ENCLOSURE 3

NEDC 07-082, Rev. 2

Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station

(Non-proprietary Version)

Title: Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station	Calculation Number: NEDC 07-082
System/Structure: MS, MSIVs, PC, SC, ECCS, SLC, Alternate Leakage Treatment, CREFS	CED/EE Number: EE 06-025
	Setpoint Change/Part Eval Number: N/A
Component: N/A	Discipline: DED Mechanical
Classification: [X] Essential; [ ] Non-Essential	SQAP Requirements Met? [ ] Yes; [X] N/A

Proprietary Information Included? [X] Yes; [ ] No

Description:

This calculation determines the radiological dose consequences for a postulated design basis Loss of Coolant Accident (LOCA) at Cooper Nuclear Station using an Alternative Source Term (AST).

This calculation determines the dose to a Control Room occupant and to a person at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) at the Cooper Nuclear Station site following a design basis Loss of Coolant Accident (LOCA). The analysis is performed using an Alternative Source Term (AST) in accordance with the guidance provided by the NRC in Regulatory Guide 1.183 (July 2000) and as authorized by 10CFR50.67. This calculation has been prepared as a Status 3 calculation for NRC review and will be taken to Status 1 after NRC approval of the LOCA AST methodology.

This calculation incorporates by attachment Alion Science & Technology Calculation No. ALION-CAL-NPPD-3236-002, Rev. 1, in accordance with CNS Engineering Procedure 3.4.7. The calculation also presents results from additional shine sources for Control Room occupants.

Conclusions and Recommendations:

The results are tabulated in Section 5 of this calculation for each of the three (3) receptor locations:

1. Control Room
2. Low Population Zone (LPZ)
3. Exclusion Area Boundary (EAB)

All calculated doses were found to be below the stipulated limits. It is therefore concluded that the regulatory dose limits will not be exceeded following a postulated design basis LOCA at Cooper Nuclear Station.

ATTACHMENTS:

1. Alion calculation ALION-CAL-NPPD-3236-002, Rev. 1 (including attachments thereto).

Rev. Number	Status	Prepared By/Date	Reviewed By/Date	IDVed By/Date	Approved By/Date
2	3	Alion Science & Technology 8/19/08	M J Bennett 10/1/08, J Drasler 10/1/08 M. J. Bennett/Jim Drasler	N/A	10-1-08 Stan Domikaitis
1	3	Alion Science & Technology 8/19/08	M. J. Bennett/Cory Kelsey 9/14/08	N/A	Stan Domikaitis 9/16/08
0	3	Alion Science & Technology 7/12/07	Jim Drasler/ Billy W. Reid 7/25/07	N/A	Stan Domikaitis 9/19/07

Status Codes

1. Active
2. Information Only
3. Pending
4. Superseded or Deleted
5. OD/OE Support Only
6. Maintenance Activity Support Only
7. PRA/PSA

Revision Summary

Revision 0 - Initial issue.

Revision 1 - Revision 1 revised the MSIV pathway leakage from 100 scfh/MS line & 200 scfh total MS pathway leakage to 150 scfh/MS line & 300 scfh total MS pathway leakage. In Sections 2.4.1 and 2.4.2, new Main Steam Line decontamination factors were calculated for particulate and elemental deposition, based on the increased MSIV leakage flowrates; however, credit for decontamination via these mechanisms was removed. Section 2.4.2 was revised to remove the development of the condenser elemental iodine removal. New Appendix A, "Condenser Effective Filter Efficiency Calculation," was developed to calculate condenser effective filter efficiency for iodine removal. The results of Appendix A have been included as input parameters for Drywell-MSIV Leakage Pathway as calculated in Appendices D and E based on the BWROG model documented in NEDC 35858P-A, Design Input 15. Only calculations for leakage through the MSIVs have been affected. The other two pathways, leakage from the drywell through the reactor building and release from the suppression pool through ESF components have remained unaffected. All input changes to this calculation are conservative.

Revision 2 - Revision 2 clarifies that the limiting single failure assumed in the AST LOCA dose calculation is the failure of a filter heater in one of the two Standby Gas Treatment System trains. The Alion calculation uses language implying consideration of an MSIV failure; no such failure was assumed or required to be assumed in the calculation. Revision 2 also clarifies that the MS Pathway also includes leakage through the inboard MS drain line (through penetration X-8); therefore, the total leakage (300 scfh at P_a) includes leakage through this flow path.

Nebraska Public Power District

DESIGN CALCULATION CROSS-REFERENCE INDEX

ITEM NO.	DESIGN INPUTS	REV. NO.	PENDING CHANGES TO DESIGN INPUTS
1	NOT USED		
2	Cooper Nuclear Station USAR, Chapter VI, XIV	loep. xxii5	UCR 2007-015, UCR 2007-016
3	JELCO Drawing 2841-2	N04	
4	B&R Drawing #2052	N31	DCN 06-0624
5	B&R Drawing 2019, Sh. 1	N41	DCN 05-1819
6	Drawing EC93877GA, Sh. 1	N02	DCN 06-0364
7	NEDC 94-034C	3	C/N 3C1 (EE 03-118)
8	Proc. 2.2.73	47	
9	NEDC 07-071	0	
10	Proc. 6.MISC.501	6	
11	Proc. 6.SC.201	25	
12	B&R Drawing #2001 SH 2	N07	
13	NEDC 07-056	0	
14	NEDC 05-045	1	
15	GE Topical Report NEDC-31858P-A	2	
16	NRC Reg Guide 1.183	0	
17	Drawing EC93877E3	B	
18	Drawing EC93877GA, Sh. 2	M	DCN 06-0365
19	Drawing EC93877E2-A	A	
20	Drawing M-81762	7	
21	Drawing DC93877EP	N01	
22	Drawing CC93877EP-A-13	N01	
23	Drawing CC93877EP-A-12	N01	
24	Drawing CC93877EP-A-11	N01	
25	Drawing CC93877EP-A-10	N01	
26	Drawing CC93877EP-A-9	N02	
27	Drawing DC93877SC-H	N02	
28	Drawing EC93877SC-1A	N01	DCN 06-0362
29	Drawing CNS-MS-43	N04	DCN 07-1606

Nebraska Public Power District

DESIGN CALCULATION CROSS-REFERENCE INDEX

ITEM NO.	AFFECTED DOCUMENTS*	REV. NUMBER
1	Inservice Testing Program	6
2	Appendix J (PCLRT)	9
3	CNS Procedure 6.MISC.501	6
4	USAR (and ESAR) Section XIV-6.3	loep.xxii5
5	USAR (and ESAR) Section III-9.2	loep.xxii5
6	CNS Technical Specifications, Section 3.1.7 & Bases	Amend. 178
7	CNS Technical Specifications, Section 3.6.1.3.10 & Bases	Amend. 220
8	DCD 10	04/18/08
9	DCD 19	10/26/04
10	DCD 31	04/18/08
11	DCD 39	11/08/06
12	ATLAS Database	N/A
13	NEDC 99-033	5
14	EQ Program	N/A
15	CNS Procedure 6.SC.201	25
	*Actual document changes will be via the implementation process per EE 06-025 and NRC review of AST methodology. Both of these processes could generate additional document changes or remove those listed here. Since the EE is the governing implementing document, it has the final authority. Therefore, any changes made by the implementing EE are not to be reflected by a revision to this calculation solely for that purpose.	

The purpose of this form is to assist the Preparer in screening new and revised design calculations to determine potential impacts to procedures and plant operations.®[1]

	SCREENING QUESTIONS	YES	NO	UNCERTAIN
1.	Does it involve the addition, deletion, or manipulation of a component or components which could impact a system lineup and/or checklist for valves, power supplies (breakers), process control switches, HVAC dampers, or instruments?	[ ]	[X]	[ ]
2.	Could it impact system operating parameters (e.g., temperatures, flow rates, pressures, voltage, or fluid chemistry)?	[ ]	[X]	[ ]
3.	Does it impact equipment operation or response such as valve closure time?	[X]	[ ]	[ ]
4.	Does it involve assumptions or necessitate changes to the sequencing of operational steps?	[ ]	[X]	[ ]
5.	Does it transfer an electrical load to a different circuit, or impact when electrical loads are added to or removed from the system during an event?	[ ]	[X]	[ ]
6.	Does it influence fuse, breaker, or relay coordination?	[ ]	[X]	[ ]
7.	Does it have the potential to affect the analyzed conditions of the environment for any part of the Reactor Building, Containment, or Control Room?	[X]	[ ]	[ ]
8.	Does it affect TS/TS Bases, USAR, or other Licensing Basis documents?	[X]	[ ]	[ ]
9.	Does it affect DCDs?	[X]	[ ]	[ ]
10.	Does it have the potential to affect procedures in any way not already mentioned (refer to review checklists in Procedure EDP-06)? If so, identify: Affected Documents identified on Page 3.	[X]	[ ]	[ ]

If all answers are NO, then additional review or assistance is not required.

If any answers are YES or UNCERTAIN, then the Preparer shall obtain assistance from the System Engineer and other departments, as appropriate, to determine impacts to procedures and plant operations. Affected documents shall be listed on Attachment 2.

Nebraska Public Power District

DESIGN CALCULATIONS SHEET

PURPOSE:

This calculation incorporates by attachment Alion Science & Technology Calculation No. ALION-CAL-NPPD-3236-002, Rev. 1, in accordance with CNS Engineering Procedure 3.4.7. This calculation determines the dose to a Control Room occupant and to a person at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) at the Cooper Nuclear Station site following a design basis Loss of Coolant Accident (LOCA). The analysis is performed using an Alternative Source Term (AST) in accordance with the guidance provided by the NRC in Regulatory Guide 1.183 (July 2000) and as authorized by 10CFR50.67. This calculation has been prepared as a Status 3 calculation for NRC review and will be taken to Status 1 after approval of the LOCA AST methodology by the NRC. This calculation also presents the results from additional shine sources to the Control Room occupants.

Revision 1 increases MSIV leakage from 100scfh/MS line and 200 scfh MS pathway total, to 150 scfh/MS line and 300 scfh MS pathway total to the condenser. In Sections 2.4.1 and 2.4.2, new Main Steam Line decontamination factors were calculated for particulate and elemental deposition, based on the increased MSIV leakage flowrates; however, credit for decontamination via these mechanisms was removed. Section 2.4.2 was revised to remove the development of the condenser elemental iodine removal. New Appendix A, "Condenser Effective Filter Efficiency Calculation," was developed to calculate condenser effective filter efficiency for iodine removal. The results of Appendix A have been included as input parameters for Drywell-MSIV Leakage Pathway as calculated in Appendices D and E based on the BWROG model documented in NEDC 35858P-A, Design Input 15. As expected, the dose results presented in Tables 6-1 and 6-2 show an increase in dose from increasing the allowed leakage past the MSIVs. However, the resultant doses remain significantly below the acceptance criteria. The change in calculated doses are shown below. An additional review summary has been added for review and acceptance Appendix A of Revision 1 as this is added to the calculation by Revision 1.

EXTENT OF REVIEW

Alion's calculation was preformed under their own QA program, which included an independent technical review. Therefore, the NPPD review does not include in-depth checks of mathematical calculations, but rather focuses on general acceptability of design inputs, assumptions, methodology, and conclusions. Any significant comments or concerns identified during the review have been resolved with Alion and incorporated.

REVIEW SUMMARY

Alion's calculation is organized into a single main portion along with Attachments A through K, which include the computer files as well as Alion's Design Review Checklist.

1. Purpose - The purpose of the calculation is as given above and as stated in Section 1 of Alion's calculation.

2. Design Inputs - Design Inputs are contained in the Cross Reference Index given on page 6 of 292 of Alion's calculation and are discussed in Sections 2 and 3 of Alion's calculation. The design inputs were reviewed and found to be acceptable, with minor clarifications shown in the Cross Reference Index on Page 2 of 8.

3. Assumptions - Major assumptions are identified in Section 4 of Alion's calculation. Additional assumptions are inferred in the input documents used and identified throughout Alion's calculation by inference according to context and use. The assumptions were reviewed and found to be acceptable.

   Throughout ALION-CAL-NPPD-3236-002, the calculation refers to a "faulted" or "failed" MSIV (e.g., pages 21, 40, D3, D10, D14, H4, H14, H22 and K7. This language is potentially misleading, as the limiting single failure assumed in the analysis is the failure of a filter heater in one of the Standby Gas Treatment System trains. This is discussed in Section 2.3.2 of the calculation, and identified in Table 5-5, "Input Parameters for Drywell-Reactor Building Leakage Pathway." As discussed in NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Section 5.1.2, "Credit for Engineered Safeguards Features," the single active failure that results in the most limiting radiological consequences should be assumed. Therefore there is no requirement to assume an additional engineered safeguard failure, nor any impact to the dose consequences to assume the failure of an MSIV. In reality, the AST LOCA dose analysis assumed that a large portion (50%) of the MS Pathway leakage traveled down one MS line, and 50% traveled down a second MS line to calculate conservative deposition efficiencies for the MS lines. However, in Revision 1 of this calculation, credit for deposition in the MS lines was removed.

   ALION-CAL-NPPD-3236-002, Section 2.4.3 states that the two possible flowpaths from the MSIV leakage to the condenser: through the MS lines via the Alternate Leakage Treatment (ALT) pathway, or through the MS drain lines originating just downstream of the MSIVs. This is accurate; however, the total MS Pathway leakage includes the leakage from the MSIVs and the leakage from the inboard MS drain line, through Containment isolation valves MS-MOV-MO74 and MS-MOV-MO77 via Containment penetration X-8. Therefore, the total (aggregate) leakage will apply, as a limit, to the sum total of the leakage from all four MSIVs plus the leakage from Containment penetration X-8. This is acceptable, and does not alter either the model or the dose consequences reported in

ALION-CAL-NPPD-3236-002, because no credit is taken for deposition in the MS lines, and no time is assumed in the RADTRAD model for transport of MSIV leakage from the Drywell to the Condenser, and therefore no credit is taken for any radionuclide decay in the transport of the MSIV leakage between these two compartments.

4. Methodology - The methodology is described in Section 5 of Alion's calculation. The methodology was reviewed and found to be acceptable, with the following clarification. Dose consequences computed by RADTRAD include air immersion (shine) and inhalation (DI / Ref. 2). As stated in the PURPOSE, control room occupants could also receive immersion (shine) doses from other sources. These doses have been determined in calculation NEDC 05-045 (DI / Ref. 46). The results are given in the next section.

5. Conclusions and Recommendations - Results and conclusions are given in Section 6 of Alion's calculation. The results and conclusions were reviewed and found to be acceptable, with the following clarification. As stated in the PURPOSE, control room occupants could also receive immersion (shine) doses from other sources. These doses have been determined in calculation NEDC 05-045 (DI / Ref. 46). The results are given in the following tables, which replace those in Attachment 1. The revision 1 doses including the increase in MSIV leakage from 100/200 scfh to 150/300 scfh and the condenser effective filter efficiency are included under column for Rev 1.

TEDE Dose as a Function of Release Path

Dose Location	Leakage (rem TEDE)		MSIV w/Bypass (rem TEDE)		ESF (rem TEDE)		Additional Shine (rem TEDE)	
	Rev 0	Rev 1	Rev 0	Rev 1	Rev 0	Rev 1	Rev 0	Rev 1
Control Room	0.374	NC*	0.562	2.401	0.102	NC	0.153	0.319
EAB	0.458	NC	0.136	0.374	0.170	NC	N/A	N/A
LPZ	1.559	NC	0.489	2.311	1.727	NC	N/A	N/A

* NC – No change from previous Revision 0.

Total LOCA TEDE Dose

Dose Location	Total Dose (rem TEDE)		Accident Dose Criteria (rem TEDE)
	Rev 0	Rev 1	
Control Room	1.191	3.196	5
EAB	0.763	1.002	25
LPZ	3.775	5.596	25

6. References - References are listed in Section 8 of Alion's calculation. The references were reviewed and found to be acceptable, with minor clarifications shown on the Cross Reference Index on Page 2 of 8.

REVIEW SUMMARY FOR APPENDIX A- Condenser Effective Filter Efficiency Calculation

ALION's Appendix A Revision 1 of NEDC 07-082 is a stand alone calculation that requires inputs to the Design Input Sheet and is reviewed and accepted per Procedure 3.4.7 below: The PURPOSE and EXTENT OF REVIEW remain the same as provided above.

Purpose- The purpose this calculation is stated in Section 1 of Appendix A and is to determine the main condenser free volume above the highest penetration transporting MSIV leakage, and the main condenser's effective filter efficiency (radionuclide removal efficiency) to be credited in the main body of this calculation (NEDC 07-082 Rev. 1).

Design Inputs-Design Inputs specific to the Appendix A calculation have been added to the Design Inputs in the Cross Reference Index on Page 2 of 8. The design inputs required for Appendix A were reviewed and found to be acceptable with minor clarifications shown in the Cross Reference Index.

Assumptions- Major assumptions are identified in Section A3 of Appendix A. As some drawings did not include all required dimensioning for satisfying this calculation, some drawings were printed to scale and component measured. An additional 5 % margin was removed from the volume of the condenser as a conservative measure and was found to be conservative and thus acceptable.

Methodology- The methodology is described in Section A4 of the Appendix A. The methodology was reviewed and found to be acceptable, based on proven documentation and accepted industry references.

The analysis implements the methodology and determines the inputs for the computer calculation. Computations for determining the condenser effective filter efficiency is based on a NRC approved BWROG model documented in NEDC-31858P-A, Design Input 15. This methodology is found to be acceptable for application at CNS.

Conclusions and Recommendations- This is included in Section A6 of Appendix A. The condenser free volume was conservatively determined and the main condenser effective filter efficiencies were determined for three separate MSIV leak conditions. These are found to be acceptable and the appropriate values have been added to the body of this calculation, Table 5.4.2, and input to the RADTRAD model.

References-References are listed in Section A7 of Appendix A. In this appendix, design inputs were mixed in with the references; these design inputs have been added to the Design Inputs on Page 2 of 8. This has been reviewed and found to be acceptable.

ATTACHMENTS

1. Alion calculation ALION-CAL-NPPD-3236-002, Rev. 1 (including attachments thereto). Attachment A is new to this revision and has been found to be acceptable as stated above.

NEDC 07-082 Rev. 1

ATTACHMENT 1

ALION Calculation ALION-CAL-NPPD-3236-002, Revision 1

(including attachments thereto)

ALION
SCIENCE AND TECHNOLOGY

6000 Uptown Boulevard NE, Suite 300, Albuquerque, NM 87110
tel: 505.872.1089 fax: 505.872.0233
www.alionscience.com

September 5, 2008

Mr. Michael Bennett
Cooper Nuclear Station
P.O. Box 98
Brownville, NE 68321

Subject: AST Methodology Implementation at CNS

Reference: GSA No. 05A-MS3

Enclosure:

1. ALION-CAL-NPPD-3236-002, Revision 1, "Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station"
2. Certificate of Conformance for ALION-CAL-NPPD-3236-002 Revision 1.
3. Controlled Document Transmittal

Dear Mr. Bennett,

Enclosed please find the subject deliverable ALION-CAL-NPPD-3236-002, Revision 1(RESUBMITTED) Please complete the enclosed Alion Controlled Document Transmittal form and return it to the specified address at your earliest convenience. If you have any questions or comments regarding any of the information in the deliverables of this correspondence please do not hesitate to call me at (402) 545-3871.

Sincerely,

Jan Bostelman

Janice Bostelman
Project Manager

Enclosure
Cc: Project File (NPPD-3236)
B. Griego (QA) w/att
C. Garves

CERTIFICATE OF COMPLIANCE/CONFORMANCE

ALION
SCIENCE AND TECHNOLOGY

6000 Uptown Blvd., Suite 300 Albuquerque, NM 87110
Voice: 505·872·1089 Fax: 505·872·0233

Customer Name: Nebraska Public Power District (NPPD) Date: August 22, 2008

Customer P.O. Number: GSA 05A-M3 4700001039 Amendment 1, Dated July 2, 2008

ITSO Project Number: NPPD-3236

ITSO Project Plan Number: ALION-PLN-NPPD-3236-01 Rev. 2

P.O Item #	Description
2	ALION-CAL-NPPD-3236-002, Revision 1, entitled, "Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station"

This Certificate of Conformance/Compliance provides testimony that the services and resultant deliverables listed above have been performed and completed in accordance with requirements specified in the purchase order(s) cited above, including all referenced documents and any other customer requirements specified in formal correspondence.

This Certificate also provides verification that services specified in the purchase order(s) cited above were performed in accordance with the ITSO Quality Assurance Program, Revision 10, dated November 27, 2007. The ITSO QA Program complies with 10CFR50 Appendix B, 10CFR21, and NQA-1, 1989 requirements as applicable.

Project Manager: ______ Date: 9/3/08
Signature

QA Manager: ______ Deputy QA Manager Date: 9/3/08
Signature

Form 2.1.7
Revision 2
Effective Date: 2/28/07

ALION
SCIENCE AND TECHNOLOGY

DESIGN CALCULATION AND ANALYSIS COVER PAGE

Calculation No: **ALION-CAL-NPPD-3236-002**	Revision: 1	Page 1 of 51

Calculation Title: **Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station**

Project No: NPPD-3236

Project Name: **AST Methodology Implementation at CNS**

Client: Nebraska Public Power District (NPPD)

Document Purpose/Summary:
The purpose of this calculation is to determine the dose to the control room occupant and to a person at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) at the Cooper Nuclear Station site following a design basis Loss of Coolant Accident (LOCA). The analysis is performed using an Alternative Source Term (AST) in accordance with the guidance provided by the NRC in Regulatory Guide 1.183 (July 2000) and as allowed by 10 CFR 50.67. All calculated doses are shown to be below the regulatory limits for all three stipulated locations.

Per NPPD General Services Agreement No. 05A-MS3, Task Authorization 4700000649, Revision 0, July 15, 2005, the format of this calculation meets the requirements of Nebraska Public Power District procedure NPPD CNS Operations Manual Engineering Procedure 3.4.7"Design Calculations". Although the calculation format of the NPPD procedure differs slightly from ALION procedure QAP 3.4, Design Calculation and Analysis, revision 9, the formats are similar enough to ensure that the technical and programmatic content of the calculation is neither jeopardized nor compromised and meets the intent of the ALION and NPPD Quality Assurance programs

This calculation is safety related and complies with the Alion Science and Technology QA Program.
Appendix A is proprietary to Alion Science and Technology and cannot be duplicated or disclosed outside of NPPD.

All software used in the preparation of this calculation meets QAP 3.5, *Use of Computer Software and Error Reporting requirements.*

Preparer Signature: [signature] Date 18 Aug 2008

DESIGN VERIFICATION METHOD	QA APPLICABILITY LEVEL
☒ Design Review	☒ Nuclear Safety Related
☐ Alternative Calculation	☐ Quality Significant
☐ Qualification Testing	☐ Nuclear Non-Safety Related

Professional Engineer Approval (if required) N/A Date ______
Signature

	Printed/Typed Name	Signature	Date
Prepared By:	Terry Heames	[signature]	18 Aug 2008
	Ronald Anderson	[signature]	8/19/08
Reviewed By:	Jan Bostelman	[signature]	8/19/08
	Juan Cajigas	[signature]	08-19-08
Approved By:	Todd Anselmi	[signature]	08.19.08

Form 3.4.1
Revision 3
Effective Date: 2/28/07

ALION
SCIENCE AND TECHNOLOGY

REVISION HISTORY LOG

Document Number: **ALION-CAL-NPPD-3236-002** Revision: 1

Document Title: Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station

Instructions:
Project Manager to provide a brief description of each document revision, including rationale for the change and, if applicable, identification of source documents used for the change.

REVISION	DATE	Description
0	13 July 2007	Original Issue
1	7 August 2008	Modify MSIV leak path decontamination