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International Agreement Report

Assessment of a Pressurizer Spray Valve Faulty Opening Transient at Asco Nuclear Power Plant with RELAP5/MOD2

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ABSTRACT

The Asociación Nuclear Ascó has prepared a model of Ascó NPP using RELAP5/MOD2. This model, which include thermalhydraulics, kinetics and protection and controls, has been qualified in previous calculations of several actual plant transients.

One of the transients of the qualification process is a "Pressurizer spray valve faulty opening" presented in this report. It consists in a primary coolant depressurization that causes the reactor trip by overtemperature and later on the actuation of the safety injection.

The results are in close agreement with plant data.

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EXECUTIVE SUMMARY

Ascó Nuclear Power Plant is a nuclear station with two PWR of 930 MWe of Westinghouse design.

The Thermalhydraulic analysis group of the Asociación Nuclear Ascó (ANA) has prepared a model of the plant using RELAP5/MOD2. This model includes thermalhydraulics, kinetics and protection and controls.

ANA's commitment with the International Code Assessment and Application Program (ICAP) is the participation with two cases.

One of the transients selected for this purpose is the "Assessment of a Pressurizer Spray valve faulty opening".

This transient has been chosen because of two main reasons:

- Plant instrumentation data are reasonably accurate for the entire transient.
- The assessment qualifies the behaviour of different systems and components under abnormal conditions (spray valve, safety injection, Reactor protection system).

The main conclusions of the analysis are the following:

- Close agreement between results and data.
- Relap5/mod2 Ascó model is a valuable tool to analyze plant transients.

F O R E W O R D

This report has been prepared by Asociación Nuclear Ascó in the framework of the ICAP-UNESA Project.

The report represents one of the application calculations submitted in fulfilment of the bilateral agreement for cooperation in thermalhydraulic activities between the Consejo de Seguridad Nuclear of Spain (CSN) and the United States Nuclear Regulatory Commission (USNRC) in the form of Spanish contribution to the International Code Assessment and Applications Program (ICAP) of the USNRC whose main purpose is the validation of the TRAC and RELAP system codes.

The Consejo de Seguridad Nuclear has promoted a coordinated Spanish Nuclear Industry effort (ICAP-SPAIN) aiming to satisfy the requirements of this agreement and to improve the quality of the technical support groups at the Spanish Utilities, Spanish Research Establishments, Regulatory Staff and Engineering Companies, for safety purposes.

This ICAP-SPAIN national program includes agreements between CSN and each of the following organizations:

- Unidad Eléctrica (UNESA)
- Unión Iberoamericana de Tecnología Eléctrica (UITESA)
- Empresa Nacional del Uranio (ENUSA)
- TECNATOM
- LOFT-ESPAÑA

The program is executed by 12 working groups and a generic code review group and is coordinated by the "Comité de Coordinación". This committee has approved the distribution of this document for ICAP purposes.

1. INTRODUCTION

In 1986, the Asociación Nuclear Ascó (ANA) created a group for plant and core thermal-hydraulic analysis. The objectives of the group are as follows:

1. Create and update core and plant thermal-hydraulic models based on best-estimate criteria.
2. Provide off-line engineering support to the different technical branches of ANA (i.e., technical services, reactor operation):
 - a. Analyze operating events that result in event reports.
 - b. Assess plant systems and/or equipment modifications as well as plant operating procedures and emergency instructions.
 - c. Analyze plant behavior under incident or accident conditions in the abovementioned cases.
 - d. Scenarios and core damage evaluation for probabilistic risk assessment.
3. Review final safety analysis report transients and accidents based on best-estimate criteria.
4. In the future and if appropriate, participate in Ascó individual plant examination.

The plant analysis activities developed so far include the following:

1. Implementation of RELAP5/MOD2 (Ref.1) cycle 36.05 in its IBM version in an IBM 4381 and 3090 and cycle 36.04 and its Control Data Corporation version in a Cyber 180/830. The results of both versions for the scenarios analyzed are in close agreement.
2. Thermal-hydraulic model of both the primary and secondary systems. /2/

3. Kinetic model specifically adapted to Ascó
4. Simulation of control and protection systems /3/.
5. Revision and detailed study of all start-up tests and every transient that has occurred in either unit. A total of 60 cases were studied. Because of the influence of plant dynamics and the quality and availability of plant data, six cases were selected to validate the complete plant model:
 - a. Station blackout
 - b. Faulty pressurizer spray valve opening.
 - c. Turbine trip without steam dump and secondary relief valves available.
 - d. Loss of feedwater (LOFW)
 - e. Turbine trip with all systems available
 - f. Turbine power step
6. Simulation of the above six transients and adjustment of control parameters /4/, /5/, /6/.
7. Participation in the Internacional Code Assessment and Application Program with two cases.

8. Analysis of transients such as small-break-loss-of-coolant accident (SBLOCA), anticipated transient without scram (ATWS), and others for PRA studies.

The adjustment and qualification process is the first and most important part of plant analysis. Sufficiently accurate predictions with meaningful sets of measured data provide validation of both the model and the procedures to be used in the future to analyze various transient and accident scenarios of general interest such as SBLOCA and ATWS.

2. PLANT AND TRANSIENT DESCRIPTION

2.1 Plant Description

Ascó Nuclear Station is a nuclear power plant with two 930-Mwe Westinghouse Pressurized Water Reactors (PWR). The first criticalities were reached in June 1983 and September 1985, respectively, for units I and II. Today, both units are in their sixth and fourth cycles of normal operation.

The main characteristics of both units are given in Table I. The core contains 157 fuel assemblies of (17x17 -25) fuel rods and the steam generators are typical ones with U-tubes and preheaters (model D-3). All other major components are standard Westinghouse components.

2.2 Plant Data Acquisition System

The plant data used in each assessment calculation is that produced by the plant process computer on the post-trip report. It types a value of each pre-selected variable every 10 seconds. The post-trip report of this particular transient is given in Annex I.

2.3 Transient description

The transient started from 100% rated conditions because of an unnoticed failure on the primary pressure control system.

The failure turned off all the pressurizer heaters and opened the spray valve to its maximum mass flow rate. Immediately the primary pressure started a continuous decrease. The reactor scrammed 117 seconds after the failure of the pressure control system by the overtemperature protection system. The reactor trip further decreases the pressure causing the injection of safety water 10 seconds after reactor trip.

3. MODEL DESCRIPTION

Figure 1 shows the model used to simulate the plant. It consists of 134 volumes, 146 junctions, 32 heat structures, and 259 control variables. The model includes the vessel, the three primary loops, the pressurizer, the three steam generator, the three secondary loops, and the steam lines. The turbine, condenser, and feedwater tank are modeled as time-dependent volumes.

The usual practice of implementing RELAP5 by homogenizing the multiple steam generator loops into two loops (one including the pressurizer) is not followed in this model because of the following reasons:

- a) non - symmetric distribution of auxiliary feed-water among the three steam generators
- b) Different length of the steam lines of each loop.
- c) Different number of plugged tubes in each steam generator.
- d) Non - symmetric transients (loss of feed water, steam generator tube break, small break in the primary circuit, one reactor coolant pump trip, and so on) that require modeling of different actuations or boundary conditions, in each loop.
- e) Computer availability.

3.1 Thermal - Hydraulic Model

a.- Primary System.

The model /2/ of the primary system includes the main components of the plant. The core is modeled by volume 120 and the proper heat structures. Volume 130 simulates the by-pass region between the core baffle and the core barrel. Volume 140 model the upper plenum and volumes 150 and 160 the vessel upper head. The three hot lines depart from the core

pressurizer through volume 510. The pressurizer is divided into two volumes.

The lower one (volume 520) is divided into five nodes. Heat structures, simulating the pressurizer actual heaters, are attached to the first two nodes. Volume 525 is a branch in order to model the junctions connecting the pressurizer with the safety and relief valves and with the spray system.

Volume 420 models the remaining of the hot leg. Volumes 430 and 440 simulate the water boxes of the steam generator. The primary side of the steam generator is modeled by volume 431 divided into nine nodes. Volumes 450, 465, 466 and 470 represent the cold leg. Volume 460 model the primary coolant pump, proper homologous curves, given by the vendor, have been used for this purpose. Volume 468 models the Safety Injection System.

b.- Secondary System

The model of the secondary system starts with Time Dependent Volume 870 (Loop 3) that represent the feed-water going to the steam generator. Volume 871 models the Auxiliary Feed Water. The downcomer is simulated by means of volumes 800, 801, 822 and 825. The steam generator preheater is modeled by volumes 806 and 807, and the remaining of the tubes zone by volumes 808, 806, 809 and 810. The steam separator with volume 820. Volume 830, 840 and 850 model the steam dryer and the dome of the steam generator. The steam line starts at volume 880. Safety and relief valves (components 886 and 884) are connected to volume 881. Component Valve 885 models the isolation valve. Time Dependent Volumes 994 and 999 represent the free atmosphere. The steam is conducted throughout volume 883 to the steam-collector, volume 900. Finally Valves 906, 903 and 907 model the by-pass to condenser valve, and the turbine stop and control valves, respectively.

Proper heat structures are used to connect thermally the primary side of the steam generator with the secondary side. Actual values are used for all the variables except for the hydraulic diameter, heat transfer surface, and thermal conductivity of the tubes material where some changes were introduced in order to achieve the actual heat transfer rate without any change in primary average temperature and secondary pressure.

The data used to model volumes and junctions as well as heat structures were taken from plant design information /7/.

3.2 Kinetic Model

The kinetic model /3/ was prepared using the RELAP5/Mod2 space-independent reactor kinetics option with data from the ANA Nuclear Analysis Group. The model includes a scram table of reactivity versus time. The total control rod drop time is the actual value measured at plant. This table is activated by reactor trip.

The control model supplies the reactivity of the C and D control rod banks.

This control reactivity is added to the feedback reactivities calculated by the kinetic model from the data supplied for the specific burn-up condition of each transient.

3.3 Control and Protection Systems Model

The protection and control systems were modeled using RELAP5/MOD2 control blocks and following specific setpoint studies, logical diagrams and technical specifications of the plant /3/, /8/, /9/. The model includes the following systems:

a.- Reactor Trip System

The reactor can be tripped in this ASCO model because of the following effects:

- Low primary pressure.
- High primary pressure.
- Low speed at any pump.
- High pressurizer level.
- High reactor power.
- Low level at any steam generator.
- Overtemperature.
- Overpower.
- Turbine trip.
- Safety injection.

In Figure 2 the logic of the reactor protection system is presented.

b.- High pressure Injection System

Using the following signals:

- Very low Average Temperature.
- Low steam generator pressure.
- High steam mass flow rate.
- Low primary pressure.
- Large pressure difference between S.G.

the logic of the safety injection system was reproduced. The massflow rate injected is modeled by means of the pumps characteristic curves.

c.- Turbine Trip and Control System

The position of the turbine control valve is controlled as a function of the difference between the Required Power and

Actual Power, with the proper control block to model the actual logic of the plant.

The Turbine Trip (closure of the turbine stop valve) is also modeled. The signals of Safety Injection, very high steam generator level and Reactor trip, are used to trip the turbine.

Turbine run-back has not yet been modeled.

d.- Feed Water Control System

The feed water control system has been modeled as shown in Figure 3. The massflow rate calculated by the control system is injected by means of a time dependent junction.

The auxiliary feed water system is also included in the model.

e.- Pressurizer level and pressure control system

The model of the pressure control system actuates upon heaters, spray valve and charging pump. Pressurizer safety and relief valves and level control system are also simulated.

f.- Steam Dump control system

In Figure 4 the model used for the steam dump control system is represented.

g.- Average Temperature Control System

The average temperature control system modeled with RELAP5/MOD2 is shown in Figure 5. As can be observed this system controls both the primary average temperature and the primary-secondary power mismatch.

Other systems modeled are:

h.- Steam-line Isolation logic.

i.- Main Feedwater Isolation logic.

j.- Safety and Relief valves of the secondary.

4.- STEADY STATE CALCULATION

A steady state calculation was performed with the plant at 100% rated condition. The objective is to obtain a stable condition to start transients.

In Table 3 a comparison between the model results and the plant data is given for the main plant variables. A complete description of the plant sensors and signals is given in ref. 10.

5.- TRANSIENT CALCULATION AND COMPARISON VERSUS ACTUAL DATA

The comparison between model predictions (see table 5) against plant data (see table 4) shows an overall good agreement with some minor disagreements that will be explained later.

Table 6 shows a comparison of the cronology of the main events.

As can be seen in figures 6 to 18 the transient which begun at second 100, starts with an unnoticed opening of the spray valves to its fully open position (figure 13). This causes a continucus decrease in primary pressure (figure 6), reactor power because of the correspondent decrease in moderator density, primary average temperature (figure 7) and secondary pressure (Figure 10). This event produces a decrease in the overtemperature setpoint signal that (see Figure 17) reaches the actual delta-temperature and trips the reactor. The trip takes place in the model at second 216.3 (see Table 6) while in plant it was at second 217, but in both cases for the same reason: overtemperature. The low pressure setpoint of the protection system was close to be reached but it is not the signal that trips the plant.

Figure 13 shows the pressurizer spray mass flow rate accordingly with delta-P between cold leg and pressurizer (see figure 18).

After the trip a further decrease in plant pressure occur, because of the power reduction and some 10 seconds later, at second 236.4, the high Pressure Injection System became active injecting about 25 Kg/s into the primary system (Figure 14) until the end of the transient.

Because of turbine trip the steam-dump valves open, controlled by primary average temperature, (see Figures 10 and 16) lowering secondary pressure and in same seconds steam dump valves close. Afterwards the continucus decrease in secondary pressure is caused by steam extractions.

Control system causes main feed water trip (see Figure 15) and auxiliary feed-water is injected only to steam generators 1 and 3 (because of loop 2 disfunction).

The comparison in the behaviour of the main variables is explained below.

a.- Primary Pressure.

In figure 6 pressure at the hot leg is presented. As can be seen the comparison is good but the pressure decrease predicted by RELAP because of the spray actuation (second 100 to plant trip), has a lower slope than the actual one. It may be caused by a more effective vapor condensation in the plant, caused by the liquid droplet, than in the model.

b.- Primary Temperature

In figure 7 model primary average temperature with and without signal processing is compared versus the plant value. In figure 8 model hot leg temperature of loop 1, without signal processing is compared against plant data.

It can be observed that when signal processing is introduced the comparison is fairly good.

c.- Pressurizer level.

In figure 9 the comparison between model prediction and plant data is given. The overall comparison is fair although a larger increase in plant level is observed both before the reactor trip and when level recovery starts. It may be due to limitations to the actual simulation of the surge line in the model.

d.- Secondary Pressure

In figure 10 the comparison of this variable versus actual data is given. It can be observed that the short term behaviour of the steam dump valves is not well reproduced in the model and a faster decrease of secondary pressure than the actual one is obtained. The long term behaviour is correctly predicted. This problem is being studied also

with other transients. Except for this point a good comparison is obtained.

e.- Steam Generators level.

In figure 11 the comparison between steam generator 1 narrow range level (steam generator 3 behaviour is totally similar since auxiliary feed water is injected at both steam generators) and actual plant data is given.

Figure 12 shows the comparison for steam generator 2 which does not receive auxiliary feed-water in this transient.

In both level predictions some oscillations can be observed when narrow range level decrease under about 30%. Some different nodalizations have been tested before choosing the best one in order to minimize these oscillations. The overall comparisons showed in figures 11 and 12 are good, except for steam generator 2 during the last 40 seconds, where plant data show a quicker decrease that has not yet an explanation.

6.- RUN STATISTICS

Calculations were carried out on a CYBER 180/830 NOS 2.5 property of Fundacion Leonardo Torres Quevedo located at Santander - Spain.

RELAP5/Mod. 2 cycle 36.04 was used in all the calculations.

In Table 8 a Typical run statistics is presented.

7.- CONCLUSIONS

The transient has been simulated with Relap5/mod2 Ascó model.

The results are in close agreement with plant data.

This calculation, along with those of the rest of transients of the qualification matrix, provides the validation of the model.

The function of the pressurizer spray valve is correctly predicted by the model.

The transient provides the assessment of the high pressure Safety Injection, as well as the reactor protection system.

Although in long term, the actuation of the steam-dump seems to be correctly predicted, in short term this actuation can be improved.

Some different steam generator nodalizations have been tested before choosing the best one in order to minimize level oscillations. The level predictions are, any way in fair agreement with actual data.

The model of Ascó using Relap5/mod2 is a valuable tool to analyze plant transients.

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TABLES

TABLE 1. DESCRIPTION OF THE MAIN CHARACTERISTICS OF ASCO I AND II NUCLEAR STATION.

TABLE 2. MAIN EVENTS THAT TOOK PLACE DURING THE TRANSIENT.

TABLE 3. COMPARISON BETWEEN RELAP5/MOD2 VALUES AND ACTUAL DATA FOR STEADY STATE.

TABLE 4. DESCRIPTION OF PLANT DATA MEASUREMENTS.

TABLE 5. DESCRIPTION OF RELAP5/MOD2 VARIABLES.

TABLE 6. COMPARISON OF THE CRONOLOGY OF THE MAIN EVENTS.

TABLE 7. RUN STATISTICS.

- Electrical power	930 Mwe
- Thermal reactor power	2686 Mwth
- Fuel	UO ₂
- Number of assemblies	157
- Fuel rods per fuel assembly	(17x17 - 25) = 264
- Active length of fuel rods	3.657 m
- Outside diameter of fuel rods	4.75×10^{-3} m
- Cladding tube material	Zr -4
- Cladding tube wall thickness	0.655×10^{-3} m.
- Average linear heat generation rate	17.2 Kw/m.
- Absorber rods per control assembly	24
- Absorber material	Ag - In - Cd.
- Number of coolant loops	3
- Reactor operating pressure (pressurizer)	15.51 Mpa.
- Coolant Average Temperature	581.3 °K
- Coolant flow rate	14287 Kg/s

Table 1.- Description of the main characteristics of Ascó I and II
Nuclear Station. (1 of 3)

Steam Generator

- Type	Westinghouse D-3
- Number	3
- Height	20.6 m
- Diameter (Upper shell)	4.445 m
- Tube material	Inconel 600
- Average tube length	15.94 m
- Design pressure/temperature (steam plant side)	8.17 MPa/589 °K
- Inner diameter of tubes	1.687×10^{-2} m.
- Outer diameter of tubes	1.905×10^{-2} m.

Reactor coolant Pumps

- Type	Westinghouse 93-DS
- Discharge head	86.25 m
- Design flow rate	$5.928 \text{ m}^3/\text{s}$
- Speed	155 rad/s

Table 1.- Description of the main characteristics of Ascó I and II
Nuclear Station. (2 of 3)

Pressurizer

- Height	12.835 m
- Diameter (inner)	2.134 m
- Volume	39.64 m ³
- Operating saturation pressure	15.51 MPa
- Heating power of the heater rods	1.40 Mwe

Steam/Power Conversion Plant

- Feed Water flow rate	497.5 Kg/s/loop
- Main steam flow rate	1492 Kg/s
- Steam moisture at steam generator outlet	0.25%
- Feedwater temperature	497.05 °K

Table 1 - Description of the main characteristics of Ascó I and II Nuclear Station. (3 of 3)

Second	Event
0.0	Pressure control system failure.
117.0	Reactor Trip.
117.1	Low pressure signal.
127.0	Safety Injection.

Table 2 - Main events that took place during the transient.

VARIABLE -----	RELAP5/MOD2 -----	PLANT DATA -----
PRIMARY MASS FLOW RATE (Kg/s)	14027.	14287. *
CORE BY-PASS MASS FLOW RATE (%)	2.71	2.71 *
VESSEL DELTA-T (°K)	33.24	33.26
REACTOR POWER (Mw)	2681.	2686.
PRIMARY PRESSURE (MPa)	15.50	15.51
PRIMARY AVERAGE TEMPERATURE (°K)	581.1	581.3
RECIRCULATION RATIO	2.286	2.29 *
UP-STREAM FLOW RATIO IN THE S.G. PREHEATER	.517	.520 *
STEAM GENERATOR NARROW RANGE LEVEL	.66	.66
SECONDARY PRESSURE (MPa)	6.808	6.821
STEAM COLLECTOR PRESSURE (MPa)	6.705	6.724
STEAM MASS FLOW RATE (Kg/s)	1478.5	1492.0

* DESIGN DATA

Table 3 - Comparison between RELAP5/MOD2 values and actual data for steady state.

IDENTIFICATOR -----	DESCRIPTION -----
T0421A	Average Temperature measured in loop 2.
T0424A	Vessel Delta - Temperature measured in loop 2.
F0442A	Measured Mass Flow Rate of loop 3.
P0499A	Pressure at loop 2, hot leg.
P0484A	Pressurizer pressure.
P0419A	Temperature at loop 2, hot leg.
N0482A	Pressurizer level.
P0400A	Secondary pressure, loop 1.
N0400A	Level of the steam generator 1.
N0420A	Level of the seteam generator 2.
N0440A	Level of the steam generator 3.

Table 4 - Description of plant data measurements.

VARIABLE -----	DESCRIPTION -----
P220010000	Pressure at the Hot leg of loop 2.
T220010000	Temperature at the hot leg of loop 2.
P520050000	Pressurizer pressure.
MFLOWJ 100030000	Mass Flow rate of loop 3, cold leg.
CNTRLVAR 5	Average Temperature.
CNTRLVAR 26	Processed Average Temperature.
CNTRLVAR 6	Vessel Delta-T (Hot minus cold leg Temperature).
CNTRLVAR 25	Processed Vessel Delta-T.
CNTRLVAR 13	Pressurizer level.
RKTPOW 0	Nuclear Reactor Power.
MFLOWJ 105	Mass Flow Rate at the inlet of the lower Plenum.
MFJ10001	Mass Flow Rate of loop 1, cold leg.
MFJ10002	Mass Flow Rate of loop 2, cold leg.
P681010000	Secondary pressure at the steam line of loop 1.
MFJ906	Steam-Dump Mass Flow Rate.

Table 5 - Description of RELAP5/MOD2 variables. (1 of 2)

MFJ675	Main Feed Water Mass Flow Rate of Loop 1.
MFJ672	Auxiliary Feed Water Mass Flow Rate of Loop 1.
MFJ772	Auxiliary Feed Water Mass Flow Rate of Loop 2.
CNTRLVAR 16	Steam Generator 1 narrow range water level.
CNTRLVAR 17	Steam Generator 2 narrow range water level.
CNTRLVAR 18	Steam Generator 3 narrow range water level.
MFLJ535	Pressurizer Spray Mass Flow Rate.
MFLJ367	Safety Injection Mass Flow Rate.
CNTRLVAR 147	Main Feed Water Mass Flow Rate of Loop 1.
CNTRLVAR 36	Overtemperature Setpoint.
CNTRLVAR 47	Processed Vessel Delta-T (Reactor Protection System)
P36501	Cold leg Pressure

Table 5 - Description of RELAP5/MOD2 variables. (2 of 2)

Event	Time (s)	
	Plant	Relap
Pressure control system failure	100.0	100.0
Reactor trip (overtemperature)	217.0	216.3
Steam-Dump starts to open	--	216.5
Low pressure signal	217.1	218.0
Steam-Dump fully open	--	223.5
Safety Injection	227.0	236.3
Low average temperature signal	--	236.7

Table 6 - Comparison of the cronology of the main events.

COMPUTER ASSESSMENT	CYBER 180/830
-----	-----
TRANSIENT TIME (s)	290
CPUTIME (s)	37460
C (total number of actives volumes in the model)	173
DT (Total number of time steps)	7600
CPU x 1000	
-----	28.491
C x DT	
CPUTIME/TRANSIENT TIME	129.17

Table 7 - Run Statistics.

FIGURES

- FIGURE 1. NODALIZATION DIAGRAM OF ASCO NUCLEAR POWER PLANT.
- FIGURE 2. LOGIC OF THE REACTOR PROTECTION SYSTEM.
- FIGURE 3. FEEDWATER CONTROL SYSTEM.
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- FIGURE 17. VESSEL DELTA TEMPERATURE.
- FIGURE 18. PRIMARY PRESSURE.

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NODALIZATION DIAGRAM OF ASCO NUCLEAR POWER PLANT

JULY 1980

FIGURE 2. -LOGIC OF THE REACTOR PROTECTION SYSTEM

FIGURE 3. FEED-WATER CONTROL SYSTEM

FIGURE. 5 - AVERAGE TEMPERATURE CONTROL SYSTEM

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 6
— P520050000
+ P0484A

ANA

PRESS. SPRAY VALVE OPENING

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 8
— TEMPF 22001
+ T0419A

ANA

PRESS. SPRAY VALVE OPENING

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 10
— P681010000 TIME (sec) + P0400A

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 11
CNTRLVAR 16
TIME (sec)
+ NO400A

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 12
CNTRLVAR 17
TIME (sec)
+ NO420A

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 13 TIME (sec)
—— MFLOWJ535

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 14 TIME (sec)

— MFLOWJ 36700

ANA

PRESS. SPRAY VALVE OPENING

ANA

PRESS. SPRAY VALVE OPENING

FIGURE. 16

ANA

PRESS. SPRAY VALVE OPENING

ANA

PRESS. SPRAY VALVE OPENING

FIGURE 18
P52005

TIME (sec)
P36501

END

DATE

FILMED

3/29/94