has the Westinghouse Topical Report WCAP-10263 been approved by the NRC? If not, please provide technical justification (quantitative and qualitative) for its selection.

15. Page 7-105 , Section 7.3.0.1, list of input parameters: Please provide technical justification for Items 2 and 3.

16. The plant parameter changes stated in the last paragraph of page 7-105 (and continuing on to page 7-106): The first four changes were not provided with any technical bases. Please provide technical justifications (quantitative and qualitative) for the selection of these parameters.

17. In the first sentence in the first paragraph on page 7-110, it is stated that the power uprate could result in a small degradation of the calculated thermal margin. How small is the degradation? How much margin is left?

18. On page 7-113, under the subheading of hot full power, Item 5 states that a moderator temperature coefficient (MTC) of $0.0 * 10^{-4}$ $\Delta\rho/°F$ is more conservative than a MTC of negative $0.2 * 10^{-4}$ $\Delta\rho/°F$ at beginning-of-cycle. Please explain.

19. Item 6 on page 7-113 states that the response time was increased to 0.40 seconds. Please justify.

20. Item 7 on page 7-113: Was this reactivity insertion rate changed from a prior value, and why?

21. On page 7-118, the first sentence states that the impact of the above changes result in no violation of the specified acceptable fuel design limits. Please explain. Also in the same paragraph, it states that acceptable limits were not violated. Please explain. What are these acceptable limits?

22. Item B in the first paragraph on page 7-121 states that credit was taken for the temperature difference between the modes. Please explain.

23. On page 7-121, why was the minimum response time changed from 30 minutes to 31 minutes?

Mail Envelope Properties (3B95211D.533 : 15 : 21310)

Subject: Cooper MB1419
Creation Date: 9/4/01 2:44PM
From: Mohan Thadani

Created By: MCT@nrc.gov

Recipients	Action	Date & Time
nppd.com	Transferred	09/04/01 02:45PM
dfkunse (Internet:dfkunse@nppd.com)		
elmccut (Internet:elmccut@nppd.com)		

Post Office	Delivered	Route
nppd.com		Internet

Files	Size	Date & Time
MESSAGE	1235	09/04/01 02:44PM

Options

Auto Delete: No
Expiration Date: None
Notify Recipients: Yes
Priority: Standard
Reply Requested: No
Return Notification: None

Concealed Subject: No
Security: Standard

To Be Delivered: Immediate
Status Tracking: Delivered & Opened

From: Mohan Thadani
To: Internet:dfkunse@nppd.com; Internet:elmccut@nppd.com
Date: 9/4/01 2:44PM
Subject: Cooper MB1419

Please include the following additional question in the list of other RAI for which you are preparing responses.

1. You state that the allowed value of < 49 mR/hr for the Reactor Building Ventilation Exhaust Plenum Radiation - High function was chosen to promptly detect gross failure of the fuel cladding. How was this value derived? With this value, can the assumed release from the postulated design basis FHA be detected and cause initiation of the CREF within the 90 seconds assumed in the FHA dose analysis?

Mail Envelope Properties (3B8E6C78.0A6 : 4 : 32934)

Subject: Steam Generator tube U-bend fatigue
Creation Date: 8/30/01 12:38PM
From: <jseawright@txu.com>

Created By: jseawright@txu.com

Recipients
nrc.gov
owf4_po.OWFN_DO
DHJ (David Jaffe)

Post Office	**Route**
owf4_po.OWFN_DO	nrc.gov

Files	**Size**	**Date & Time**
MESSAGE	678	08/30/01 12:38PM
Part.001	1542	
Mime.822	3621	

Options

Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

Docket No 50-445, 50-446

From: <jseawright@txu.com>
To: <dhj@nrc.gov>
Date: 8/30/01 12:40PM
Subject: Steam Generator tube U-bend fatigue

An evaluation of the potential for high cycle fatigue rupture of a steam generator tube, similar to that which occurred at North Anna Unit 1, has been performed for Comanche Peak Unit 1. Consistent with the requirements of NRC Bulletin 88-02, the anti-vibration bar configuration of the ruptured tube in North Anna, R9C51 S/G C, is used as the reference case for the tube fatigue usage calculations for Comanche Peak Unit 1. The acceptability of unsupported tubes in the steam generators is based on tube specific analysis relative to the North Anna Unit 1 R9C51 tube, including the relative flow peaking factors. This evaluation was documented in WCAP-15009, Revision 0,

"Comanche Peak Unit 1
Evaluation for Tube Vibration Induced Fatigue". The aforementioned topical report was submitted to the NRC via TXU Electric letter, logged TXX-99121, from C. L. Terry to the NRC dated July 21, 1999. Based upon the results of the fatigue analysis, CPSES Unit 1 steam generator tubes except for two tubes in steam generator 3 were shown by calculation not to have the potential to experience high cycle fatigue failure similar to that which occurred at North Anna Unit 1. Those two tubes, R10C109 and R11C109, had cable dampers and plugs installed during the sixth refueling outage (1RF06). As a result of installing these cable dampers and plugs, no additional action were required for these tubes. This completed the actions required by TXU Electric for the steam generators in CPSES Unit 1 as required by NRC Bulletin 88-02.

TXU Electric provides for existing design controls such that updates to stress ratios and fatigue usage calculations are performed in the event there are any significant changes to the SG operations parameters (e.g., steam pressure, flow, circulation ratios) relative to those assumed in the applicable WCAP-15009 analysis for the proposed uprate. Any additional preventive actions that are taken are reported to the NRC via the Reporting Criteria of CPSES Technical Specification Section 5.6.10 and/or NEI 97-06 Rev.1. The tube plugging report and the Condition Monitoring Report identifies the tubes repaired and the probable cause of the damage or anomaly.

Mail Envelope Properties (3B8D511A.288 : 19 : 25224)

Subject: Draft Info for Tomorrow's Discussion
Creation Date: 8/29/01 4:29PM
From: "Lanny Dusek" <Lanny_Dusek@pgn.com>

Created By: Lanny_Dusek@pgn.com

Recipients
nrc.gov
owf4_po.OWFN_DO
DJW1 (David Wrona)

pgn.com
TOM_MEEK CC (Tom Meek)
SPIKE_FORD CC (Spike Ford)
GINA_HUEY CC (Gina Huey)

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MESSAGE	566	08/29/01 04:29PM
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1515401 02 (X-Sect).pdf	803213	
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Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

From: "Lanny Dusek" <Lanny_Dusek@pgn.com>
To: <DJW1@nrc.gov>
Date: 8/29/01 4:31PM
Subject: Draft Info for Tomorrow's Discussion

Dear Dave,

Attached are three files (in draft) in pdf format for tomorrow's discussion about Phase II groundwater monitoring. Two of the files are in color and would be easiest to talk about if you can have them printed out in color (or viewed on screen) for reference on your end.

Also, please note that we have now seen in on site analysis that there is low level tritium contamination in one of the Phase I monitoring wells. We are in the process of evaluating the tritium and its operational and administrative effects.

Regards,

Lanny Dusek
(503)556-7409

CC: "Gina Huey" <GINA_HUEY@pgn.com>, "Spike Ford" <SPIKE_FORD@pgn.com>, "Tom Meek" <TOM_MEEK@pgn.com>

OUTLINE OF PROPOSED PHASE II WORK
GROUNDWATER MONITORING WORK PLAN
TROJAN NUCLEAR PLANT
RAINIER, OREGON

Phase I Activities and Findings

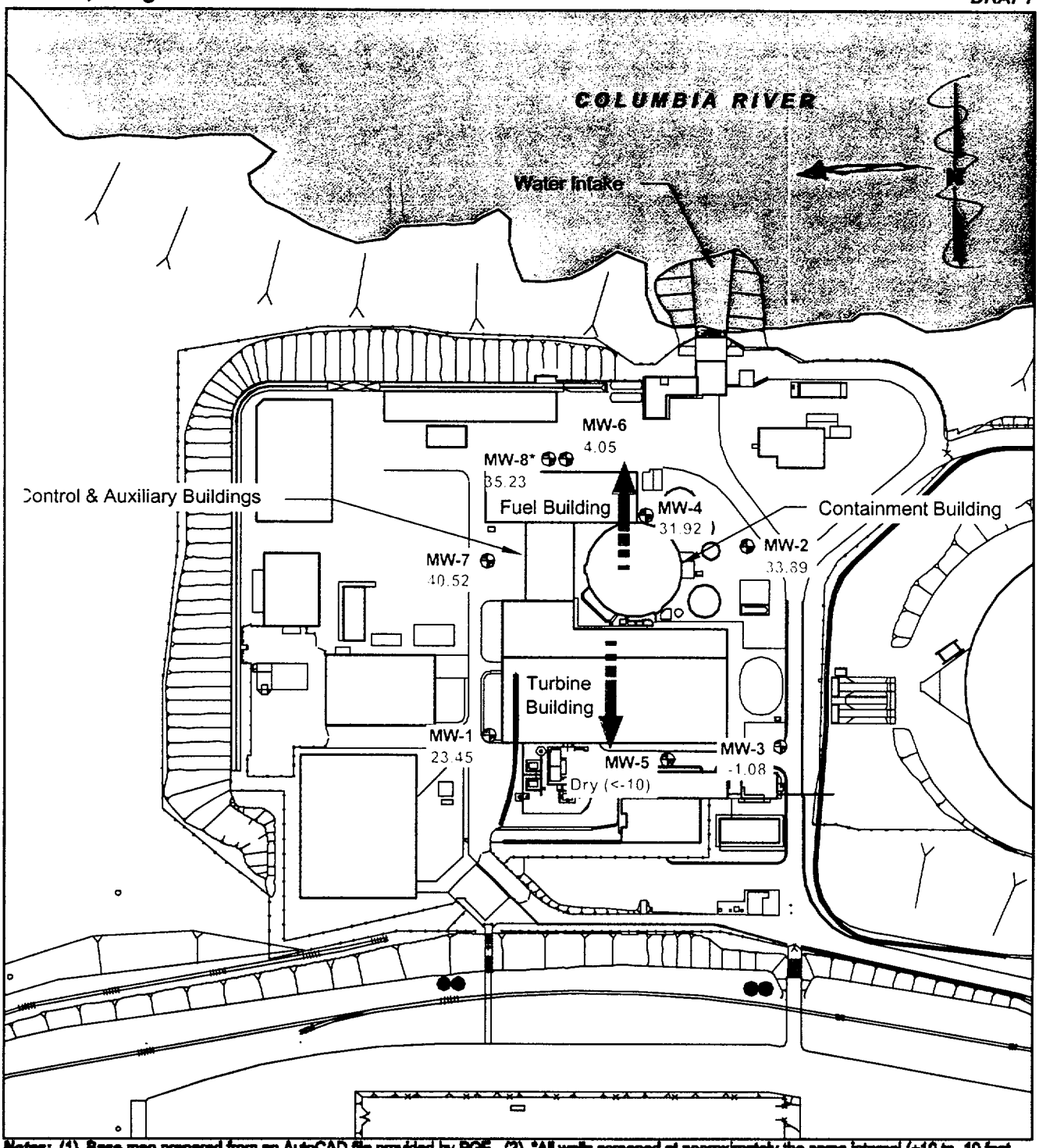
- *Monitoring Wells Installed:* Eight wells around former power plant and two background wells.
- *Hydrogeology:*
  - Relatively soft bedrock (easy drilling with air rotary).
  - Groundwater present in fractures in the bedrock (fracture flow).
  - Low groundwater yields: when wells are purged, groundwater present only in well casing and filter pack; full groundwater recovery takes up to a week.
  - Groundwater table highest in center of site, with split gradient both to east (toward the Columbia River) and west (toward buried valley).
- *Groundwater Sampling and Analysis:* Samples of water from each of the wells were analyzed for tritium and gamma emitting radionuclides at the Trojan onsite laboratory. No radionuclides above background levels were identified in seven of the eight wells. Tritium was identified in samples taken from monitoring well MW-8 located east of the Fuel Building (Figure 1). The concentration of tritium was approximately 2,500 pCi/l. No other target radionuclides were detected above background levels in samples taken from this well. Additional samples from each of the wells have been sent to the offsite contractor laboratory for complete analysis as described in the monitoring plan.

Proposed Nested Monitoring Wells in Buried Valley

- *Objective:* Assess groundwater in buried valley, per License Termination Plan.
- *Proposed Nested Monitoring Wells:* Two groups (nests) each consisting of two 2-inch-diameter wells as follows.
  - Shallow well: 25 feet deep, screen from 15 to 25 feet (target is top of groundwater table).
  - Deep well: 55 feet deep, screen from 40 to 55 feet (if bedrock is encountered, well will be installed with bottom on bedrock).
- *Well Installations:* Due to anticipated large pieces of basalt fill material, drill and install with air rotary methods. Each well will be installed in its own borehole. Well constructed and developed per the Groundwater Monitoring Work Plan.
- *Sampling and Analysis:* Per the Groundwater Monitoring Work Plan.

Site Exploration Plan and Proposed Nest Monitoring Well Locations
Trojan Nuclear Plant
Rainier, Oregon

DRAFT



JAB 1=1

F:\DATA\Jobs\15154 Bridgewater - Trojan\Proposed Wells 15154-011515401 01 (Prop Wells)

Notes: **(1) Base map prepared from an AutoCAD file provided by PGE. (2) *All wells screened at approximately the same internal (+10 to -10 feet MSL), except MW-8 which is screened at 25 to 35 feet MSL.**

Legend:

MW-1	Monitoring Well Location and Designation
23.35	Groundwater Elevation in Feet Above Mean Sea Level (MSL): August 9, 2001
	General Groundwater Flow Direction
	Proposed Nested Monitoring Well Location

0 200 400

Approximate Scale in Feet

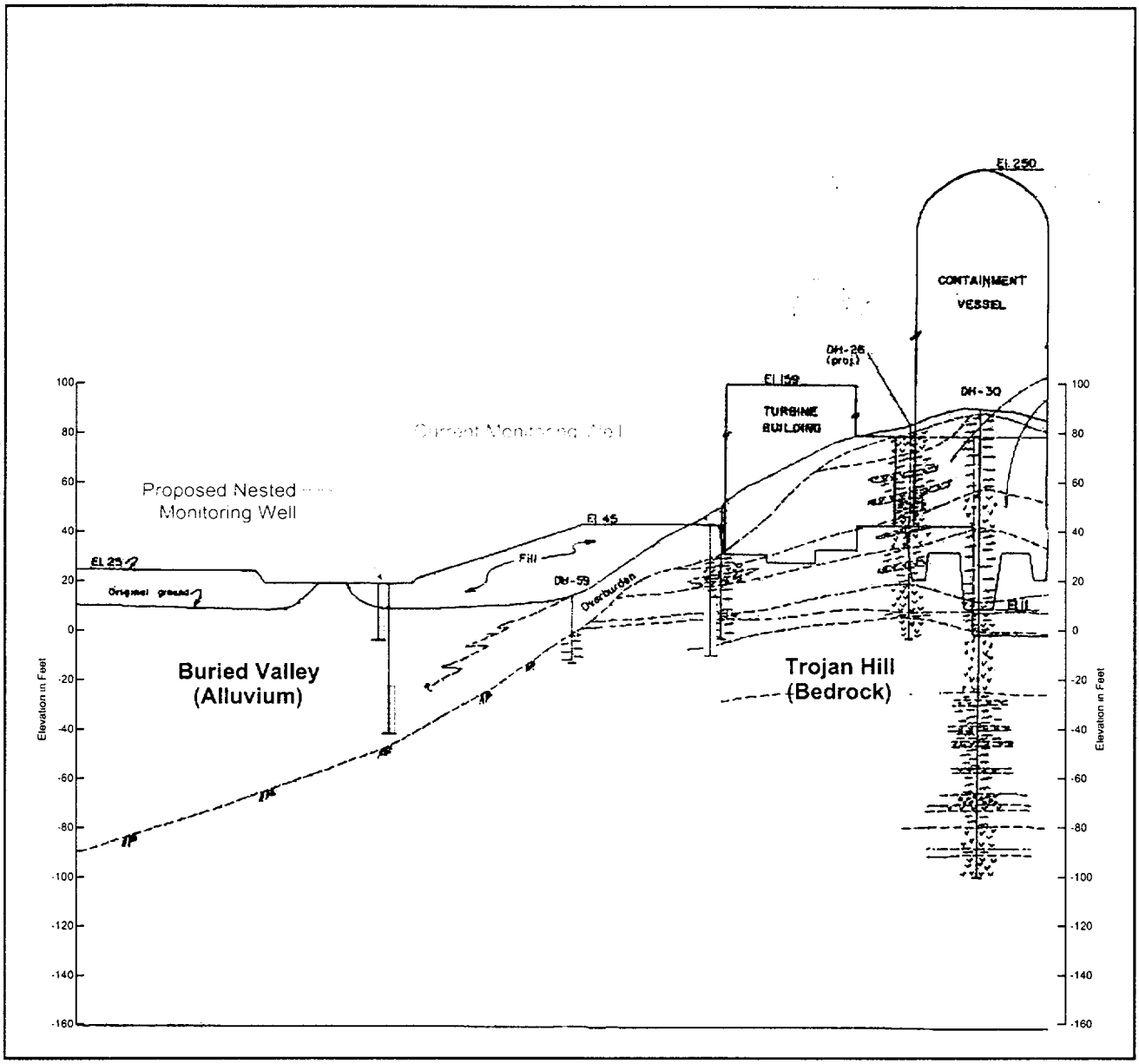
HARTCROWSER

15154-01 ***8/01***

Figure 1

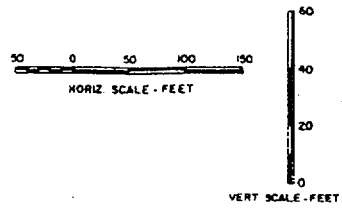
Proposed Nested Monitoring Well Cross Section
Trojan Nuclear Plant
Rainier, Oregon

DRAFT



F:\DATA\Jobs\15154 Bridgewater - Trojan\Proposed Wells 15154-01\15154-01 02 (X-Sect)

Note: Base cross section from Figure 2.5-5 of Final Safety Analysis Report for the Trojan Nuclear Plant.



Legend:

Monitoring Well

Screened Interval

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15154-01 *8/01*

Figure 2

From: Mohan Thadani
To: McCutchen, Edward L.
Date: 8/1/01 11:21AM
Subject: DRAFT RAI FOR FUEL HANDLING ACCIDENT

Ed:

In reviewing the methodology for fuel handling accidents, the staff has identified the following draft request for additional information. Please review the draft RAI and arrange a conference call with the NRC staff to discuss your responces. We will submit formal RAI if needed.

Thanks.

Mohan Thadani

<<<<<<<<<<<<<<<<<<<<<<<<<<<<>>>>>>>>>>>>>>>>>>>>>>>>>>>>>>>>>

Draft Request for Additional Information

Cooper Dose Calculation Methodology
for
Fuel Handling Accidents

1. Calculation NEDC 99-032, "Control Room Habitability and Offsite Dose for a Fuel Handling Accident," which was included in the February 28, 2001 submittal, takes credit for 67 hours of decay. By what means is the decay time controlled to be 67 hours or greater before moving fuel?

2. Does the reactor building achieve 0.25" w.g. negative pressure within 90 seconds after the onset of an FHA? Has this operation of the secondary containment under postulated FHA conditions been tested?

3. You proposed to take credit for a reduced Control Room Emergency Filtration system initiation time of 11 seconds. Has this initiation time been verified through testing?

4. In NEDC 99-032, Section 4.2, "Release Rate from the Refueling Area," you discuss the reactor building release rate as a function of time, considering factors such as the 90-second reactor building isolation damper closure period, fan coastdown, radiation monitor detection time and the effective hold-up time in ductwork. A summary was provided in this discussion of the calculations performed in another calculation not provided in the February 28, 2001 submittal. With regard to the calculations:

A. What was used in the calculations for the fan speed as a function of time during coastdown?

B. Was this information provided by the equipment manufacturer or from another means?

CC: Michelle Hart

Mail Envelope Properties (3B681E92.533 : 15 : 21310)

Subject: DRAFT RAI FOR FUEL HANDLING ACCIDENT
Creation Date: 8/1/01 11:21AM
From: Mohan Thadani

Created By: MCT@nrc.gov

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From: Mohan Thadani
To: McCutchen, Edward L.
Date: 7/18/01 9:13AM
Subject: CLARIFICATION RE: DOSE CALCULATION METHODOLOGY AMENDMENT

DRAFT QUESTION

Is the 1994 - 1998 meteorological data used in the ARCON96 calculations for the current fuel handling accident radiological assessment from electronic files dated June 5 and 8, 1999? We are trying to check the data files we have here (dated June 5 and 8, 1999) and want to be sure we are looking at the correct ones. If those are the correct ones, please check the data format? It appears that data that should be in a 1x, A5, 3x, I3, 2I, 2x, I3, I4, 1x, I2, 2x, I3, I4 format ("I" is the letter, not a numeral one) is missing the last 2x, so it may be that the last I3 and I4 fields could be misread since they would not be in the correct "columns." Further, in response to Question 17 of a March 20, 2000 submittal, some delta-T data recovery rates were given for a couple of the years. In looking over the data in the files that we have, it appears that more data are flagged as invalid than what was indicated in the Question 17 submittal. In addition, it appears that some of the upper level wind data were also flagged as invalid at the same time as the delta-T data. While it is certainly possible that both sets of data are invalid, we just want to be sure that they really are invalid. If what we have isn't right, we would like to discuss with you and get the right data to help us understand the inconsistencies.

Please send an email response. If necessary, we will send you formal RAI for this email after discussions.

CC:PUBLIC

CC: Leta Brown

Mail Envelope Properties (3B558B6A.533 : 15 : 21310)

Subject: CLARIFICATION RE: DOSE CALCULATION METHODOLOGY AMENDMENT
Creation Date: 7/18/01 9:13AM
From: Mohan Thadani

Created By: MCT@nrc.gov

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