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SUBJECT: Forwards W descriptive documentation re annual rept of LOCA model change PCT assessment,per 10CFR50.45(a)(3)(ii).

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INDIANA
MICHIGAN
POWER

June 3, 1998

AEP:NRC:1118M

Docket Nos.: 50-315
50-316

U. S. Nuclear Regulatory Commission
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Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
ANNUAL REPORT OF LOCA EVALUATION MODEL CHANGES

Pursuant to the requirements of 10 CFR 50.46(a)(3)(ii), this letter provides our annual submittal of loss-of-coolant accident (LOCA) model changes.

Attachment 1, provided to us by Westinghouse Electric Corporation, contains descriptive documentation related to the only permanent LOCA model change peak cladding temperature (PCT) assessment being reported this year. A review of past design changes to the emergency core cooling system (ECCS) was performed as part of the unit 2 power uprate program. The result of this review indicated that one of the changes could potentially affect the PCT. The change involved the installation of pressure equalizing lines on the hot leg injection, double disk gate, and isolation valves in the 1984/1985 time frame. Other changes that were reviewed either had no affect or had been incorporated into the Westinghouse ECCS model. The material in attachment 1 includes a generic description of the impact of pressure equalization lines, provided by Westinghouse Electric Corporation in 1994, and correspondence relating to the review of past design changes. Attachment 1 also discusses the plant-specific, permanent PCT assessment prepared for us by Westinghouse.

The impact of the pressure equalizing lines is the only PCT assessment reported in this submittal. However, due to the extended shutdown of both units, this letter also addresses the impact on the schedules for re-analysis that we have provided in previous reports.

Attachment 2 contains the PCTs calculated specifically for Cook Nuclear Plant units 1 and 2. In all cases, the calculated PCTs remain within the 10 CFR 50.46 limit of 2200°F. In accordance with the guidance in WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting", evaluations of plant changes performed under 10 CFR 50.59 are not reported under 10 CFR 50.46 unless an offsetting change to the evaluation model was made to provide sufficient margin to accommodate the proposed change.

A001

We have included two margin utilization sheets for the unit 1 large break loss-of-coolant accident (LBLOCA) in attachment 2. The first margin utilization sheet on page 2 of the attachment shows the chopped cosine analysis of record as described in our submittal AEP:NRC:1207, dated May 26, 1995, with LOCA model assessments from 1995, 1996, and 1997. The second sheet (page 3 of the attachment) includes the LOCA model assessment for the change in power shape methodologies previously reported by our letter AEP:NRC:1207A, dated September 26, 1995. This margin utilization sheet has been provided to comply with the requirements of 10 CFR 50.46 regarding a "...change or error in an acceptable evaluation model, or in the application of such a model that affects the temperature calculation,...". As described in our letter AEP:NRC:1207A, the power shape sensitivity model (PSSM) was replaced by the explicit shape analysis for peak cladding temperature effects (ESHAPE), and the hot leg nozzle gap model.

In discussions with the NRC staff (January 23 and 27, 1997), it was agreed that we would report the changes in the evaluation model resulting from the change in power shape methodologies as an assessment to the analysis of record. It is provided for comparison purposes with the chopped cosine power shape to demonstrate that the cosine power shape is limiting when full credit is given for the hot leg nozzle gap benefit.

Because the hot leg nozzle gap model is not yet approved, we also agreed to provide a schedule for re-analysis using acceptable evaluation models. The schedule for re-analysis was submitted by our letter AEP:NRC:1207C, dated February 6, 1997. The schedule provided was, "Although we have not yet identified the best solution, we believe a revised submittal with an approved evaluation model can be made prior to the start-up of cycle 18. Cycle 18 is currently scheduled to begin on April 20, 2000. If our discussions with Westinghouse should result in a need to change our proposed schedule, a submittal to the staff describing the change will be made." Both Cook Nuclear Plant units are currently shut down as a result of issues identified during an NRC design inspection. Although we continue to believe that a revised submittal with an approved model can be made prior to the start-up of cycle 18, the beginning date for cycle 18 is uncertain at this time.

As is the case for the unit 1 LBLOCA, PCT assessments for the LBLOCA and small break loss-of-coolant accident (SBLOCA) analyses of record for unit 2, documented in attachment 2, exceed the 50°F requirement for committing to a new analysis. As indicated in our 1996 10 CFR 50.46 annual report, our submittal AEP:NRC:1118K, dated March 22, 1996, new analyses were scheduled for submittal. Both the LBLOCA and SBLOCA re-analyses documented in attachment 3 were submitted to your staff in our letter AEP:NRC:1223, dated July 11, 1996, with subsequent clarifications submitted in our letter AEP:NRC:1223A, dated September 27, 1996. The data in attachment 3 is provided for information, although it does not reflect the current licensing basis for unit 2.

Submittals AEP:NRC:1223 and AEP:NRC:1223A consisted of evaluations and analyses to support an uprated power level for unit 2. Your staff initiated its review of these analyses. The review resulted in a number of requests for additional information (RAIs). In particular, we were asked to address the possibility of loop seal

re-plugging during post-LOCA recovery from a SBLOCA. We supplied a Westinghouse analysis of this issue in our letter AEP:NRC:1223B, dated April 30, 1997. The non-proprietary version of this report was submitted in our letter AEP:NRC:1223G, dated August 1, 1997. We responded to one other RAI that included a question related to the LOCA analyses.

However, in our letter AEP:NRC:1223M, dated September 10, 1997, we requested that the NRC review of the unit 2 uprate be temporarily suspended. At this time, no schedule has been established for a request that the staff resume the review. We will resubmit portions of the unit 2 uprate proposal that include the LOCA analyses for which the staff review has been suspended. This submittal will be made prior to 120 days after the restart of unit 2. The resubmittal will also include responses to all outstanding RAIs pertinent to the portions of the unit 2 uprate proposal that are resubmitted.

In addition, the small break LOCA assessments of attachment 3 exceed 50°F. A commitment for a new analysis will be made in our first report after the SBLOCA of attachment 3 becomes the analysis of record.

Sincerely,

J. R. Sampson

J. R. Sampson
Vice President

/vlb

Attachments

c: J. A. Abramson
MDEQ - DW & RPD
NRC Resident Inspector
C. J. Paperiello

ATTACHMENT 1 TO AEP:NRC:1118M

WESTINGHOUSE ELECTRIC CORPORATION
DESCRIPTION OF LOCA MODEL CHANGES

DOUBLE-DISK GATE VALVE PRESSURE EQUALIZATION

Background

Westinghouse completed the evaluation of a potential issue concerning use of double-disk gate valves in the emergency core cooling system (ECCS) as hot leg isolation valves. Use of these double-disk gate valves may involve an inner disc pressure equalization line that could set up a leak path into the hot leg during cold leg injection following a loss of coolant accident (LOCA). This condition could lead to inadequate cold leg injection resulting in an increase in PCT.

The design characteristic of a double-disk gate valve provides isolation by the downstream disk sealing against the valve seat. The mechanical seating force and the hydraulic force from the upstream pressure (SI pump) act to provide force to the valve seal surfaces. The double-disk gate valve design results in a volume of fluid which is enclosed between the discs when the valve is closed. As the fluid volume heats up, pressure greater than system pressure may develop and may cause the disks to bind against the seats to the extent that the valves can not be opened. To avoid this, many double-disk gate valves have been modified to include a pressure equalization line or a small hole in one of the disks to relieve the pressure between the disks. Based on generic leakage calculations it was determined that the double-disk gate valves modified to eliminate concerns for thermal binding could leak as much as 30 gpm per valve. This leakage into the RCS hot legs will increase steam binding during reflood and result in an increase in the calculated peak cladding temperature.

Affected Evaluation Models

1975 SBLOCA Evaluation Model
1985 SBLOCA Evaluation Model
1978 ECCS Evaluation Model
1981 ECCS Evaluation Model
1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

Estimated Effect

The PCT effect on the Large Break LOCA Evaluation Model for this issue varied depending on the affected plant ECCS configuration and capability. The specific PCT penalty, for affected plants, is shown on the attached PCT Summary Sheet. An assessment of this issue on Small Break LOCA Evaluation Model PCT results showed a nominal benefit which is being reported generically as a 0°F impact.

AEP-97-191

Westinghouse Electric Corporation

Energy Systems

Nuclear Services Division

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-SAE-ESI-97-623

November 05, 1997

Mr. John Olvera
American Electric Power
500 Circle Drive
Buchanan, Michigan 49107

AMERICAN ELECTRIC POWER
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
<u>ECCS Double Disk Valve Leakage and Revised LOCA PCT Margin Utilization Sheets</u>

References:

(1) Westinghouse letter AEP-94-214, "American Electric Power Service Corporation D. C. Cook Units 1 and 2, 10 CFR 50.46 Notification and Reporting Information", February 8, 1994.
(2) American Electric Power letter A3600-16, "Review of Design Changes for Potential Impact on ECCS Flow for the Donald C. Cook Nuclear Plant - Supplemental Information", September 26, 1997.
(3) Westinghouse letter AEP-97-203, "American Electric Power Service Corporation, D. C. Cook Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting", February 17, 1997.
(4) Westinghouse letter AEP-97-105, "American Electric Power, Donald C. Cook Nuclear Plant Unit 2, LOCA Peak Clad Temperature Margin Utilization Sheets for Future Analysis of Record", July 8, 1997.

Dear Mr. Olvera:

The purpose of this letter is to provide the results of an evaluation of the effect of ECCS double-disk valve leakage on the Donald C. Cook Nuclear Plant Units 1 and 2 LOCA analyses. The LOCA PCT margin utilization sheets for Donald C. Cook Nuclear Plant Units 1 and 2 have been updated to reflect the results of the evaluation, and the updated sheets are provided in Attachment 1 of this letter.

A potential issue was previously evaluated by Westinghouse to address concerns related to the use of double-disk gate valves in Emergency Core Cooling Systems (ECCS) to isolate the hot leg flow paths from the discharge of the SI pumps and the RHR pumps. It was determined that inner disk pressure equalization may be utilized in these valves which results in leak paths into the hot legs during cold leg injection following a LOCA, and that the diversion of flow from the cold legs into the hot legs may affect the calculated PCT. This issue was reported to American Electric Power in Westinghouse Nuclear Safety Advisory Letter NSAL-94-004A which was transmitted by Reference 1. There were no LOCA PCT assessments made for this issue for Donald. C. Cook Nuclear Plant Units 1 and 2 since the available information at that time indicated that the plants were not affected. However, it was subsequently determined that this issue is applicable for Donald C. Cook Nuclear Plant Units 1 and 2. Reference 2 indicates that equalizing lines have been added to the double-disk gate valves which provide isolation of the hot leg injection headers during cold leg injection and recirculation. The equalizing lines relieve pressure between the disks to a point downstream of the valves, which may result in a small amount of leakage to the hot legs during cold leg injection and recirculation.

NSD-SAE-ESI-97-623
AEP-97-191
November 05, 1997

Westinghouse has performed an evaluation to assess the impact on the Donald C. Cook Nuclear Plant Units 1 and 2 LOCA analyses due to the potential ECCS leakage from the cold legs to the hot legs. It was previously determined that the leakage would not adversely affect the small break LOCA analysis. However, the diversion of the water from the cold legs to the hot legs does affect the core inlet flooding rate for a large break LOCA, which results in a small PCT penalty for the large break LOCA analysis. The maximum ECCS leakage into the hot legs was conservatively estimated to be 3.2 gpm per valve for the large break LOCA analysis conditions. Although only one ECCS train is assumed to operate for the large break LOCA analysis, the RHR cross-tie valves are normally closed and the HHSI cross-tie valves are normally open, such that the isolation valve in both of the ECCS trains would be subject to the leakage. It was determined that a total leakage rate of 6.4 gpm would result in an increase in the large break LOCA analysis PCT of 8 °F for Donald C. Cook Nuclear Plant Units 1 and 2. It is also concluded that the ECCS double-disk valve leakage will not adversely affect the capability to provide adequate cold leg recirculation flow.

The LOCA PCT Margin Utilization Sheets provided previously in References 3 and 4 have been updated to reflect the large break LOCA PCT penalty due to the ECCS valve leakage, and the revised sheets are attached for your information and use. It is noted that the PCT for the large break LOCA analyses remain below the 10 CFR 50.46 limit of 2200 °F. The small break LOCA PCT margin utilization sheets are also included for completeness, but the PCT remains the same as reported in References 3 and 4.

Should you have any questions, please contact Mr. Don Peck (412-374-5683) or me.

Very Truly Yours,

Donald E. Peck /for

Joe Waleko
Account Manager
North American Field Sales

DEP/kk

cc: Jeb Kingseed - AEP
Vance VanderBurg - AEP

ATTACHMENT 2 TO AEP:NRC:1118M

WESTINGHOUSE ELECTRIC CORPORATION
DETERMINATION OF EFFECT OF LOCA MODEL CHANGES ON
COOK NUCLEAR PLANT LOCA ANALYSES

LARGE BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 1

Comments: Evaluation Model:BASH, FQT=2.15, FdH=1.55, SGTP= 30% Other: RHR Cross Tie Valve Closed, 3250 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT =	2164°F
B.	PRIOR LOCA ASSESSMENTS - 1995[1]		
	1. SALIBRARY Double Precision Errors	ΔPCT =	-5°F
	2. Skewed Power Shape Penalty[2]	ΔPCT =	N/A
	3. Hot Leg Nozzle Gap Benefit[2]	ΔPCT =	N/A
C.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	+6°F
D.	1997 10 CFR 50.46 MODEL ASSESSMENTS		
	1. ECCS Double Disk Valve Leakage	ΔPCT =	+8°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	2173°F

Notes:

1. All permanent loss-of-coolant accident (LOCA) model assessments prior to 1995 were resolved by the new analysis.

2. This large break loss-of-coolant accident (LBLOCA) analysis uses the power shape sensitivity model (PSSM) with a cosine axial power distribution. Since the PSSM has been withdrawn from NRC review, the explicit shape analysis for peak cladding temperature effects (ESHAPE) methodology has been incorporated into the Westinghouse LOCA evaluation model. However, because the calculated peak cladding temperature (PCT) with a cosine power shape is reached at approximately 77 seconds, the effect of the skewed power shape penalty and hot leg nozzle gap benefit is not significant. The application of the skewed power shape (shown on page 3 of this attachment) with the hot leg nozzle gap model shows that the cosine power shape remains limiting.

LARGE BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 1

Comments: Evaluation Model:BASH, FQT=2.15, FdH=1.55, SGTP= 30% Other: RHR Cross Tie Valve Closed, 3250 MWt Reactor Power

A.	ANALYSIS OF RECORD[1]	PCT =	2266°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1995[2]		
	1. SALIBRARY Double Precision Errors	ΔPCT =	-5°F
	2. Hot Leg Nozzle Gap Benefit[1]	ΔPCT =	-237°F
C.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	+6°F
D.	1997 10 CFR 50.46 MODEL ASSESSMENTS		
	1. ECCS Double Disk Valve Leakage	ΔPCT =	+8°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	2038°F

Notes:

1. This PCT is the result of the change in axial power distribution models from the PSSM to the ESHAPE methodology. Because the PSSM has been withdrawn from NRC review, the ESHAPE methodology has been incorporated into the analysis. An analysis for this case with the limiting skewed power shape resulted in a PCT increase of 102°F over the analysis performed using the cosine axial power distribution. The application of the hot leg nozzle gap model results in a benefit of 237°F. This shows that the cosine power shape (shown on page 2 of this attachment) remains limiting.

2. All permanent LOCA model assessments prior to 1995 were resolved by the new analysis.

SMALL BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 1

Comments: Evaluation Model:NOTRUMP, FQ=2.32, F*H=1.55, SGTP=30% Other: HHSI Cross Tie Valve Closed, 3250 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT	=	1443°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1995[1]	ΔPCT	=	+5°F
C.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT	=	+10°F
D.	1997 10 CFR 50.46 MODEL ASSESSMENTS	ΔPCT	=	+0°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT	=	1458°F

Notes:

1. All permanent LOCA model assessments prior to 1995 were resolved by the new analysis.

LARGE BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model:BASH, FQT=2.335, FdH=1.644, SGTP= 15%, Other: RHR Cross Tie Valve Closed, 3413 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT	=	2090°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1989[1]	ΔPCT	=	N/A
C.	PRIOR LOCA MODEL ASSESSMENTS - 1990	ΔPCT	=	0°F
D.	PRIOR LOCA MODEL ASSESSMENTS - 1991	ΔPCT	=	+30°F
E.	PRIOR LOCA MODEL ASSESSMENTS - 1992	ΔPCT	=	-25°F
F.	PRIOR LOCA MODEL ASSESSMENTS - 1993	ΔPCT	=	-6°F
G.	PRIOR LOCA MODEL ASSESSMENTS - 1994	ΔPCT	=	0°F
H.	PRIOR LOCA MODEL ASSESSMENTS - 1995	ΔPCT	=	+16°F
I.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT	=	+15°F
J.	1997 10 CFR 50.46 MODEL ASSESSMENTS			
	1. ECCS Double Disk Valve Leakage	ΔPCT	=	+8°F
K.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT	=	2128°F

Notes:

1. Analysis of record was completed in January 1990. No prior LOCA model assessments were made.

LARGE BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model:BASH, FQT=2.22, FdH=1.62, SGTP= 15%, Other: RHR Cross Tie Valve Open, 3588 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT =	2140°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1989[1]	ΔPCT =	N/A
C.	PRIOR LOCA MODEL ASSESSMENTS - 1990	ΔPCT =	0°F
D.	PRIOR LOCA MODEL ASSESSMENTS - 1991	ΔPCT =	+30°F
E.	PRIOR LOCA MODEL ASSESSMENTS - 1992	ΔPCT =	-25°F
F.	PRIOR LOCA MODEL ASSESSMENTS - 1993	ΔPCT =	-6°F
G.	PRIOR LOCA MODEL ASSESSMENTS - 1994	ΔPCT =	0°F
H.	PRIOR LOCA MODEL ASSESSMENTS - 1995	ΔPCT =	+16°F
I.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	+15°F
J.	OTHER MARGIN ALLOCATIONS		
	1. Power Margin[2]	ΔPCT =	-98°F
K.	1997 10 CFR 50.46 MODEL ASSESSMENTS		
	1. ECCS Double Disk Valve Leakage	ΔPCT =	+8°F
L.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	2080°F

Notes:

1. Analysis of record was completed in January 1990. No prior LOCA model assessments were made.

2. This value was obtained by temporarily allocating 4.9% of power margin using a sensitivity of 20°F/% power. See the justification for the use of power margin in the Cook Nuclear Plant unit 2 large break PCT rack up on page 7 of this attachment.

JUSTIFICATION FOR USE OF POWER MARGIN
IN COOK NUCLEAR PLANT UNIT 2 LARGE BREAK PCT RACK UP

A sensitivity to power was previously determined for the Cook Nuclear Plant Unit 2 large break analysis. It was conservatively demonstrated that a reduction of $20°F_{PCT}/\%$ power could be applied for reduced power. This sensitivity will be applied to the reduction in power from the unit 2 analysis power of 3588 MW_t to the licensed operating condition of 3413 MW_t (a 4.9% reduction in power):

$$(20°F_{PCT}/\% \text{ Power})\ (4.9\% \text{ Power}) = 98°F$$

This sensitivity is conservative because it only accounts for the assumed power reduction in the LOCBART run. A similar reduction in the assumed power for the SATAN run produces an added benefit to PCT during the blowdown portion of the transient. A reduction in power in the blowdown portion of the transient (i.e., SATAN) would be an added benefit that was not accounted for in this sensitivity.

SMALL BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model:NOTRUMP, FQT=2.45, FdH=1.666, SGTP= 15%, Other: HHSI Cross Tie Valve Closed, 3250 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT	=	1956°F
B.	PRIOR LOCA MODEL ASSESSMENTS - October 1993	ΔPCT	=	-13°F
C.	PRIOR LOCA MODEL ASSESSMENTS - March 1994	ΔPCT	=	-16°F
D.	PRIOR LOCA MODEL ASSESSMENTS - December 1994	ΔPCT	=	+69°F
E.	PRIOR LOCA MODEL ASSESSMENTS - 1995	ΔPCT	=	+20°F
F.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT	=	-28°F
G.	1997 10 CFR 50.46 MODEL ASSESSMENTS	ΔPCT	=	0°F
H.	Burst and Blockage/Time in Life[1]	ΔPCT	=	0°F
I.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT	=	1988°F

Notes:

1. It should be noted that the burst and blockage assessment is subject to change as other model assessments are made because the magnitude of the burst and blockage assessments depends on the PCT without burst and blockage.

SMALL BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model: NOTRUMP, FQT=2.44, FdH=1.644, SGTP= 15%, Other: HHSI Cross Tie Valve Closed, 3413 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT =	1947°F
B.	PRIOR LOCA MODEL ASSESSMENTS - October 1993	ΔPCT =	-13°F
C.	PRIOR LOCA MODEL ASSESSMENTS - March 1994	ΔPCT =	-16°F
D.	PRIOR LOCA MODEL ASSESSMENTS - December 1994	ΔPCT =	-33°F
E.	PRIOR LOCA MODEL ASSESSMENTS - 1995	ΔPCT =	+20°F
F.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	-28°F
G.	1997 10 CFR 50.46 MODEL ASSESSMENTS	ΔPCT =	0°F
H.	Burst and Blockage/Time in Life[1]	ΔPCT =	+52°F
I.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	1929°F

Notes:

1. It should be noted that the burst and blockage assessment is subject to change as other model assessments are made because the magnitude of the burst and blockage assessments depends on the PCT without burst and blockage.

SMALL BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model:NOTRUMP, FQT=2.32, FdH=1.62, SGTP= 15%,
Other: HHSI Cross Tie Valve Open, 3588 MWt Reactor Power

A.	ANALYSIS OF RECORD	PCT =	1531°F
B.	PRIOR LOCA MODEL ASSESSMENTS - October 1993	ΔPCT =	-13°F
C.	PRIOR LOCA MODEL ASSESSMENTS - March 1994	ΔPCT =	-16°F
D.	PRIOR LOCA MODEL ASSESSMENTS - December 1994	ΔPCT =	+35°F
E.	PRIOR LOCA MODEL ASSESSMENTS - 1995	ΔPCT =	+20°F
F.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	-28°F
G.	1997 10 CFR 50.46 MODEL ASSESSMENTS	ΔPCT =	0°F
H.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	1529°F

ATTACHMENT 3 TO AEP:NRC:1118K

WESTINGHOUSE ELECTRIC CORPORATION
DETERMINATION OF EFFECT OF LOCA MODEL CHANGES ON
UNAPPROVED COOK NUCLEAR PLANT LOCA ANALYSES

5

LARGE BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model: BASH, FQT=2.335, FdH=1.644, SGTP= 15%, Other: RHR Cross Tie Valve Closed, 3588 MWt Reactor Power

A.	NOT YET APPROVED ANALYSIS OF RECORD	PCT =	2051°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1995[1]	ΔPCT =	0°F
C.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	+2°F
D.	1997 10 CFR 50.46 MODEL ASSESSMENTS		
	1. ECCS Double Disk Valve Leakage	ΔPCT =	+8°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	2061°F

Notes:

1. All prior permanent LOCA model assessments were resolved by the new analysis.

SMALL BREAK LOCA

PLANT NAME: Donald C. Cook Nuclear Plant Unit 2

Comments: Evaluation Model: NOTRUMP, FQT=2.32, FdH=1.62, SGTP= 15%, Other: HHSI Cross Tie Valve Closed, 3588 MWt Reactor Power

A.	NOT YET APPROVED ANALYSIS OF RECORD	PCT =	2065°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1995[1]	ΔPCT =	+5°F
C.	PRIOR LOCA MODEL ASSESSMENTS - 1996	ΔPCT =	-28°F
D.	1997 10 CFR 50.46 MODEL ASSESSMENTS	ΔPCT =	0°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT =	2042°F

Notes:

1. All permanent LOCA model assessments prior to 1995 were resolved by the new analysis.