NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

DOCKET NUMBER: 50-331

FILE NUMBER:

TO: Mr. Edson G. Case

FROM: Iowa Electric Light & Pwr. Company, Cedar Rapids, Iowa, Lee Liu

DATE OF DOCUMENT: 5/5/77

DATE RECEIVED: 5/5/77

☒ LETTER ☒ ORIGINAL ☐ COPY ☐ NOTORIZED ☒ UNCLASSIFIED

PROP

INPUT FORM

NUMBER OF COPIES RECEIVED: 3 SIGNED

DESCRIPTION

Ltr. trans the following:

DO NOT REMOVE

(1-P)

ENCLOSURE

Amdt. to OL/change to Appendix A tech specs...to amend their application dated 1/31/77 to reflect the replacement of Dresser safety relief valves with Target Rock safety relief valves.....

(26-P)

PLANT NAME: Duane Arnold

RJL

ACKNOWLEDGED

FOR ACTION/INFORMATION

SAFETY		ENVIRO
ASSIGNED AD:		ASSIGNED AD:
BRANCH CHIEF:	Lear (S)	BRANCH CHIEF:
PROJECT MANAGER:	Wetmore	PROJECT MANAGER:
LIC. ASST.:	Parrish	LIC. ASST.:

INTERNAL DISTRIBUTION

REG FILE	SYSTEMS SAFETY	PLANT SYSTEMS	SITE SAFETY & [illegible]
NRC PDR	HEINEMAN	TEDESCO	ENVIRO ANALYSIS
I & E (2)	SCHROEDER	BENAROYA	DENTON & MULLER
OELD		LAINAS	
GOSSICK & STAFF	ENGINEERING	IPPOLITO	ENVIRO TECH.
MIPC	MACARRY	KIRKWOOD	ERNST
CASE	BOSNAK		BALLARD
HANAUER	SIHWEIL	OPERATING REACTORS	SPANGLER
HARLESS	PAWLICKI	STELLO	
			SITE TECH.
PROJECT MANAGEMENT	REACTOR SAFETY	OPERATING TECH.	GAMMILL
BOYD	ROSS	EISENHUT	STEPP
P. COLLINS	NOVAK	SHAO	HULMAN
HOUSTON	ROSZTOCZY	BAER	
PETERSON	CHECK	BUTLER	SITE ANALYSIS
MELTZ		GRIMES	VOLLMER
HELTEMES	AT & I		BUNCH
SKOVHOLT	SALTZMAN		J. COLLINS
	RUTBERG		KREGER

EXTERNAL DISTRIBUTION

LPDR: Cedar Rapids Ia	NAT. LAB:	BROOKHAVEN NAT. LAB.
TIC:	REG V.IE	ULRIKSON (ORNL)
NSIC:	LA PDR	
ASLB:	CONSULTANTS:	
ACRS 16 CYS ~~HOLDING~~/SENT	AS CAT B	

CONTROL NUMBER

Ap 2

771260235 (D)

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
CEDAR RAPIDS, IOWA

May 5, 1977
IE-77-884

LEE LIU
VICE PRESIDENT – ENGINEERING

Mr. Edson G. Case
Acting Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Case:

In accordance with 10CFR50.59 and 50.90, we transmitted our application for amendment of DPR-49 and the Technical Specifications (Appendix A to License) for the Duane Arnold Energy Center (DAEC) for Cycle 3 operational limits and safety limits. We hereby amend our application dated January 31, 1977 to reflect the replacement of Dresser safety relief valves with Target Rock safety relief valves.

This amendment consists of:

1) Supplemental submittal in the format of Appendix A to NEDO 20360 which includes appropriate safety and transient analyses.

2) Amended proposed Technical Specification (RTS-80A) reflecting the results of the above analyses.

The DAEC is presently shutdown with restart scheduled for May 6, 1977.

Three signed and 40 additional copies of this submittal are transmitted herewith. This submittal, consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

Very truly yours,

Lee Liu
Vice President, Engineering

LL/KAM/ms
Encls.
cc: K. Meyer
D. Arnold
R. Lowenstein
J. Wetmore (NRC)
L. Root
File J-60a

PROPOSED CHANGE RTS-80A TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Specifications 1.1, 1.2, 3.2, 3.3, 3.6, 3.12 and 5.2 contain Safety Limits, Limiting Conditions for Operation and Bases which are applicable for cycle 2.

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Change the "80 psi" on sheet 3.6-24 of proposed Technical Specification Change RTS-80 previously submitted to the NRC (Letter IE-77-220, Mr. D. Arnold, President, Iowa Electric Light and Power Company to Mr. B.C. Rusche, Director, Office of Nuclear Reactor Regulation, Dated January 31, 1977) to "79 psi". On the same sheet change "40 psi" to "36 psi".

All other portions of RTS-80 remain as previously submitted.

III. Justification for Proposed Change

This change is proposed in order to incorporate into the Technical Specification bases the reactor vessel pressure margin to the code allowable over pressure limit and the pressure margin between the relief valve and safety valve settings which results from removing the Dresser Industries Relief Valves and replacing them with Target Rock Relief Valves.

The safety evaluation for this change is contained in the attached.

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

the direct scram (valve position scram) results in greater than a 79 psi margin to the code allowable overpressure limit of 1375 psig if a flux scram is assumed. In addition, the generic analyses have been conducted which show an approximate 20 psi sensitivity increase for each relief valve failure.

The analysis of the plant isolation transient (Turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraphs 14.5.1.2 and 14.5.1.3 and is evaluated in each reload analyses. These analyses show that the six relief valves assure greater than 36 psi margin below the setting of the safety valves. Therefore, the safety valves will not open. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to

SAFETY EVALUATION

SAFETY/RELIEF VALVE REPLACEMENT

DUANE ARNOLD ENERGY CENTER

May 1977

1. INTRODUCTION

This document provides a safety analysis for the replacement of six Dresser safety/relief valves with six Target Rock (TR) S/RV. The effect of the 3.6 % reduced capacity of the TR can be compared with the safety analysis presented in reference 4 for the Dresser valves.

The description for TR indicates that the same specifications, codes and standards used for the Dresser valves are applied. Both the Targer Rock and Dresser valves were built to the ASME Code 1968 edition - Winter 1968 addenda. The valve operating characteristics are the same (i.e., time for actuation and valve stroke time). Therefore, the only discernible effect of the replacement on transient and safety analyses will be the change in valve capacity.

The increase in transient pressure and small break LOCA analysis increase in PCT are summarized in section 4.

2.0 DESCRIPTION

The nuclear system pressure relief system includes 2 safety and 6 relief valves all of which are located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. (See Table 2-1). The safety valves provide protection against overpressure of the nuclear system and discharge directly to the interior space of the drywell.

The relief valves, which discharge to the suppression pool, provide three main protection functions:

1. Overpressure relief operation. The valves are opened (self-actuated) to limit the pressure rise and prevent safety valve opening.

2. Manual operation. The valves can be manually opened by the operator.

3. Depressurization operation. The required valves are opened automatically or manually by indirectly operated devices, as part of the Core Standby Cooling System (CSCS), for small breaks in the nuclear system process barrier.

The connections for the new Target Rock safety relief valves will be the same as those used for the old Dresser valves. The inlet connections are 6-inch, 1500-pound special facing flanges. The outlet connections are 10-inch, 300-pound raised face flanges, ANSI B16.5.

The relief valves are designed, constructed, and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, and in accordance with USAS B31.1.0 and B16.5. Popping-point tolerance (pressure at which valve "pops" wide open) is in accordance with ASME Section I, Paragraph PG-72(c). Each valve is self-actuating at the set relieving pressure, but may also be actuated by indirectly operated devices to permit remote manual or automatic opening at lower pressures. For depressurization operation, each relief valve is provided with a power actuated device capable of opening the valve at any steam pressure above 100 psig and capable of holding the valve open until the steam pressure decreases to about 50 psig. The control system

for the actuator is described in subsection 7.4 of the FSAR, "Core Standby Cooling Controls and Instrumentation." Pressure containing parts of the valve body are fabricated of ASTM A216, Grade WCB. The relief valve is designed for operation with saturated steam containing 1% moisture or less. The relieving pressures for overpressure relief are adjustable between 1060 and 1160 psig with a maximum back pressure of 40% of the set pressure. The delay time (maximum elapsed time between overpressure signal and actual valve motion) is equal to or less than 0.4 seconds and the response time (maximum valve stroke time) is equal to or less than 0.1 seconds.

Each relief valve consists of three main sections. The pilot valve section is a relatively small, self-actuated relief valve, integral with the main valve, which provides pressure sensing and main valve control functions. The main element of this pilot valve is a precision-machined spring-bellows, the expansion of which accurately controls the main valve. It is actuated by externally supplied air pressure to a diaphragm. The main valve section is a hydraulically operated, reverse seating globe valve which, when actuated by the pilot valve, provides the pressure relief function by opening to discharge nuclear system steam to the suppression pool.

A typical sequence of operation for overpressure relief self-actuation can be described as follows (refer to Figures 2-1 and 2-2):

1. In the closed position (Figure 2-1), the bellows is mechanically extended a slight amount by the preload spacer to provide a preload force on the pilot disc. This seats the pilot valve tightly and prevents reverse leakage at low system pressures or high back pressures. The main valve disc is tightly seated by the combined forces exerted by the main valve preload spring and the system internal pressure acting over the area of the main valve disc. In the closed position, the static pressures will be equal in the valve body and in the chamber over the main valve piston. This pressure equalization is made possible by leakage through the piston orifice.

2. As system pressure increases, the preload force on the pilot disc is reduced to zero as the bellows is extended farther and the disc is held

closed by the internal pressure acting over the pilot valve seat area. This hydraulic seating force, which is significantly greater than the initial preload, increases with increasing system pressure and discourages leakage or "simmering" at pressures near the valve set pressure.

3. As system pressure further increases, bellows expansion reduces the abutment gap between the stem and the disc yoke. When the stem abuts against the yoke, further pressure increase reduces the net pilot seating force to zero and lifts the first stage pilot valve from its seat.

4. Once the pilot valve starts to open, the hydraulic seating force is eliminated, resulting in a net increase in the force tending to open the pilot valve. This increase in net force produces the "popping" action during pilot valve opening (Figure 2-2).

5. Opening of the first stage pilot valve admits fluid to the operating piston of the second stage valve, causing it also to open.

6. Opening of the second stage pilot valve vents the chamber over the main valve piston to the downstream side of the valve. This venting action creates a differential pressure across the main valve piston almost equal to the system pressure and in a direction tending to open the valve. The main valve piston is sized so that the resulting opening force is greater than the combined preload and hydraulic seating force. Therefore, opening the pilot opens the main valve.

7. As in the case of the pilot valve, once the main valve disc starts to open, the hydraulic seating force is reduced, causing a significant increase in opening force and the characteristic full opening or "popping" action.

8. When the pressure has been reduced sufficiently to permit the pilot valve to close, leakage of system fluid past the main valve piston repressurizes the chamber over the piston, eliminates the hydraulic opening force, and permits the preload spring to close the valve. Once closed, the additional hydraulic seating force due to system pressure acting on the main valve disc seats the main valve tightly and prevents leakage.

The air powered diaphragm operated valve also displaces the second stage piston which in turn controls the main valve as shown in Figures 2-1 and 2-2. Using this system, the relief valve can be remotely opened by supplying pressure on the diaphragm of this actuator.

The relief valves are installed so that each valve discharge is piped through its own uniform diameter discharge line to a point below the minimum water level in the primary containment suppression pool to permit the steam to condense in the pool. Water in the line above suppression pool water level would cause excessive pressure at the relief valve discharge when the valve again opened. For this reason, a vacuum relief valve is provided on each relief valve discharge line to prevent drawing water up into the line due to steam condensation following termination of relief valve operation. The relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity for removing sections of this piping when the reactor head is removed for refueling. In addition, the relief valves, as well as the safety valves, are more accessible during a quick shutdown to correct possible valve malfunctions when located on the steam lines.

Four of the six relief valves are used for automatic depressurization and are equipped with an air accumulator and check valve arrangement. These accumulators are provided to assure that the valves can be held open following failure of the air supply to the accumulators, and they are sized to contain sufficient air for a minimum of five valve operations.

The automatic depressurization feature of the nuclear system pressure relief system serves as a backup to the High Pressure Coolant Injection (HPCI) system under loss of coolant accident conditions. If the HPCI system does not operate and one of the Low Pressure Coolant Injection (LPCI) or core spray pumps is available, the nuclear system is depressurized sufficiently to permit the LPCI and core spray systems to operate to protect the fuel barrier. Depressurization is accomplished through automatic opening of some of the relief valves to vent steam to the suppression pool. For small line breaks when the HPCI system fails, the nuclear system is depressurized in sufficient time to allow the core spray or LPCI systems to provide core cooling to prevent any fuel cladding

melting. For large breaks, the vessel depressurizes rapidly through the break without assistance. The signal for the relief valves to open and remain open is based upon simultaneous signals from: 1) drywell high pressure, 2) reactor vessel low water level, 3) core spray or LPCI pump running.

A manual depressurization of the nuclear system can be effected in the event the main condenser is not available as a heat sink after reactor shutdown. The steam generated by core decay heat is discharged to the suppression pool. The relief valves are operated by remote-manual controls from the main control room to control nuclear system pressure.

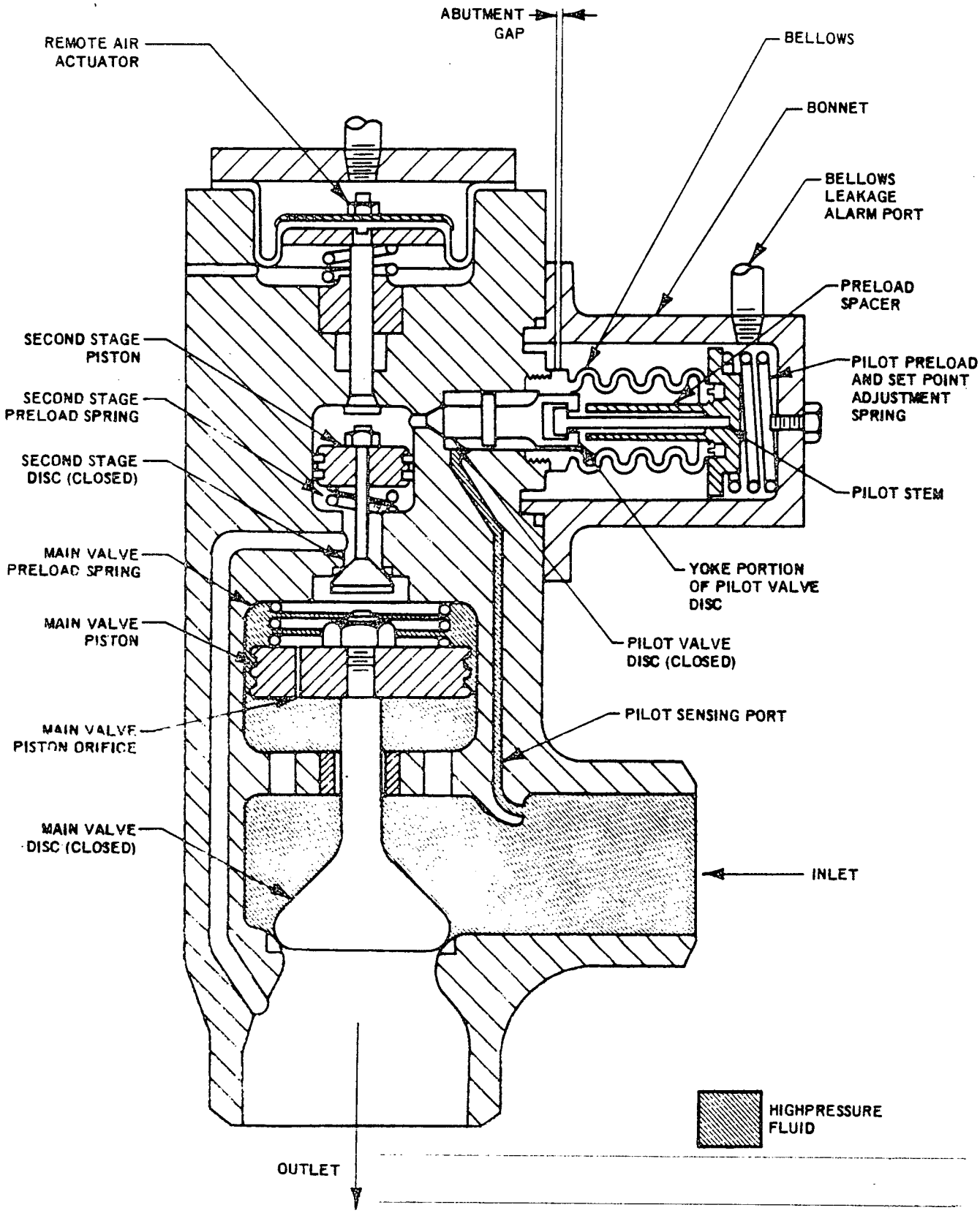
The number, set pressures, and capacities of the relief valves and safety valves are shown in Table 2.1.

TABLE 2-1

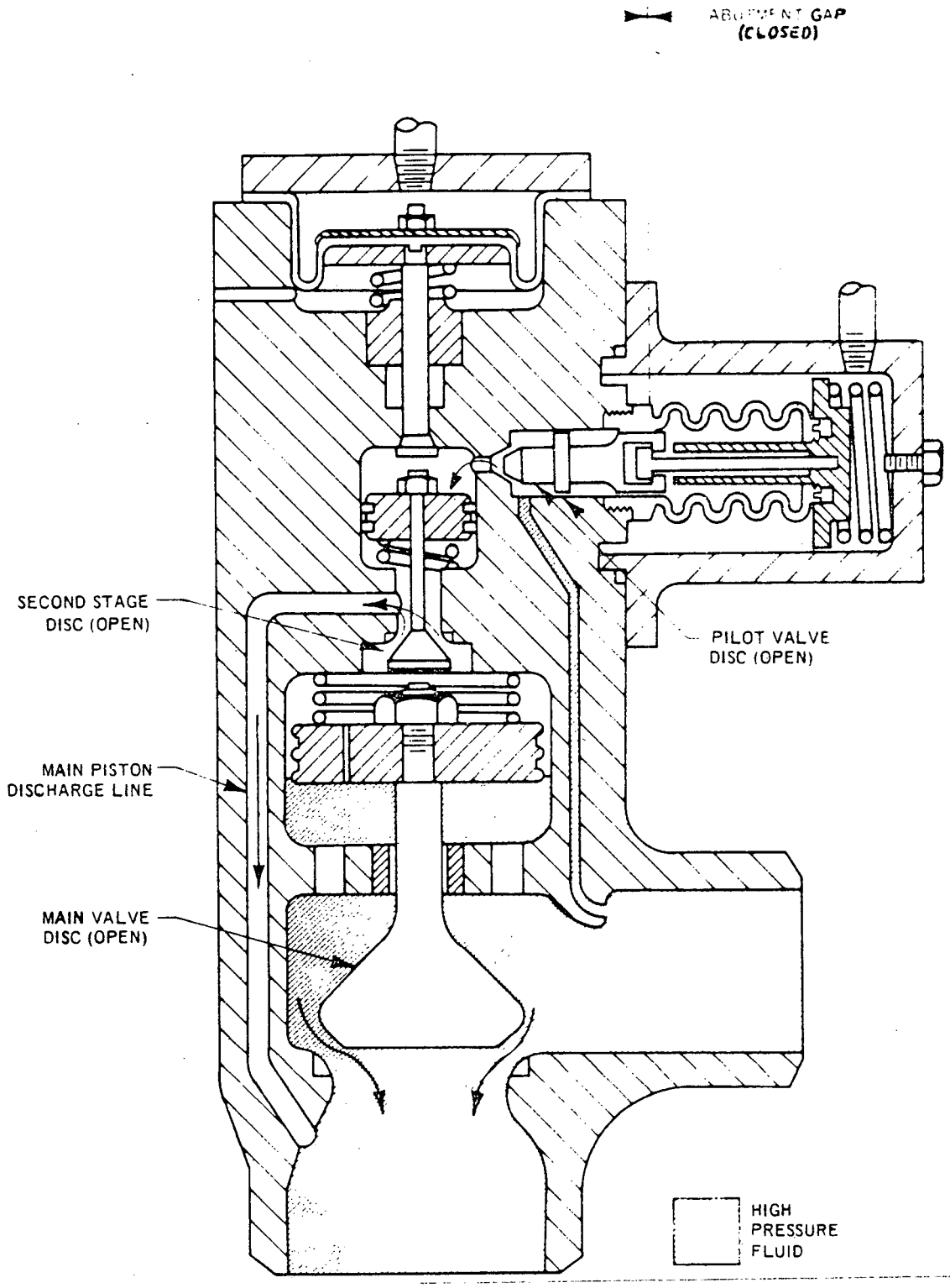
NUCLEAR SYSTEM SAFETY AND RELIEF VALVE

	Number of Valves	Set Pressure (psig)	Capacity at 103% of Set Pressure (each), (lb/hr) Old Dresser Valves	New Target Rock Valves
Relief Valves	2	1090	869,000	838,000
	2	1100	877,010	845,000
	2	1110	884,000	853,000
Total*	6(4)			
Safety Valves	2	1240	642,000	No Change

*The number in parentheses indicates the number of relief valves which serve in the automatic depressurization capacity.



Nuclear System Relief Valve
Closed Position
FIGURE 2-1



Nuclear System Relief Valve
Open Position
FIGURE 2-2

3. SAFETY ANALYSIS

3.1 INTRODUCTION

The safety analysis consists of three categories: (1) generic safety analysis, which is applicable to all reloads; (2) bounding analysis; and (3) specific analysis. Wherever a bounding analysis is applied for an accident or transient, the key parameters need only to be compared with the worst case and, if they are within "bounds," all limits and margins applicable to the accidents or transients will be met.

3.2 MODEL APPLICABILITY TO 8x8 FUEL

Information on the applicability to the 8x8 design of existing models used for safety analyses is given in Reference 1.

3.3 RESULTS OF SAFETY ANALYSES

3.3.1 Core Safety Analyses

The General Electric Thermal Analysis Basis (Reference 3) is used to establish thermal margins in reload cores. The operating limits, margins, and fuel damage limits previously used in reference 4 are applicable to this analysis for the use of target rock safety relief valves.

3.3.2 Accident Analysis

3.3.2.1 Main Steam Line Break Accident

The consequences of the main steam line break analysis depend on the basic thermal-hydraulic parameters of the overall reactor, as discussed in Reference 1.

3.3.2.2 Refueling Accident

The description and analyses of the refueling accident provided in reference 4 are applicable to this analysis for Target Rock safety relief valves.

3.3.2.3 Control Rod Drop Accident

The description of the control rod drop accident provided in reference 4 are applicable to this analysis for Target Rock safety relief valves.

3.3.2.4 Loss-of-Coolant Accident

The analyses given in Reference 1 are applicable to this reload. These analyses were performed for the Reload-2 fuel in accordance with Appendix K of 10CFR Part 50 and are presented in reference 4.

Generic ADS out of service analysis performed in the spring and summer of 1976 indicate that the increase in PCT for small breaks is less than or equal to 130°F for a 10% reduction in the ADS capacity of BWR/3 and BWR/4 plants. Therefore, the effect of 3.6% reduction in the ADS capacity of Duane Arnold can be conservatively bounded as a maximum small break PCT increase of 100°F. to 1673F which provides 527F margin to the limit (2200F).

3.3.2.5 Loading Error Accident

The description and analysis of the loading error accident provided in reference 4 is applicable to this analysis for Target Rock safety relief valves.

3.3.3 <u>Abnormal Operation Transients</u>

3.3.3.1 Transients and Core Dynamics

3.3.3.1.1 Analysis Basis

This subsection contains the analyses of the most limiting abnormal operational transients for Duane Arnold Energy Center Cycle 3. All transients which are the basis of the existing license were reviewed, and those transients which have been limiting in the past with respect to safety margins and are significantly sensitive to the core transient parameter deviations were reanalyzed.

Transient analysis have been performed to obtain an operating limit MCPR from BOC3 to 2 GWD/T before EOC3, 2 GWD/T to 1 GWD/T before EOC3, and 1 GWD/T before EOC3 to EOC3.

Turbine trip without bypass is the most limiting transient when considering the effect of installing Target Rock safety relief valves.

3.3.3.1.2 Input Data and Operating Conditions

The input data and operating conditions are shown in Table 3-1 for these analyses. Each transient is considered at these conditions unless otherwise specified.

3.3.3.1.3 Transient Summary

A summary of the transients analyzed and their consequences is provided in Table 3-2.

TABLE 3-1

TRANSIENT INPUT PARAMETERS

Thermal Power	(MWt)	1657	104%
NBR Steam Flow	(lb/hr)	7.16×10^6	105%
NBR Core Flow	(lb/hr)	49.0×10^6	100%
Dome Pressure	psig	1020	
Turbine Pressure	psig	960	
RV Set Point (nominal/analysis)	psig	1090/1101	
RV/Capacity (at Set Point)	No./%NBR	6/72.0 (Old Value 6/74.7)	
RV Time Delay	(msec)	400	
RV Stroke Time	(Msec)	100	
SV Set Point (nominal/analysis)	psig	1240/1253	
SV/Capacity (at set point)	No./%NBR	2/18.9	

		EOC3	EOC3-1 GWD/T	EOC3-2 GWD/T
Dynamic Void Coefficient	(-c/%Rg)	13.26	14.47	14.40
Doppler Coefficient	(-c/°F)	0.2152	0.2092	0.2016
Average Fuel Temperature	(°F)	1435	1435	1435
Scram Reactivity Curve		Fig 6.6*	Fig 6.6.a*	Fig 6.6.b*
Scram Worth	(-$)	30.16	29.20	28.48

* See reference 4.

TABLE 3-2a

TRANSIENT DATA SUMMARY

DRESSER VALVES

Transient	Power (%)	Core Flow (%)	ϕ (% ref)	Q/A (% ref)	Psl (psig)	Pv (psig)	ΔCPR 8x8	ΔCPR 7x7
Turbine Trip w/o Bypass								
EOC	104	100	439	119	1200	1243	.37	.29
1 GWD/T Before EOC	104	100	465	118	1199	1241	.36	.28
2 GWD/T Before EOC	104	100	371	114	1188	1230	.29	.21

TABLE 3-2b

TRANSIENT DATA SUMMARY

TARGET ROCK VALVES

Transient	Power (%)	Core Flow (%)	ϕ (% ref)	Q/A (% ref)	Psl (psig)	Pv (psig)	ΔCPR 8x8	ΔCPR 7x7
Turbine Trip w/o Bypass								
EOC	104	100	439	119	1204	1246	.37	.29
1 GWD/T Before EOC	104	100	465	118	1202	1244	.36	.28
2 GWD/T Before EOC	104	100	371	114	1191	1233	.29	.21

3.3.3.2 Transient Descriptions

Abnormal operating transients which are limiting according to safety criteria and which also are sensitive to nuclear core parameter changes have been analyzed and are evaluated in the following narrative.

3.3.3.2.1 Turbine Trip With Failure of the Bypass Valves

This transient produces the most severe reactor isolation. The primary characteristic of this transient is a pressure increase due to the obstruction of steam flow by the turbine stop valves. The pressure increase causes a significant void reduction which yields a pronounced positive void reactivity effect. The net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by scram initiated from 90% open switches on the turbine stop valves and by void increase after the safety/relief valves have automatically opened on high pressure. Figure 3-1 illustrates this transient.

The parameters of concern are the peak steamline pressure margin to the first spring safety valve set point and the ΔMCPR for the event. Neutron flux, the precursor of heat flux, and the resulting ΔMCPR which determines the GETAB operating critical power ratio is given in Table 3-2.

The peak streamline pressure is limited to 1204 psig as a result of the high-pressure actuation of the six safety/relief valves which provides a 36 psi margin to the 1240-psig set point of the first spring safety valve.

3.3.3.2.2 Loss of a Feedwater Heater

The description and analysis of the feedwater transient provided in reference 4 are applicable to this analysis for Target Rock safety relief valves.

3.3.3.2.3 Rod Withdrawal Error

The description and analysis of the rod withdrawal error transient provided in reference 4 are applicable to this analysis for Target Rock safety relief valves.

3.3.4 ASME Vessel Pressure Code Compliance

All Main Steamline Isolation Valve Closure-Flux Scram (Safety Valve Adequacy)

The pressure relief system must prevent excessive overpressurization of the primary system process barrier and the pressure vessel to preclude an uncontrolled release of fission products.

The Duane Arnold Energy Center pressure relief system includes six dual function safety/relief valves and two spring safety valves located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. These valves provide the capacity to limit nuclear system overpressurization.

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequences of pressure in excess of the vessel design pressure:

a. A peak allowable pressure of 110% of the vessel design pressure is allowed (1375 psig for a vessel with a design pressure of 1250 psig).

b. The lowest qualified safety valve set point must be at or below vessel design pressure.

c. The highest safety valve set point must not be greater than 105% of vessel design pressure (1313 psig for a 1250 psig vessel).

Duane Arnold Energy Center's safety/relief and spring safety valves are set to selfactuate at the pressures shown in Table 3-1 thereby satisfying b. and c., above. Requirement a. is evaluated by considering the most severe isolation event with indirect scram.

The event which satisfies this specification is the closure of all main steamline isolation valves with indirect (flux) scram. The initial conditions assumed are those specified in Table 3-1. Figure 3-2 graphically illustrates the event for exposure at EOC3. The response indicates a $\geq$ 79 psi margin to the vessel code limit of 1375 for EOC3. Thus, requirement a. is satisfied and adequate overpressure protection is provided by the pressure relief system.

3.3.5 Thermal-Hydraulic Stability Analysis

Descriptions of the types of thermal-hydraulic stability considered and the analytical method used for evaluation are given in Reference 1. The results for Duane Arnold Energy Center previously reported in reference 4 are applicable to this analysis for the installation of target rock safety relief valves.

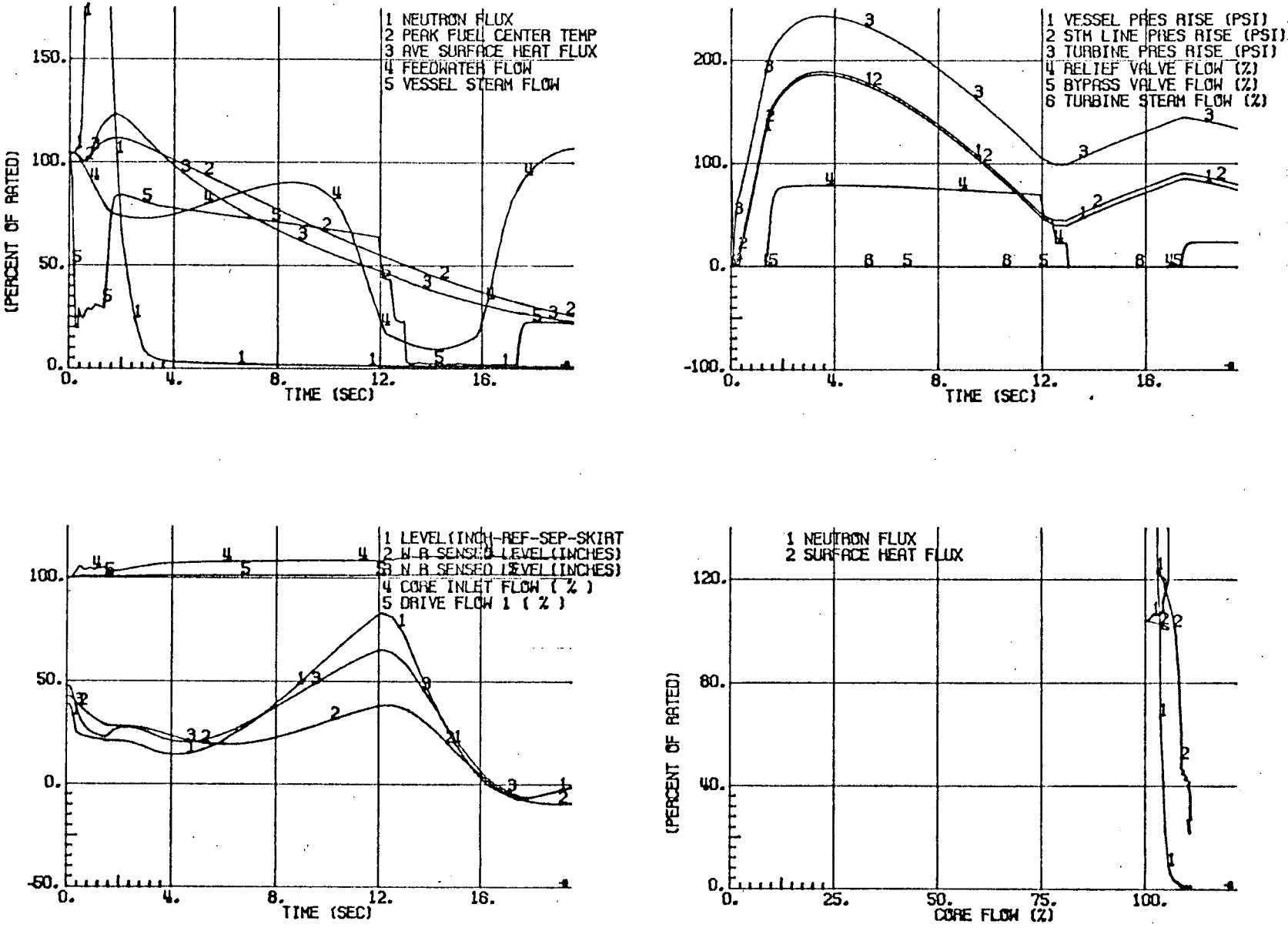


Figure 3.1a DUANE ARNOLD C03 VALVE REPLACEMENT STUDY
TURBINE TRIP W/O BYPASS TRIP SCRAM

EOC3

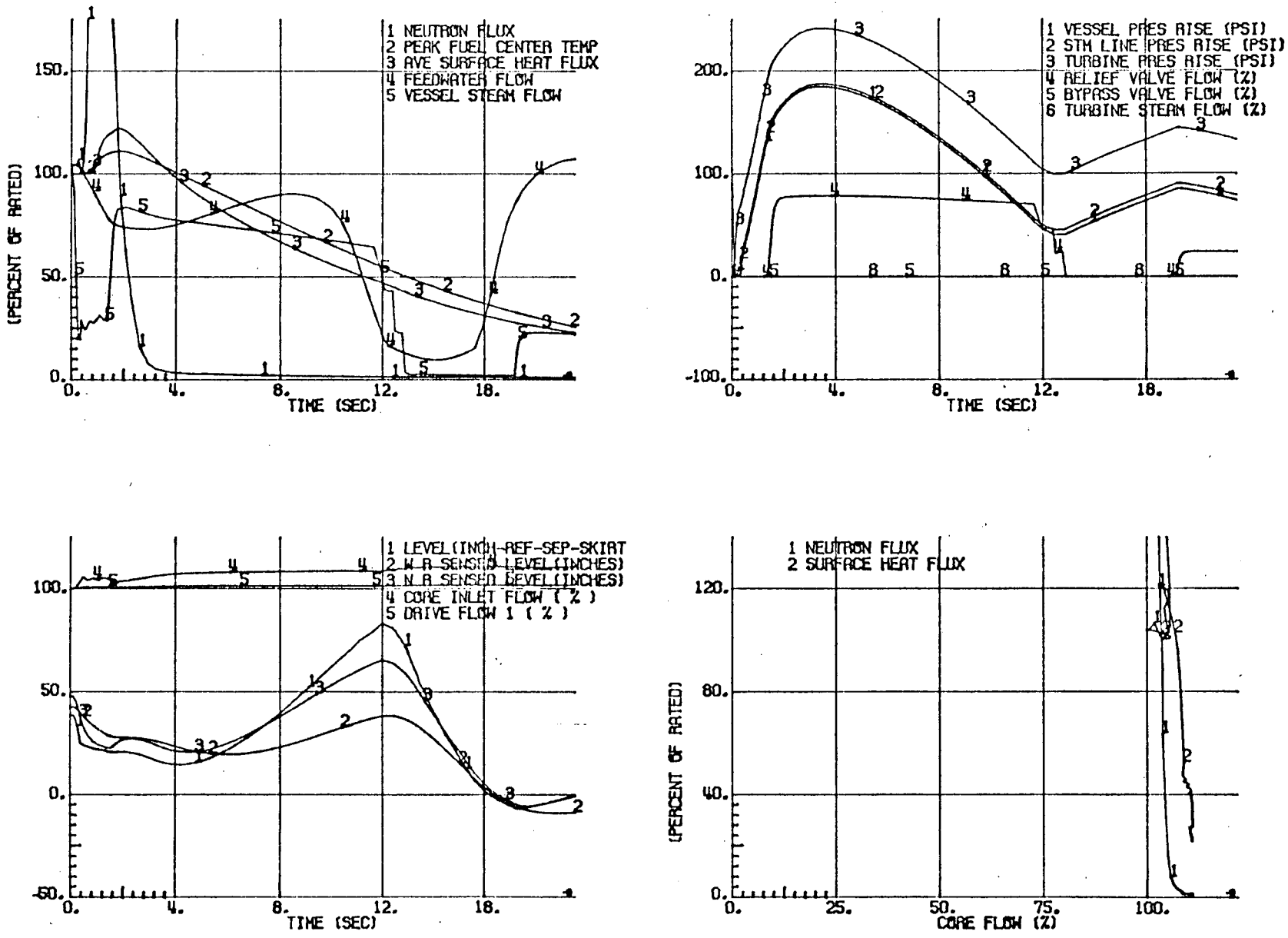


Figure 3.1b DUANE ARNOLD C03 VALVE REPLACEMENT STUDY
TURBINE TRIP W/O BYPASS TRIP SCRAM

E-1K

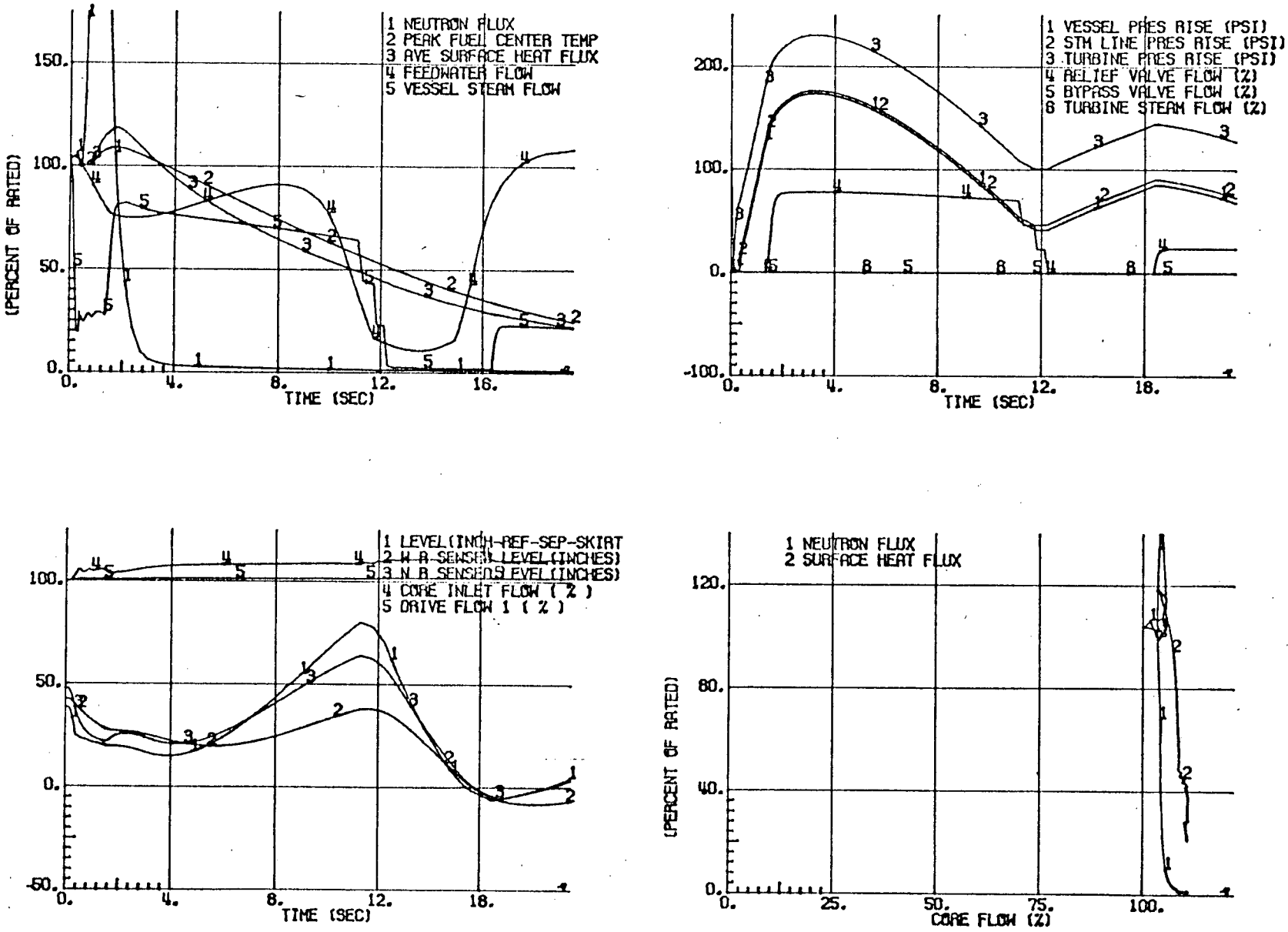


Figure 3.1c DUANE ARNOLD C09 VALVE REPLACEMENT STUDY
TURBINE TRIP W/O BYPASS TRIP SCRAM

E-2K

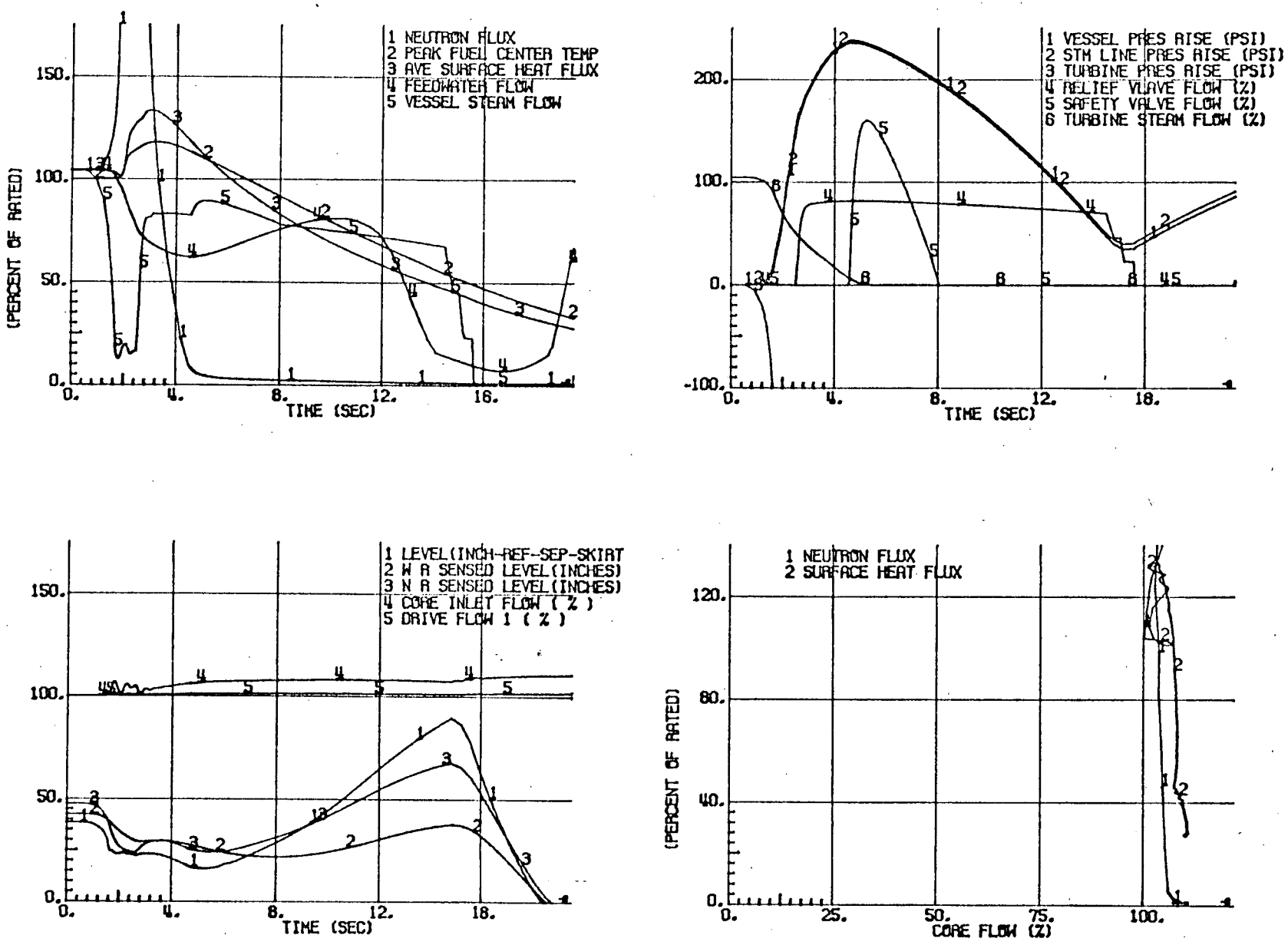


Figure 3-2 DUANE ARNOLD C03 VALVE REPLACEMENT STUDY
MSIV CLOSURE FLUX SCRAM

EOC 9

4. CONCLUSIONS

The safety implications of replacement of the safety/relief valves at Duane Arnold have been evaluated. The only discernable effect on the transient and safety analysis concerns the 3.6 percent change in valve capacity.

The effect of reduced valve capacity on the small break LOCA analysis will be less than a 100 F increase in PCT.

Turbine trip without bypass transient produces the most severe reactor isolation. The peak steamline pressure increased 4 psi as a result of the reduced valve capacity.

Compliance with ASME Boiler and Pressure Vessel Code requirements for overpressure protection was evaluated by analyzing closure of all MSIV with flux scram. The peak vessel bottom pressure increased 5 psi as a result of the reduced valve capacity.

Therefore, it has been concluded that the installation of Target Rock S/RV does not represent an undue risk to the health and safety of the public.

REFERENCES

1. GE/BWR Generic Reload Licensing Application for 8x8 fuel, Rev. 1, Supplement 4 (NEDO-20360) April 1976.

2. Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration, January 1976 (NEDE 21156-Class III).

3. General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application, General Electric Company, BWR Systems Department, November 1973 (NEDE-10958-Class III).

4. GE/BWR Reload Number 2, Licensing Submittal for Duane Arnold Energy Center, NEDO 21082-02, January 1977.