NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Duane Arnold Energy Center	05000-331	1 of 5

TITLE (4)
Plant Shutdown Due to Shutdown Margin Calculation Error

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
04	17	95	95	-- 003 --	00	05	10	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE: 2
POWER LEVEL: 0

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

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Ronald McGee, Licensing Specialist	(319) 851-7602

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

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

SUPPLEMENTAL REPORT EXPECTED (14)

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 17, 1995, a reactor shutdown was completed based on a manually calculated shutdown margin that did not meet the minimum shutdown margin required by Technical Specifications. Subsequent recalculation by the vendor resulted in a verified shutdown margin value that was adequate. The initial calculation error was due to vendor supplied information and manual calculation inaccuracies. Corrective actions include a review of the event by the vendor (assisted by facility personnel) and improvements to the manual calculation methodology.

This Licensee Event Report is being submitted voluntarily.

I. DESCRIPTION:

On April 17,1995, a reactor startup was in progress following a refueling outage at the Duane Arnold Energy Center (DAEC). Upon reaching initial criticality at 0650 hours, a surveillance test was performed in accordance with Technical Specifications to verify the shutdown margin (SDM). SDM is determined by manual calculation utilizing core parameters such as moderator temperature, control rod positions, and reactor period at the point of initial criticality following core refueling. These parameters are used in conjunction with appropriate graphs and tables to obtain the reactivity factors which are then summed to obtain SDM. The results of the initial calculation indicated a SDM value of 0.33% delta K/K. Technical Specifications required a minimum SDM value of 0.38% delta K/K or be in cold shutdown within 24 hours. Based on the results of the initial calculation, reactor shutdown was commenced at 0658 hours. Cold shutdown mode was achieved at 0718 hours. A more detailed calculation performed after plant shutdown indicated 0.3566% delta K/K SDM. The original data for this cycle's core configuration identified an expected SDM of 1.05% delta K/K.

The vendor, General Electric (GE), was contacted to review the parameters recorded at initial criticality and verify the SDM value. Utilizing the computerized code that the manual SDM calculations, tables and graphs are based on, GE returned a verified SDM of 0.451% delta K/K.

Following a review of this event by plant management and verification by GE that adequate SDM existed during the initial reactor startup, reactor restart was authorized. Criticality was achieved on April 17, 1995, at 1723 hours, with SDM calculated to be 0.433% delta K/K.

II. CAUSE:

Based on our review of the initial SDM parameters and calculations, it was determined that variances in three of the reactivity factors resulted in the initial calculated values being less than the Technical Specification required minimum. The SDM calculation includes the following five reactivity factors:

$K_{CRIT} - K_{SRO} + K_{MOD} - K_{PER} - K_{R} = SDM$

The DAEC manual calculation for SDM performed immediately atter the reactor shutdown used the following values:

$$0.999016 - (0.9895) + (-0.0057) - (0.00025) - (0.0) = 0.003566 = 0.3566\%$$

The GE computerized calculation utilized the following values:

$$0.99910 - (0.98908) + (-0.00525) - (0.00026) - (0.0) = 0.00451 = 0.451\%$$

As can be seen in the above calculations, the significant differences between the DAEC calculation and the vendor supplied calculation occur in the K_{SRO} and K_{MOD} reactivity factors. Of lesser significance was the change in K_{CRIT}.

K_{SRO} is the 'strongest rod out' reactivity factor. This value is supplied bv GE as part of the fuel reload process. The value initially utilized in the DAEC calculations (0.9895) was based on a rod worth estimator value. The value used in the subsequent GE calculation (0.98908) was based on an actual one-rod-out calculation. The difference between these two values is attributed to the two calculation methodologies (rod worth estimator vs. one-rod-out calculation).

K_{MOD} is the moderator temperature correction reactivity value (see attachment 1). K_{MOD} is determined from the cycle specific graph utilizing the moderator temperature and Control Rod Density (CRD) at the time of initial criticality. Since CRD at initial criticality did not correspond exactly to one of the designated curves on the graph, its value had to be interpreted between curves B and C. The manual curvilinear interpretation resulted in a value of (-0.0057). The subsequent computer calculated value supplied by the vendor was (-0.00525).

K_{CRIT} is the calculated neutron multiplication at the rod position where criticality occurs. K_{CRIT} is determined from cycle specific tables for various rod positions. Since criticality occurred at rod position 22, the value of K_{CRIT} had to be estimated from the K_{CRIT} for the rod full in (position 0) and full out (position 48). The manual calculation of K_{CRIT} was 0.999016. The subsequent computer calculated value supplied by the vendor was 0.99910.

As described above, the most significant variances involved the K_{SRO} and K_{MOD} reactivity factors. The K_{MOD} variance was due to human factors considerations (ie; curvilinear interpretation) which made it very difficult to accurately determine this value. The K_{SRO} value is vendor supplied.

III. ANALYSIS OF EVENT:

There was no effect on safe plant operation. Adequate shutdown margin was available at all times. The SDM surveillance promptly and effectively identified discrepancies related to shutdown margin and the facility took conservative actions (plant shutdown) while the discrepancies were being resolved. The Technical Specification limiting conditions for operation action statement was promptly met. All actions taken were completed without complication.

IV. CORRECTIVE ACTIONS:

DAEC will implement process improvements to enhance the facility's ability to accurately determine the values of reactivity factors that are calculated on-site (such as K_{MOD}). These improvements will be in place prior to the next scheduled refueling outage.

The vendor is currently reviewing their processes to determine the cause of: 1) misprediction of designed SDM (1.05% delta K/K) and 2) the K_{SRO} variance. Representatives from the DAEC Quality Assurance and Reactor Engineering organizations are participating in this review.

V. ADDITIONAL INFORMATION:

There were no failed components or previous similar events associated with this occurrence.

This event is being submitted as a voluntary LER.

NRC FORM 366A
(5-92)

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TEXT CONTINUATION

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Duane Arnold Energy Center	05000-331	95	-- 003 --	00	5 of 5

FIGURE 1 — MODERATOR TEMPERATURE CORRECTION