# Entergy

# Nuclear Performance Report

Entergy

# Nuclear Performance Report

MARCH 1991

# E N C L O S U R E S

## ENTERGY OPERATIONS PERFORMANCE REPORT

## MARCH, 1991

- Performance Reports
  - Consolidated for ANO, GGNS, and Waterford-3
  - Plant Status Reports
  - Licensee Event Reports
  - NRC Violations
  - Plant Specific
    - ANO
    - GGNS
    - Waterford-3

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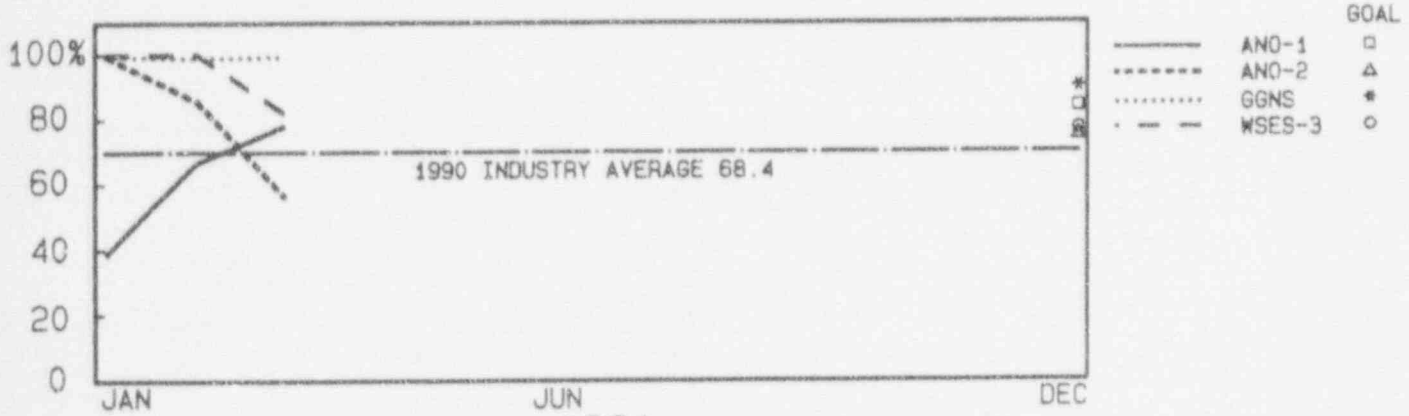
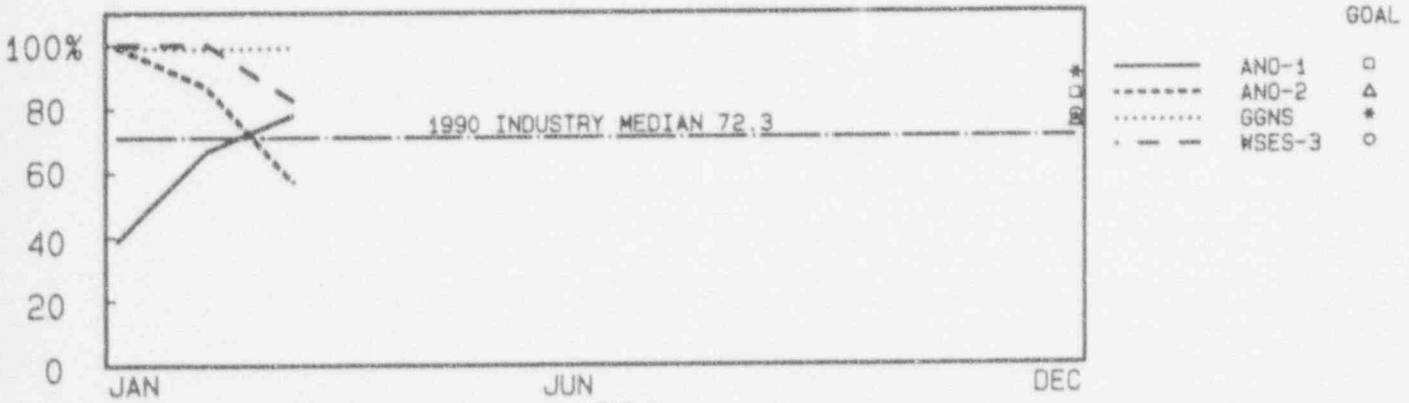
# ENTERGY OPERATIONS, INC.

## Unit Equivalent Availability
### Year-To-Date Average




1991

## Unit Capability Factor
### Year-To-Date Average




1991

### Unit Equivalent Availability

| | CURRENT MONTH | YTD | GOAL |
|---|---|---|---|
| ANO-1 | 99.84 | 78.28 | ≥84.2% |
| ANO-2 | 0.00 | 56.39 | ≥75.0% |
| GGNS | 99.95 | 99.38 | ≥90.0% |
| WSES-3 | 47.27 | 81.79 | ≥77.0% |

3 YEAR INDUSTRY BEST QUARTILE 77.8%

3 YEAR INDUSTRY MEDIAN 71.0%

UNIT EQUIVALENT AVAILABILITY MEASURES THE FRACTION OF A PERIOD THAT A UNIT WAS AVAILABLE FOR RATED SERVICE.

HIGH EQUIVALENT AVAILABILITY INDICATES POSITIVE PERFORMANCE.

### Unit Capability Factor

| | CURRENT MONTH | YTD | GOAL |
|---|---|---|---|
| ANO-1 | 99.82 | 78.21 | ≥84.2% |
| ANO-2 | 0.0 | 56.86 | ≥75.0% |
| GGNS | 99.9 | 99.0 | ≥90.0% |
| W3-SES | 48.3 | 82.2 | ≥77.0% |

1990 INDUSTRY MEDIAN 72.3%
1990 INDUSTRY BEST QUARTILE 82.9%

UNIT CAPABILITY FACTOR IS THE RATIO OF AVAILABLE GENERATION TO MDC GENERATION. ENERGY LOSSES NOT UNDER PLANT CONTROL ARE NOT CONSIDERED.

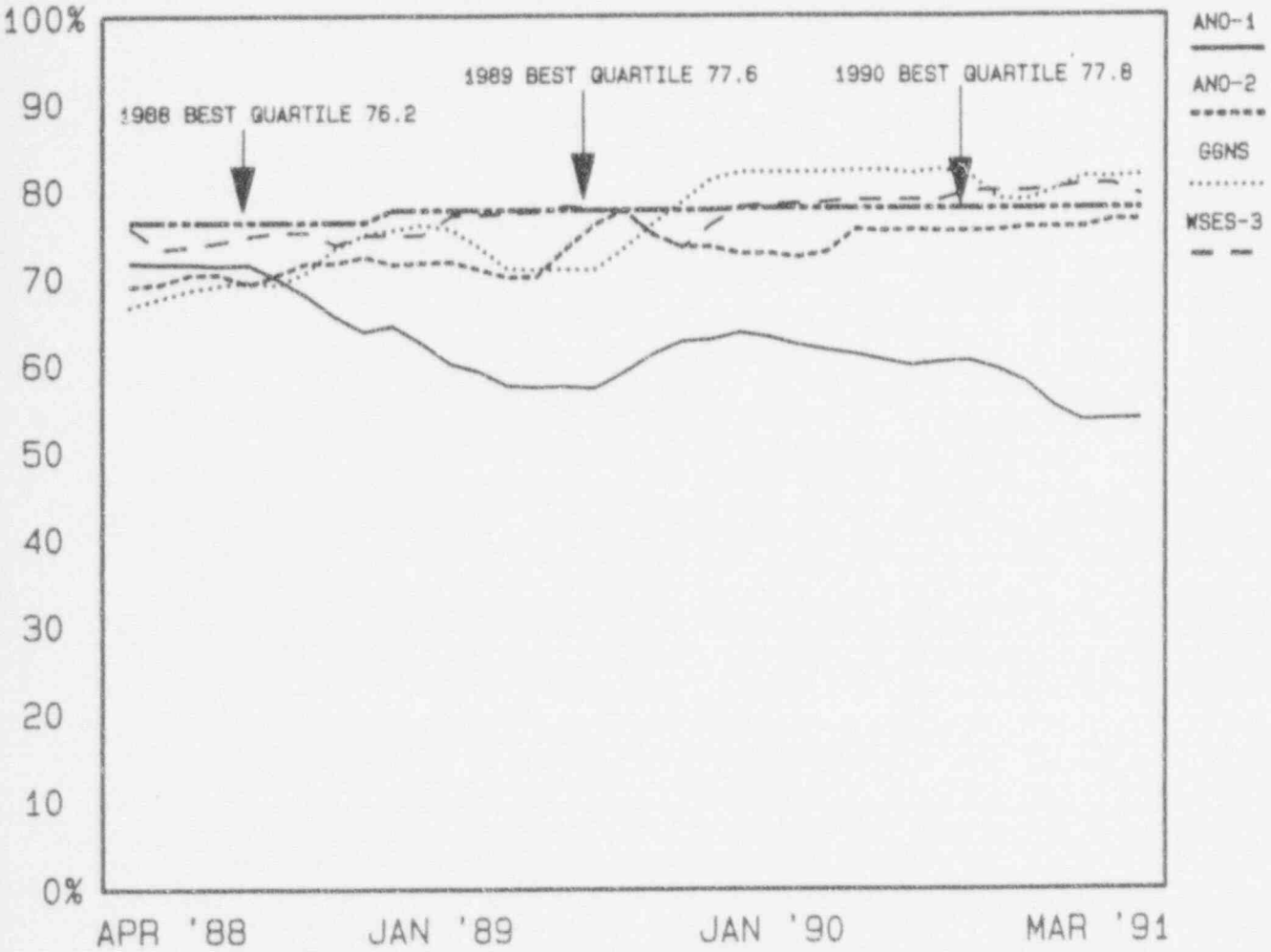
HIGH UNIT CAPABILITY FACTOR INDICATES POSITIVE PERFORMANCE.

GRAPH 1

# ENTERGY OPERATIONS, INC.

# Unit Equivalent Availability

## Three Year Moving Average




| | THREE YEAR AVERAGE |
|---|---|
| ANO-1 | 53.85 |
| ANO-2 | 76.58 |
| GGNS | 81.71 |
| WSES-3 | 79.43 |

UNIT EQUIVALENT AVAILABILITY MEASURES THE FRACTION OF A PERIOD THAT A UNIT WAS AVAILABLE FOR RATED GENERATION SERVICE.

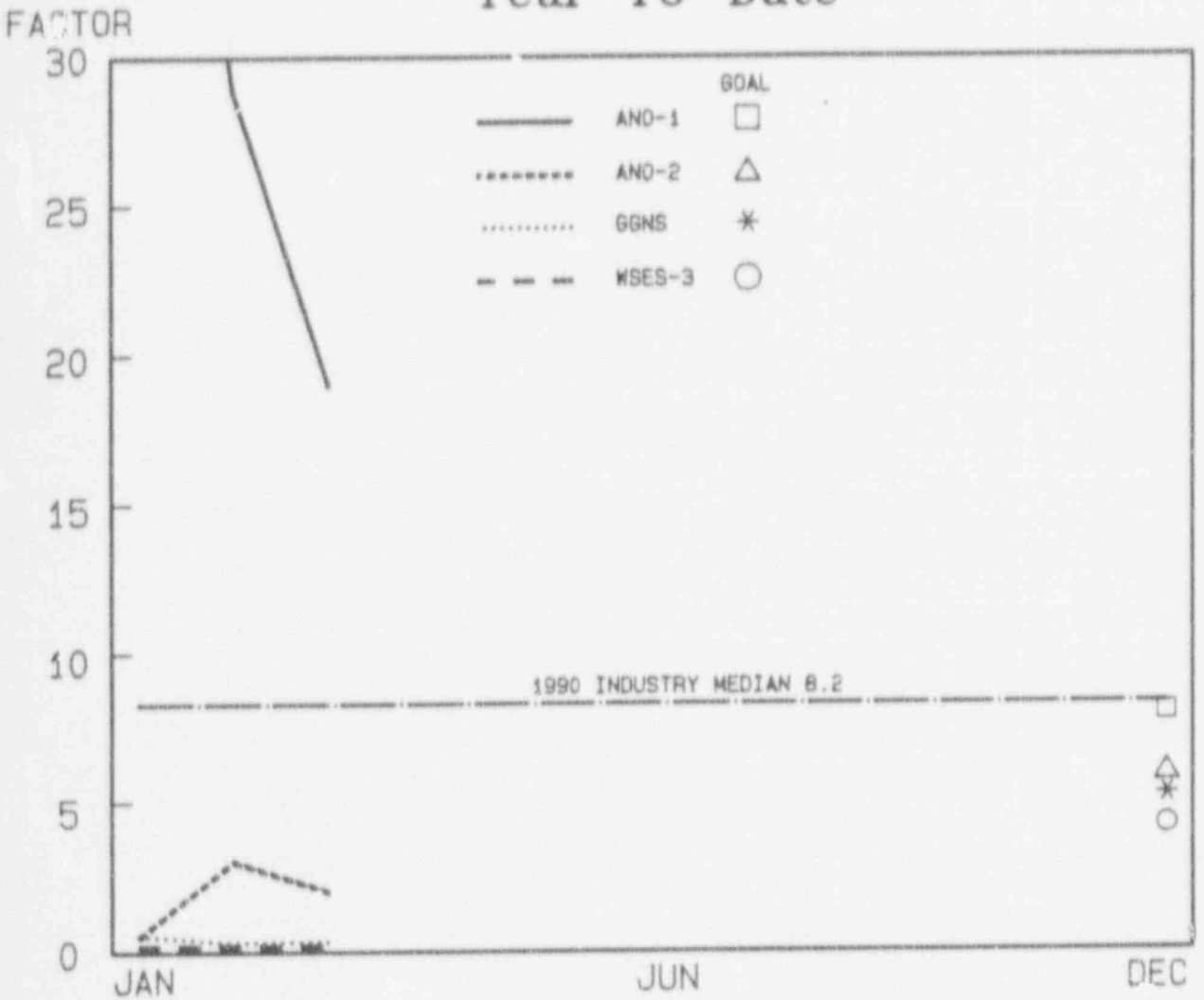
HIGH EQUIVALENT AVAILABILITY INDICATES POSITIVE PERFORMANCE.

GRAPH 2

# ENTERGY OPERATIONS, INC.

## Unplanned Capability Loss Factor

### Year-To-Date




| | CURRENT MONTH | YTD | GOAL |
|---|---|---|---|
| ANO-1 | 0.14 | 16.98 | ≤8.0 |
| ANO-2 | 0.0 | 1.99 | ≤5.9 |
| GGNS | 0.0 | 0.20 | ≤5.2 |
| WSES-3 | 0.0 | 0.0 | ≤4.2 |

1990 INDUSTRY MEDIAN 8.2
1990 INDUSTRY BEST QUARTILE 4.4

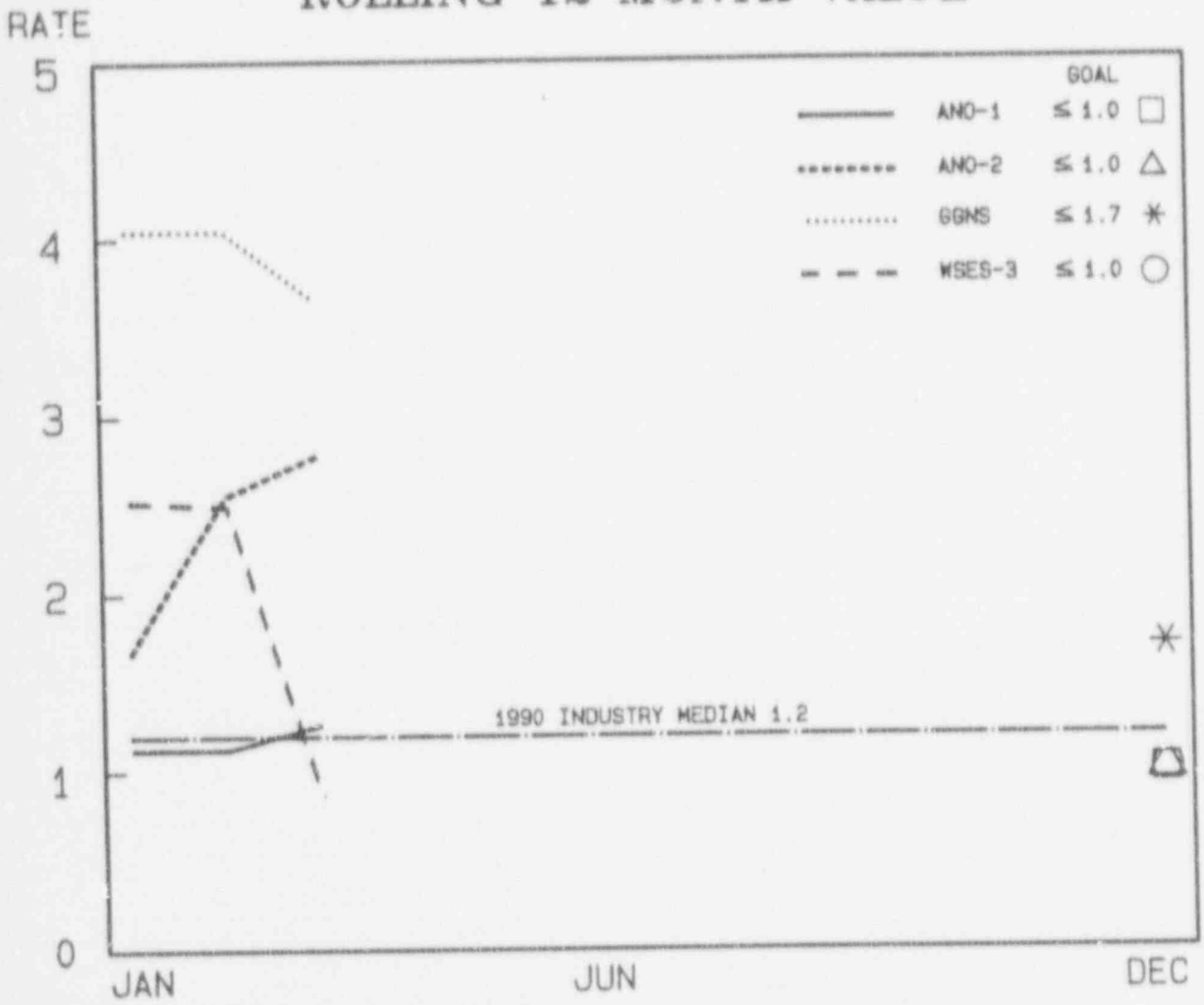
LOW UNPLANNED CAPABILITY LOSS FACTOR INDICATES POSITIVE PERFORMANCE.

GRAPH 3

# ENTERGY OPERATIONS, INC.

## Scrams Per 7000 Critical Hours

## ROLLING 12 MONTH VALUE




| | NUMBER OF SCRAMS CURRENT MONTH | SCRAMS Y-T-D | ROLLING 12 MONTH VALUE |
|---|---|---|---|
| ANO-1 | 0 | 1 | 1.2621 |
| ANO-2 | 0 | 1 | 2.7809 |
| GGNS | 0 | 0 | 3.65 |
| WSES-3 | 0 | 0 | 0.86 |

3 YEAR (88-90) INDUSTRY MEDIAN 1.9
3 YEAR (88-90) INDUSTRY BEST QUARTILE 1.2

1990 INDUSTRY MEDIAN 1.2

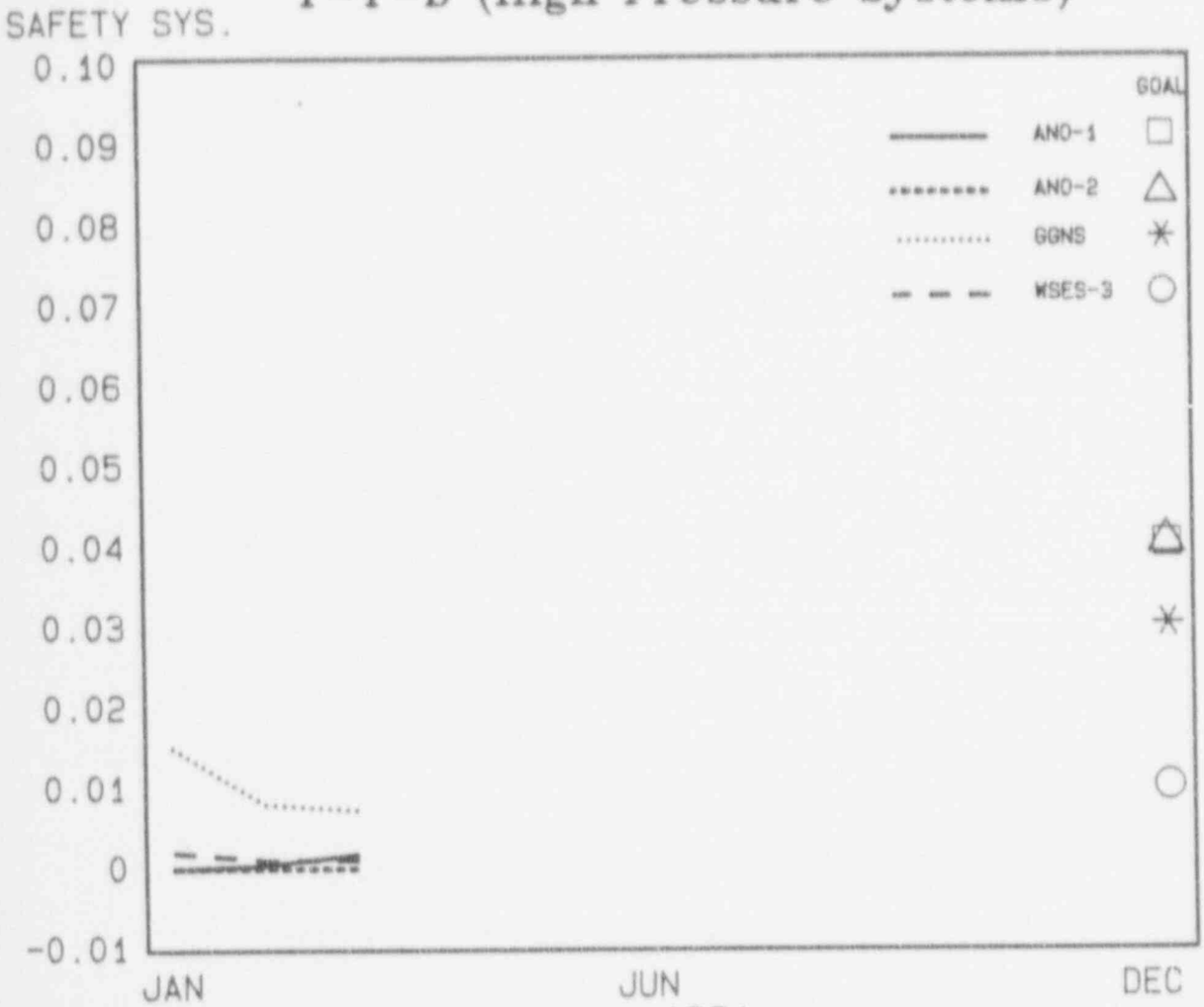
LOW NUMBERS OF SCRAMS INDICATES POSITIVE PERFORMANCE.

GRAPH 4

ENTERGY OPERATIONS, INC.

# Safety System Performance

## Y-T-D (High Pressure Systems)




| | Y-T-D | GOAL |
|---|---|---|
| ANO-1 | 0.0018 | ≤ 0.04 |
| ANO-2 | 0.0 | ≤ 0.04 |
| GGNS | 0.0073 | ≤ 0.03 |
| WSES-3 | 0.001 | ≤ 0.01 |

1990 INDUSTRY MEDIAN (BWR/PWR) 0.015 / 0.005
3 YR. (88-90) INDUSTRY MEDIAN (BWR/PWR) 0.016 / 0.008

SAFETY SYSTEM PERFORMANCE MEASURES THE READINESS OF IMPORTANT SAFETY SYSTEMS TO RESPOND TO OFF-NORMAL EVENTS OR ACCIDENTS.

A LOW SAFETY SYSTEM PERFORMANCE NUMBER INDICATES POSITIVE PERFORMANCE.

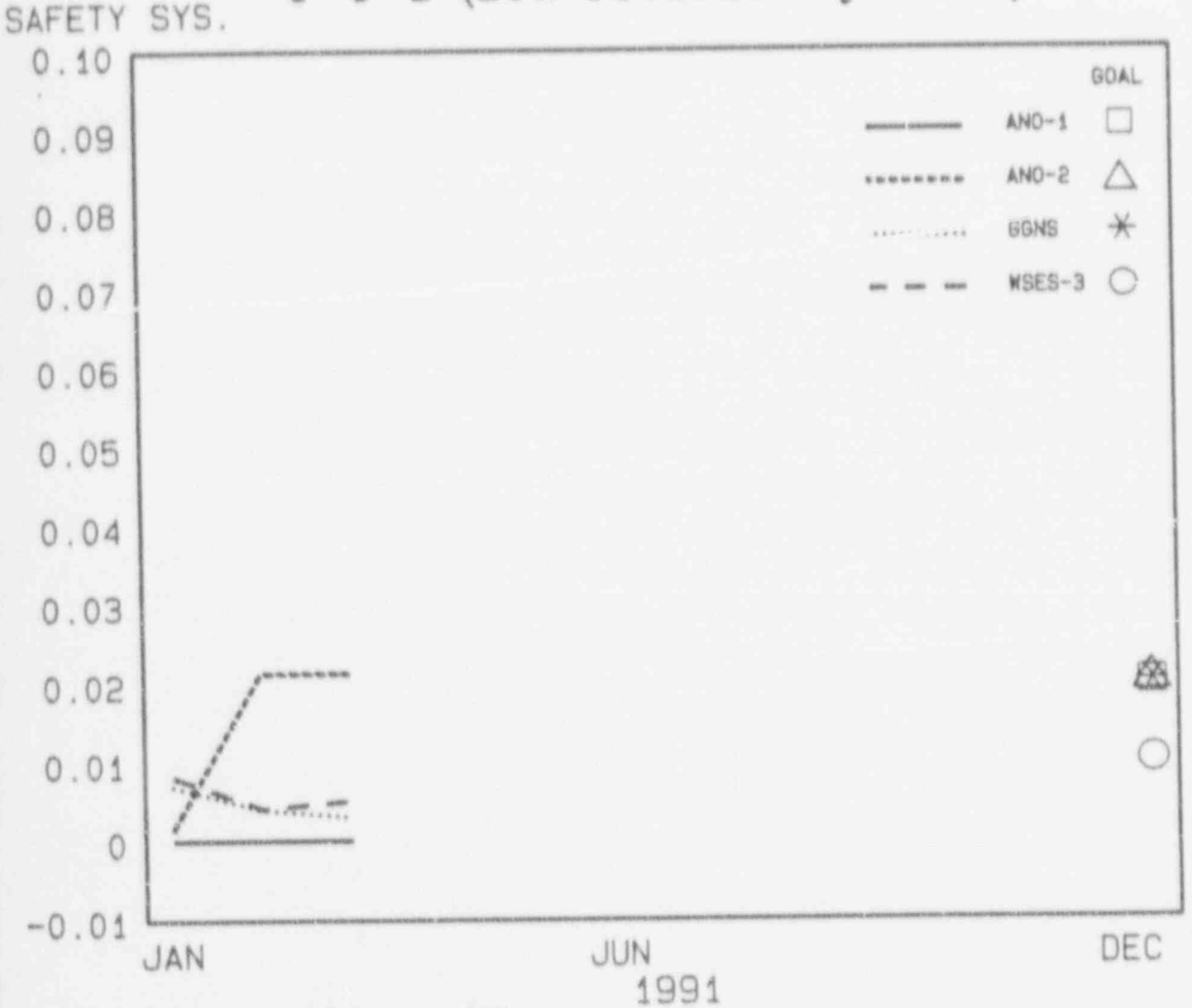
NOTE: INPO DOES NOT PROVIDE BEST QUARTILE VALUES. A 3 YEAR MEDIAN WAS PROVIDED INSTEAD.

GRAPH 5

ENTERGY OPERATIONS, INC.

# Safety System Performance

## Y-T-D (Low Pressure Systems)




| | Y-T-D | GOAL |
|---|---|---|
| ANO-1 | 0.0001 | ≤ 0.02 |
| ANO-2 | 0.0212 | ≤ 0.02 |
| GGNS | 0.0031 | ≤ 0.02 |
| WSES-3 | 0.005 | ≤ 0.01 |

1990 INDUSTRY MEDIAN (BWR/PWR) 0.008 / 0.010
3 YEAR (88-90) INDUSTRY MEDIAN (BWR/PWR) 0.005 / 0.013

SAFETY SYSTEM PERFORMANCE MEASURES THE READINESS OF IMPORTANT SAFETY SYSTEMS TO RESPOND TO OFF-NORMAL EVENTS OR ACCIDENTS.

A LOW SAFETY SYSTEM PERFORMANCE NUMBER INDICATES POSITIVE PERFORMANCE.

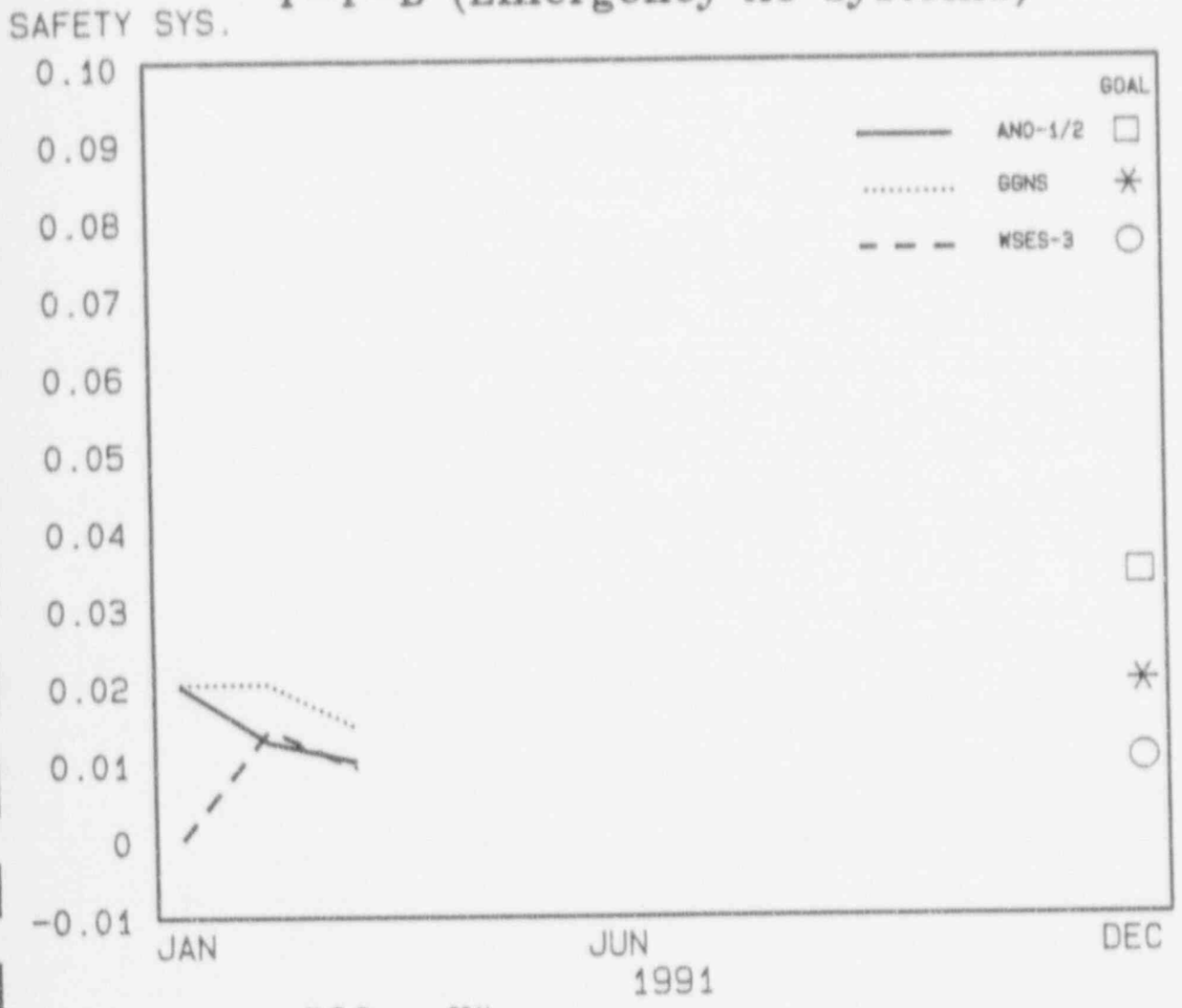
NOTE: INPO DOES NOT PROVIDE BEST QUARTILE VALUE. A 3 YEAR MEDIAN WAS PROVIDED INSTEAD.

GRAPH 6

# ENTERGY OPERATIONS, INC.

## Safety System Performance

### Y-T-D (Emergency AC Systems)




| | Y-T-D | GOAL |
|---|---|---|
| ANO-1/2 | 0.0100 | ≤ 0.034** |
| GGNS | 0.0145 | ≤ 0.02 |
| WSES-3 | 0.009 | ≤ 0.01 |

1990 INDUSTRY MEDIAN 0.015
3 YEAR (1988-1990) INDUSTRY MEDIAN 0.017

SAFETY SYSTEM PERFORMANCE MEASURES THE READINESS OF IMPORTANT SAFETY SYSTEMS TO RESPOND TO OFF-NORMAL EVENTS OR ACCIDENTS.

A LOW SAFETY SYSTEM PERFORMANCE NUMBER INDICATES POSITIVE PERFORMANCE.

** SITE VALUE FOR ANO; INPO CALCULATES SITE VALUES FOR MULTI-UNIT SITES.

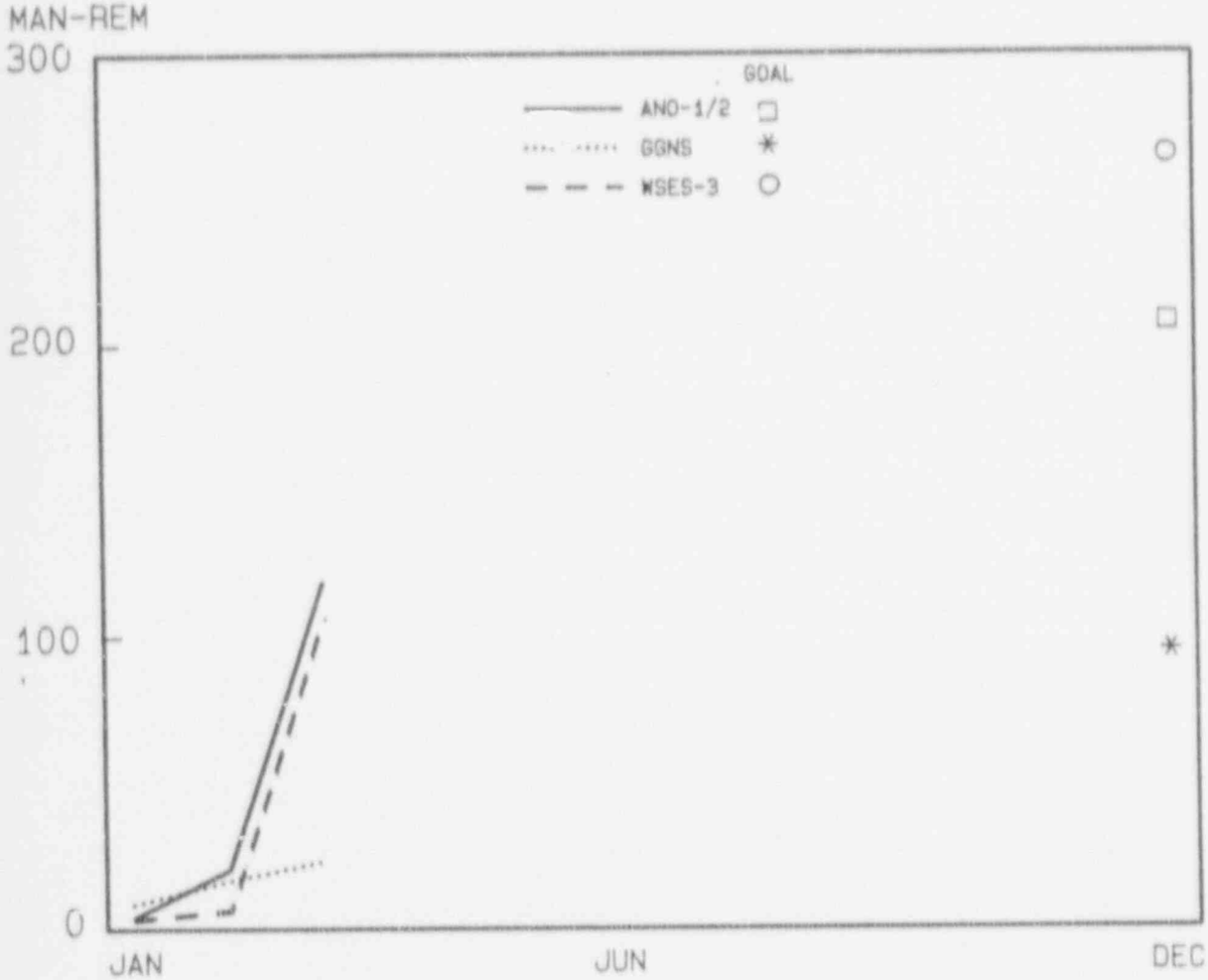
NOTE: INPO DOES NOT PROVIDE BEST QUARTILE VALUE. A 3 YEAR MEDIAN WAS PROVIDED INSTEAD.

GRAPH 7

ENTERGY OPERATIONS, INC.

# Radiation Dose Per Unit

## Year-To-Date




| | CURRENT MONTH | Y-T-D | GOAL |
|---|---|---|---|
| ANO-1/2 (PER UNIT) | 98.651* | 119.135* | ≤208 |
| GGNS | 6.668 | 23.401 | ≤ 95 |
| WSES-3 | 125.8* | 106.7** | ≤265 |

* SRD DATA
** YTD CALCULATED AS 80% OF TOTAL MONTHLY SRD VALUES
NOTE: TLD VALUES REPLACE SRD READINGS EACH QUARTER

| | PWR | BWR |
|---|---|---|
| 1990 INDUSTRY AVERAGE | 294 | 436 |
| 3 YEAR (88-90) IND. BEST QUARTILE | 219 | 277 |
| 3 YEAR (88-90) IND. MEDIAN | 284 | 460 |

TOTAL RADIATION DOSE MEASURES THE EXPOSURE OF BADGED PERSONNEL, CONTRACTORS, OR VISITORS TO RADIATION.

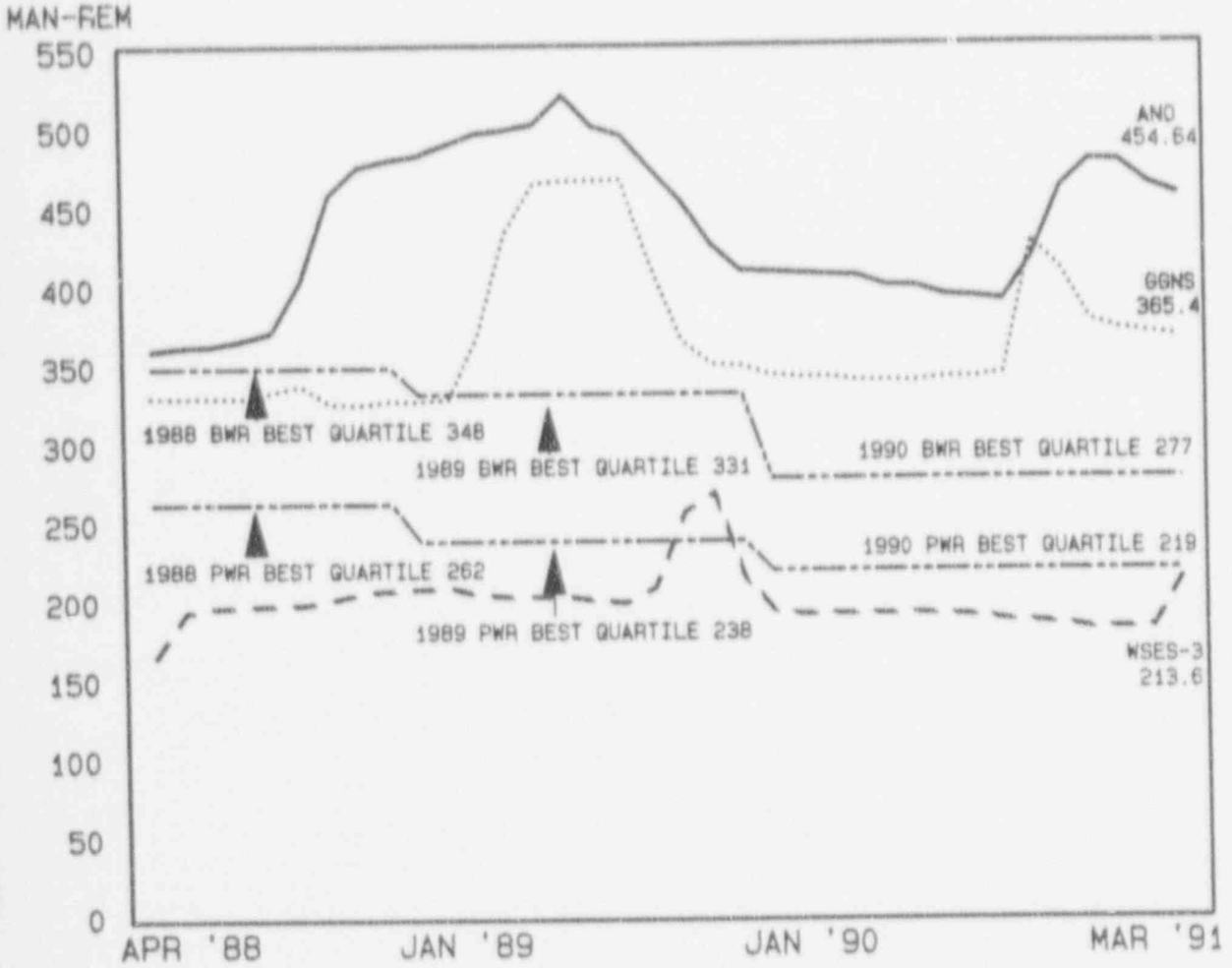
LOW TOTAL RADIATION DOSE INDICATES POSITIVE PERFORMANCE.

GRAPH 8

# ENTERGY OPERATIONS, INC.

## Radiation Dose Per Unit

### Three Year Moving Average




| | THREE YEAR AVERAGE |
|---|---|
| ANO-1/2 (PER UNIT) | 454.64 |
| GGNS | 365.4 |
| WSES-3 | 213.6 |

| | PWR | BWR |
|---|---|---|
| 1989 INDUSTRY AVERAGE | 294 | 436 |
| 3 YEAR (88-90) IND. BEST QUARTILE | 219 | 277 |
| 3 YEAR (88-90) IND. MEDIAN | 284 | 460 |

TOTAL RADIATION DOSE MEASURES THE EXPOSURE OF BADGED PERSONNEL, CONTRACTORS, OR VISITORS TO RADIATION.

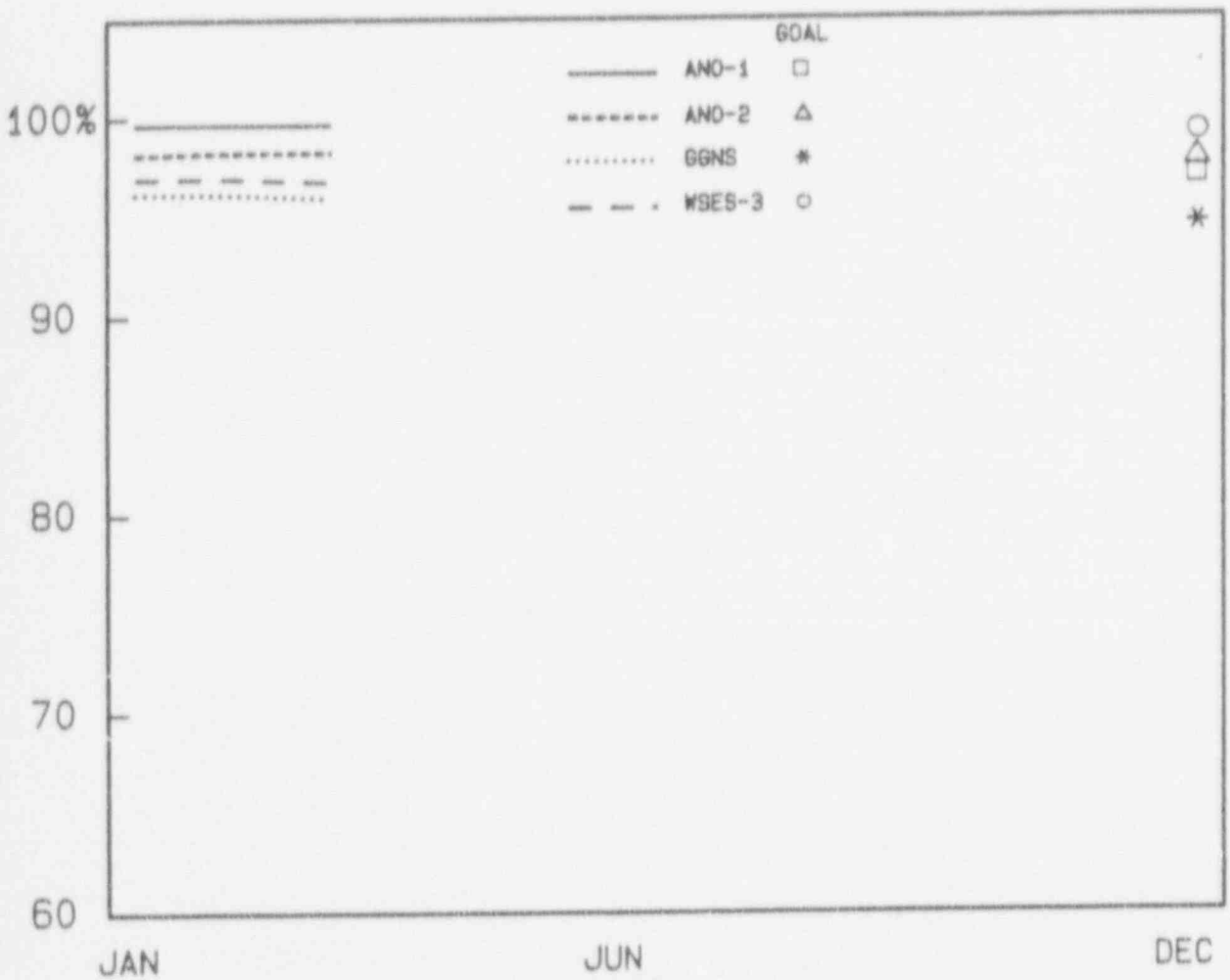
LOW TOTAL RADIATION DOSE INDICATES POSITIVE PERFORMANCE.

GRAPH 9

# ENTERGY OPERATIONS, INC.

# Thermal Performance Index

## Year-To-Date




| | MONTHLY AVERAGE | Y-T-D | GOAL |
|---|---|---|---|
| ANO-1 | 99.68 | 99.7 | ≥97.0 |
| ANO-2 | * | 98.3 | ≥97.8 |
| GGNS | 95.6 | 96.0 | ≥94.6 |
| WSES-3 | 96.5 | 96.8 | ≥99.2 |

* ANO-2 IN REFUELING OUTAGE

1990 INDUSTRY MEDIAN 98.8
1990 INDUSTRY BEST QUARTILE 99.4

THERMAL PERFORMANCE INDEX (%) IS A RATIO OF DESIGN GROSS HEAT RATE TO THE ADJUSTED ACTUAL GROSS HEAT RATE.

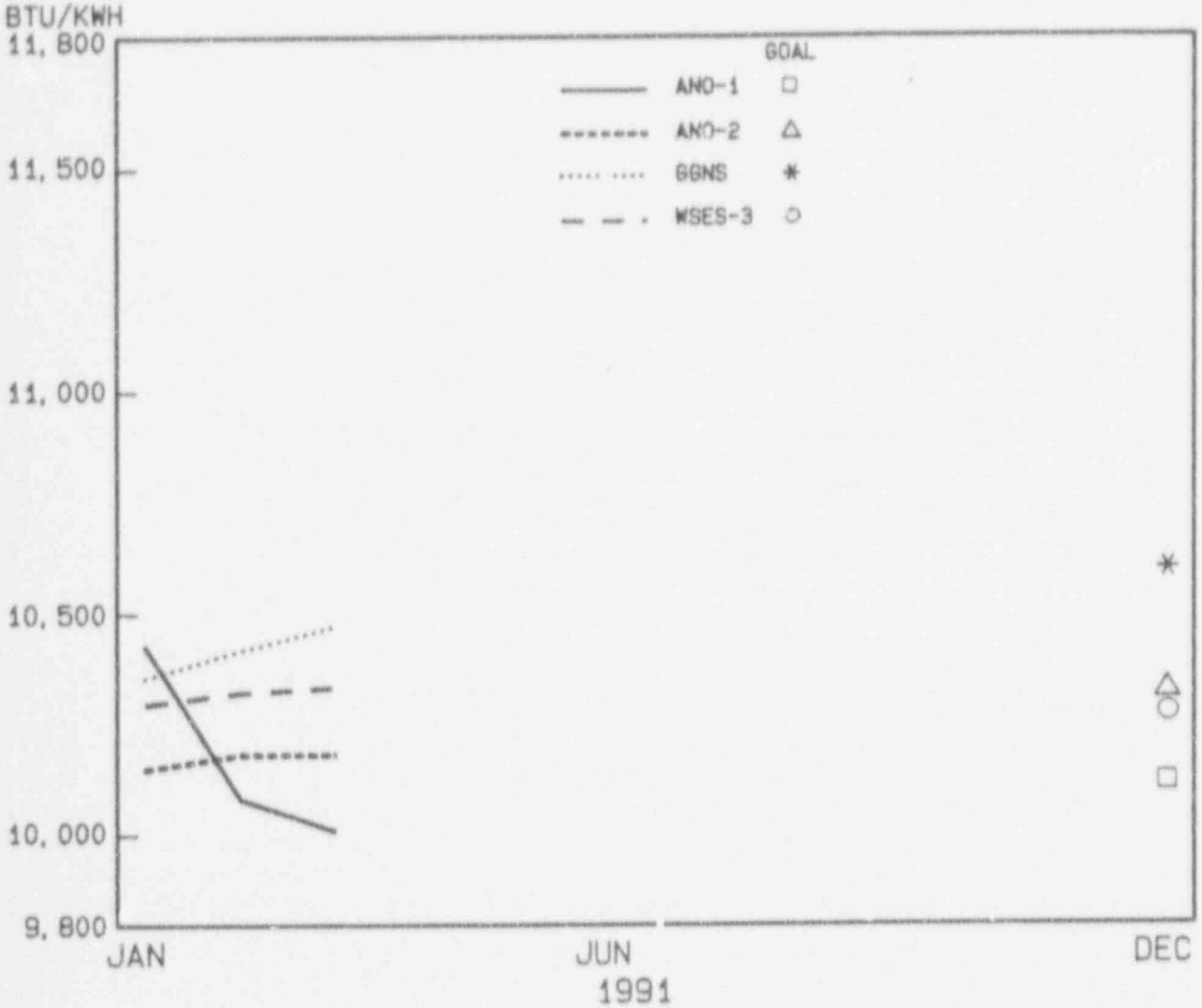
HIGH THERMAL PERFORMANCE INDEX INDICATES POSITIVE PERFORMANCE.

GRAPH 10

# ENTERGY OPERATIONS, INC.

# Unit Gross Heat Rate

## Year-To-Date




| | MONTHLY AVERAGE | Y-T-D | GOAL |
|---|---|---|---|
| ANO-1 | 9922 | 10011 | ≤10121 |
| ANO-2 | * | 10184 | ≤10325 |
| GGNS | 10575.1 | 10472.7 | ≤10600 |
| WSES-3 | 10374 | 10332 | ≤10275 |

* ANO-2 IN REFUELING OUTAGE

1990 INDUSTRY AVERAGE 10218 BTU/KWH

GROSS HEAT RATE (BTU/KWH) MEASURES THE EFFICIENCY OF THE UNIT IN CONVERTING THERMAL ENERGY INTO ELECTRICAL ENERGY.

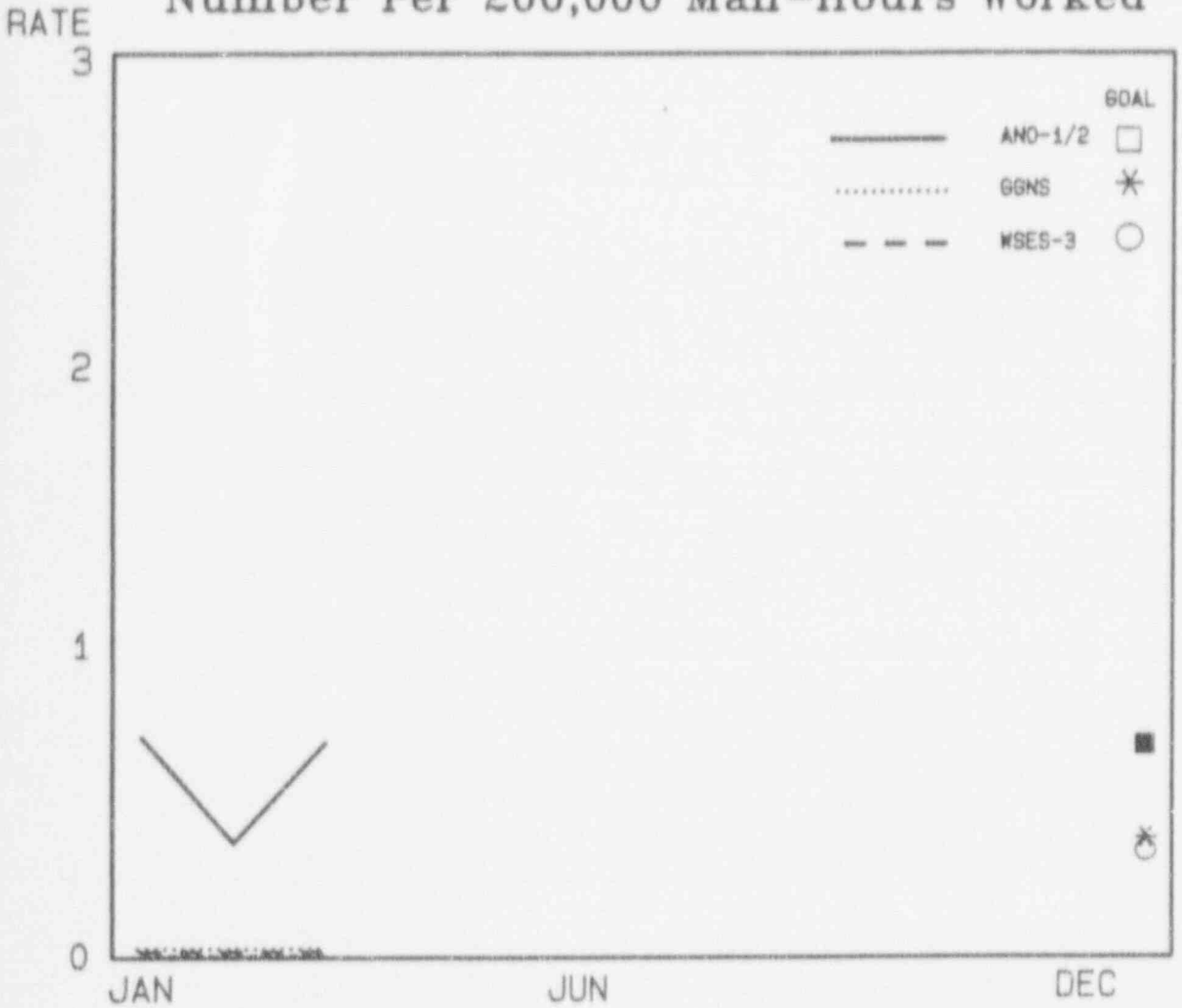
LOW GROSS HEAT RATES INDICATE POSITIVE PERFORMANCE.

GRAPH 11

# ENTERGY OPERATIONS, INC.

# Y-T-D Industrial Safety Accident Rate

## Number Per 200,000 Man-Hours Worked




| | CURRENT MONTH | YTD | GOAL |
|---|---|---|---|
| ANO-1/2 | 1.26 | 0.712 | ≤0.705 |
| GGNS | 0 | 0 | ≤0.394 |
| WSES-3 | 0 | 0 | ≤0.35 |

1990 INDUSTRY MEDIAN 0.72
1990 INDUSTRY BEST QUARTILE 0.28

LOST TIME ACCIDENT RATE MEASURES THE NUMBER OF ACCIDENTS RESULTING IN FATALITIES, ABSENCES, OR RESTRICTED WORK ACTIVITIES PER 200,000 MAN-HOURS WORKED.

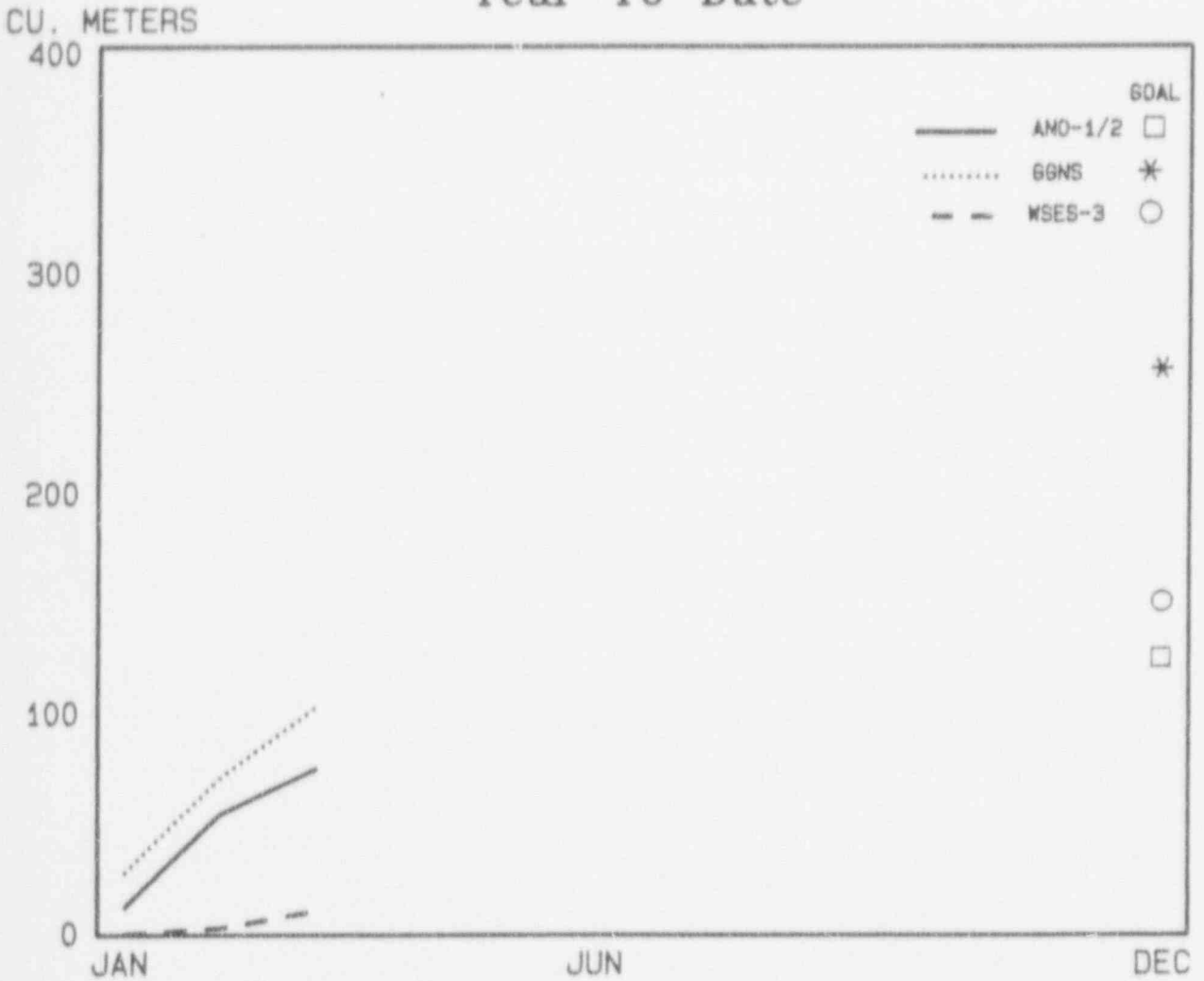
LOW INDUSTRIAL SAFETY ACCIDENT RATE INDICATES POSITIVE PERFORMANCE.

GRAPH 12

ENTERGY OPERATIONS, INC.

# Volume of Low Level Waste Per Unit

## Year-To-Date




| | CURRENT MONTH | Y-T-D | GOAL |
|---|---|---|---|
| ANO-1/2 (PER UNIT) | 20.41 | 74.93 | ≤125 |
| GGNS | 32.0 | 103.1 | ≤255 |
| WSES-3 | 7.8 | 10.6 | ≤150 |

| | PWR | BWR |
|---|---|---|
| 1990 INDUSTRY AVERAGE | 108 | 301 |
| 3 YEAR (88-90) IND. BEST QUARTILE | 94 | 260 |
| 3 YEAR (88-90) IND. MEDIAN | 138 | 300 |

TOTAL VOLUME OF LOW LEVEL SOLID RADIOACTIVE WASTE MEASURES THE AMOUNT OF WASTE READY FOR SHIPMENT OR SHIPPED DURING A GIVEN PERIOD.

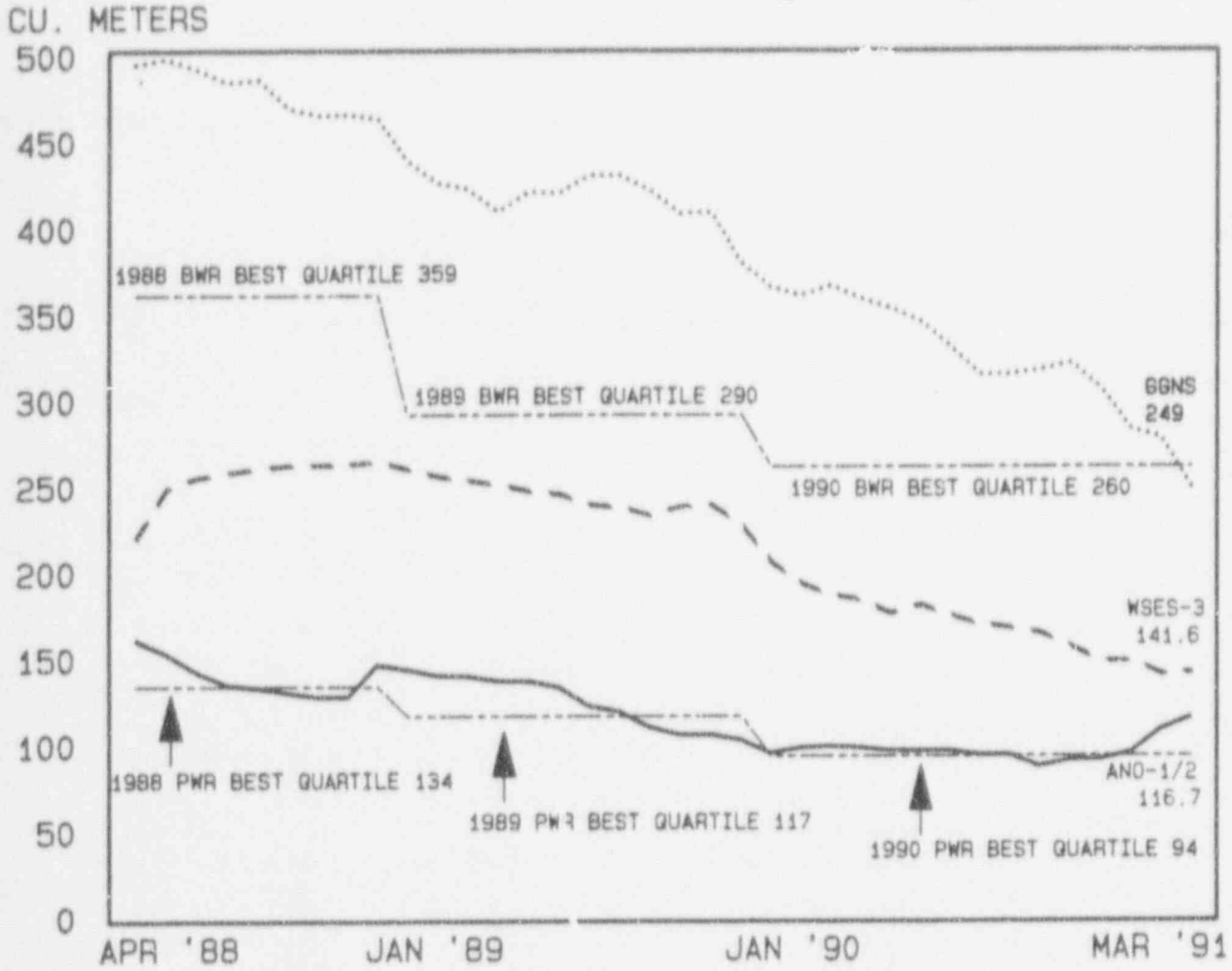
LOW VOLUMES OF LOW LEVEL WASTE INDICATE POSITIVE PERFORMANCE.

GRAPH 13

# ENTERGY OPERATIONS, INC.

## Volume of Low Level Waste

### Three Year Moving Average




TOTAL VOLUME OF LOW LEVEL SOLID RADIOACTIVE WASTE MEASURES THE AMOUNT OF WASTE READY FOR SHIPMENT OR SHIPPED DURING A GIVEN PERIOD.

LOW VOLUMES OF LOW LEVEL WASTE INDICATE POSITIVE PERFORMANCE.

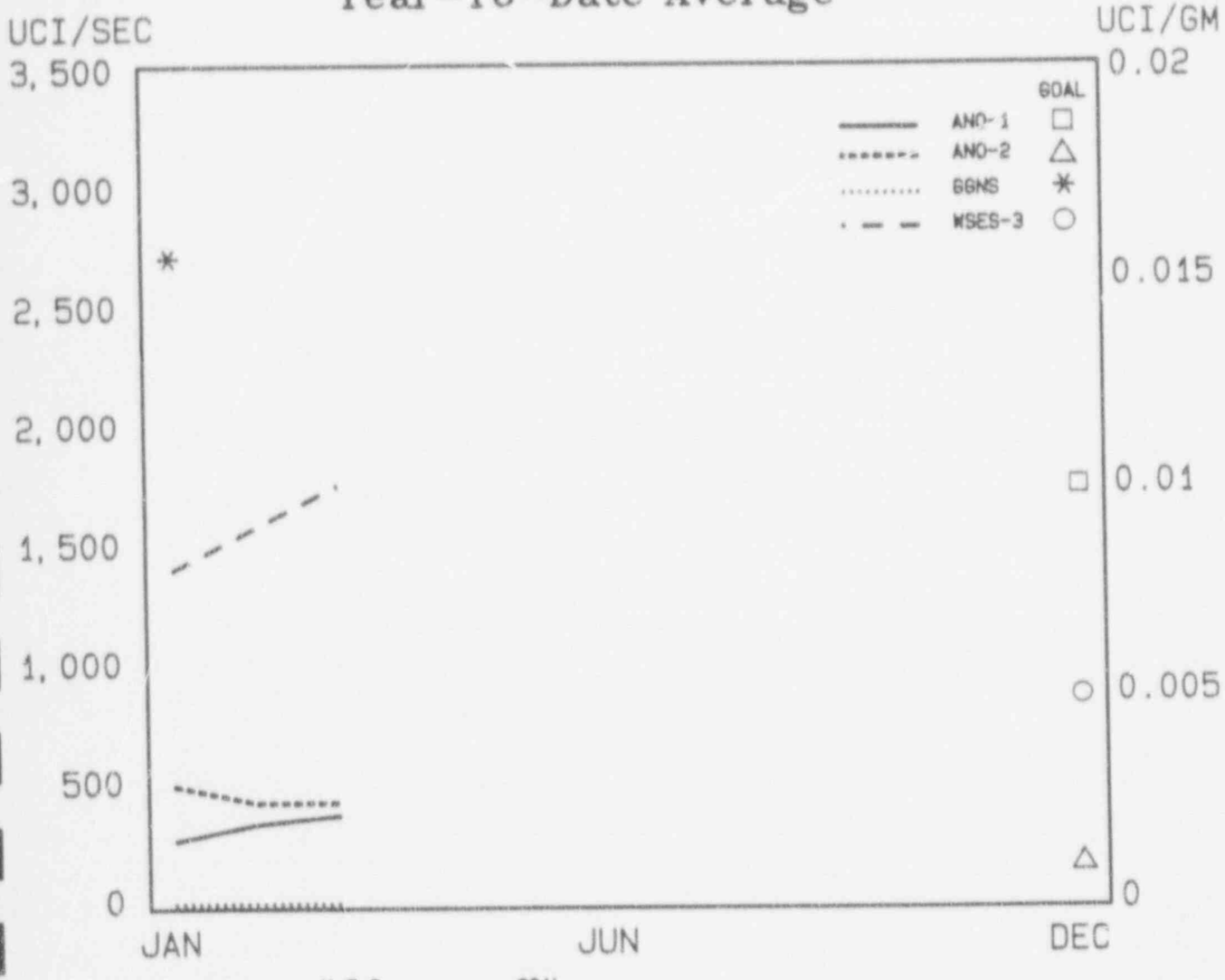
SOME OF THE DATA INCLUDES ESTIMATES OF THE VOLUME OF DRY ACTIVE WASTE PROCESSED AND READY FOR SHIPMENT AT THE COMPACTION CONTRACTOR'S FACILITIES.

GRAPH 14

ENTERGY OPERATICNS, INC.

# Fuel Reliability Index

## Year-To-Date Average




| | Y-T-D | GOAL |
|---|---|---|
| ANO-1 | 0.0022 | ≤ 0.01 UCI/GM |
| ANO-2 | 0.0025 | ≤ 0.001 UCI/GM |
| GGNS | 9.1 | ≤ 2700 UCI/SEC |
| WSES-3 | 0.010 | ≤ 0.005 UCI/GM |

| | PWR | BWR |
|---|---|---|
| 1990 IND. MEDIAN | .0012 (UCI/GM) | 99.0 (UCI/SEC) |

FUEL RELIABILITY MEASURES THE INTEGRITY OF THE UNIT'S FUEL RODS.
LOW FUEL RELIABILITY NUMBERS INDICATE POSITIVE PERFORMANCE.

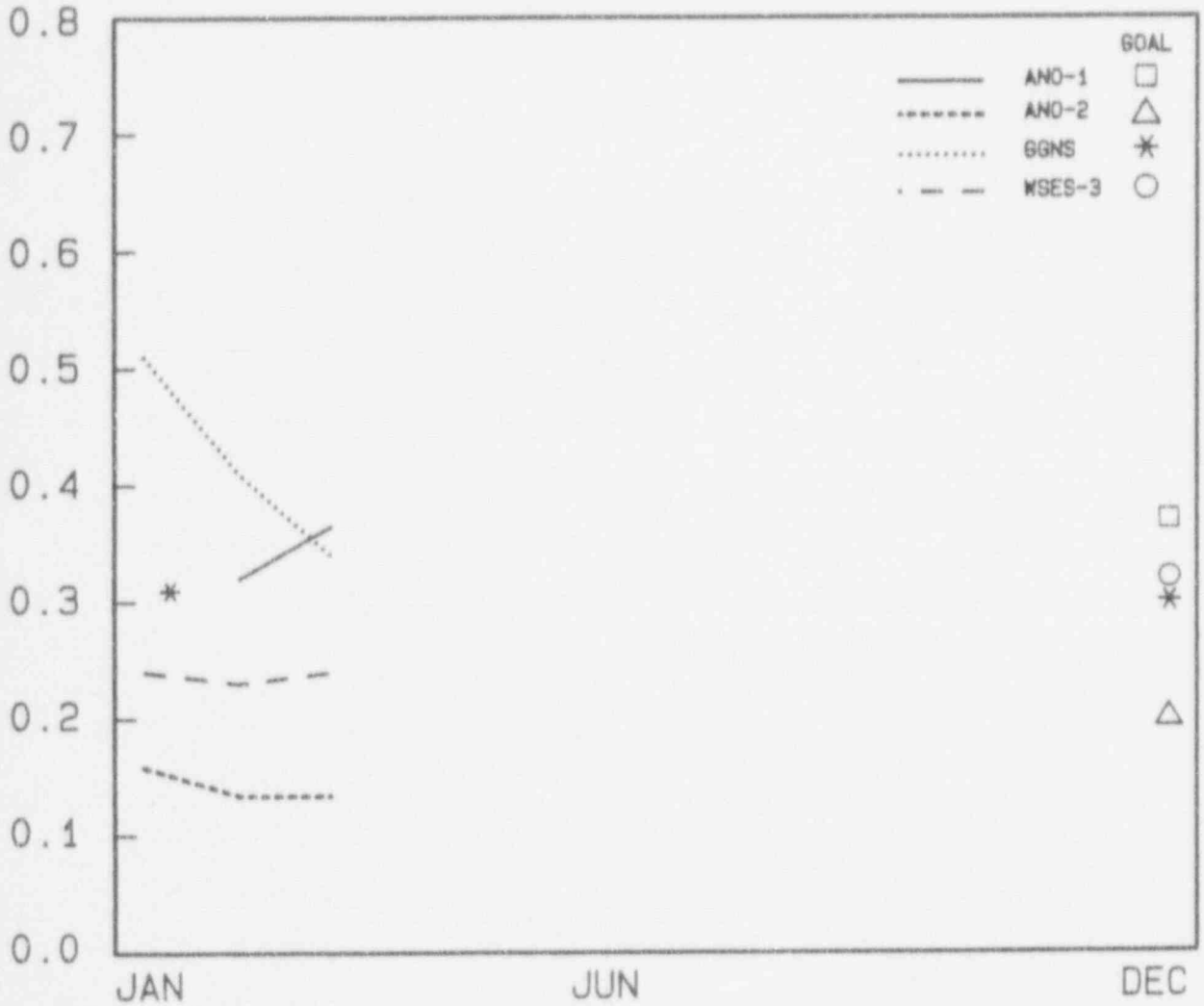
NOTE: PWR INDUSTRY DATA ARE BASED ON CORRECTED IODINE-131 (UCI/GM). BWR INDUSTRY DATA ARE BASED ON CORRECTED OFFGAS RELEASE RATES (UCI/SEC). FOR BWRs, A VALUE BELOW 300 UCI/SEC INDICATES A HIGH PROBABILITY THAT THERE ARE NO FUEL FAILURES. FOR PWRs, THE THRESHOLD VALUE IS 0.5E-3 UCI/GM.

GRAPH 15

# ENTERGY OPERATIONS, INC.

# CHEMISTRY INDEX

## Year-To-Date Average




| | Y-T-D | GOAL | 1990 INDUSTRY MEDIAN | 1990 BEST QUARTILE |
|---|---|---|---|---|
| ANO-1 | 0.365* | ≤ 0.37 | 0.45 | 0.37 |
| ANO-2 | 0.135 | ≤ 0.20 | 0.21 | 0.17 |
| GGNS | 0.34 | ≤ 0.30 | 0.36 | 0.29 |
| WSES-3 | 0.24 | ≤ 0.32 | 0.21 | 0.17 |

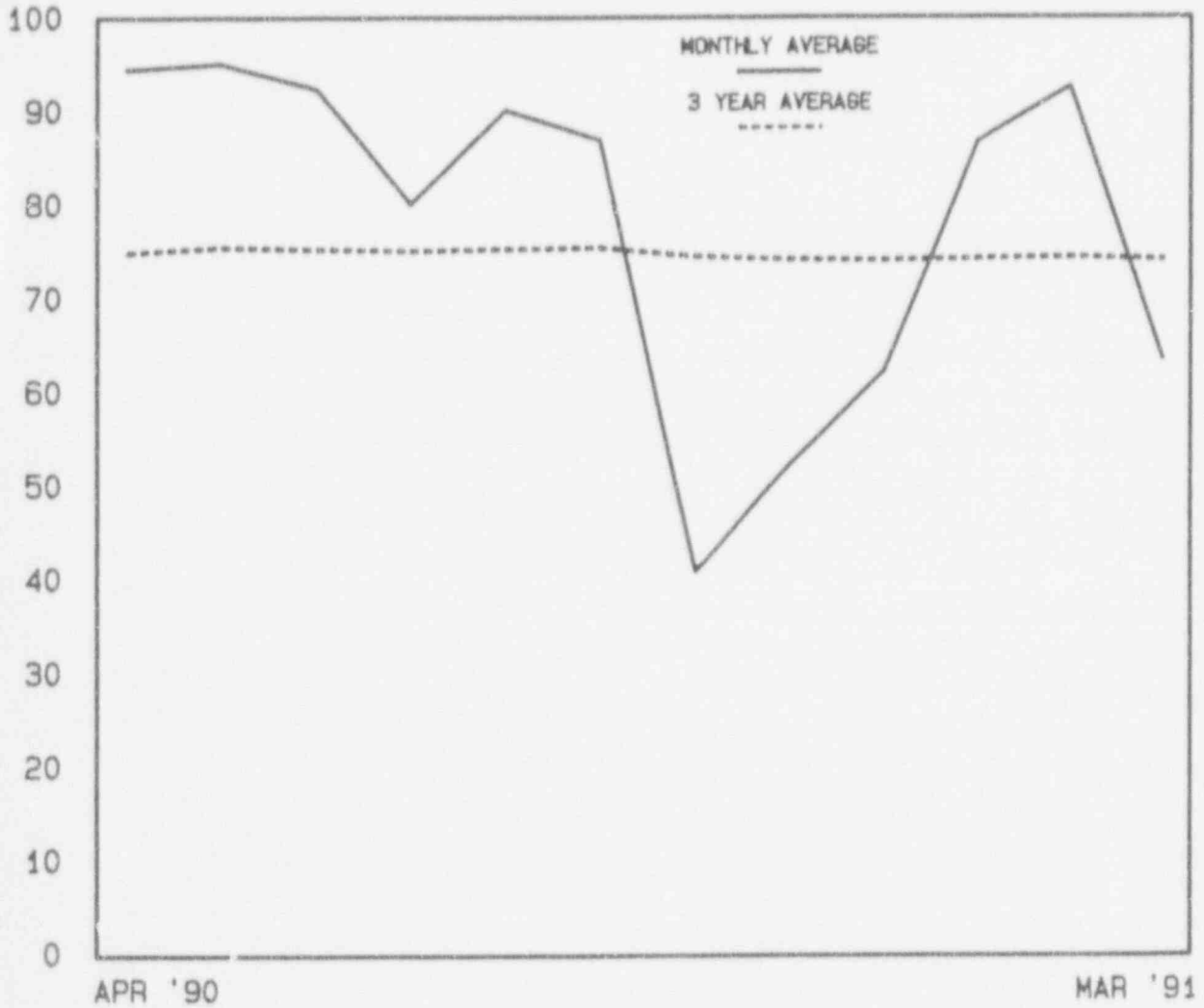
* ANO-1 JANUARY DATA NOT AVAILABLE - UNIT NOT IN STEADY STATE OPERATION

A LOW CHEMISTRY INDEX NUMBER INDICATES POSITIVE PERFORMANCE.

GRAPH 16

ENTERGY OPERATIONS, INC.

# System Nuclear Unit Equivalent Availability




| | |
|---|---|
| MONTHLY AVERAGE | 63.53 |
| Y-T-D AVERAGE | 80.61 |
| 3 YEAR AVERAGE | 74.00 |
| 12 MONTH AVERAGE | 77.95 |

3 YEAR (1988 - 1990) INDUSTRY BEST QUARTILE 77.8%

3 YEAR (1988 - 1990) INDUSTRY MEDIAN 71.0%

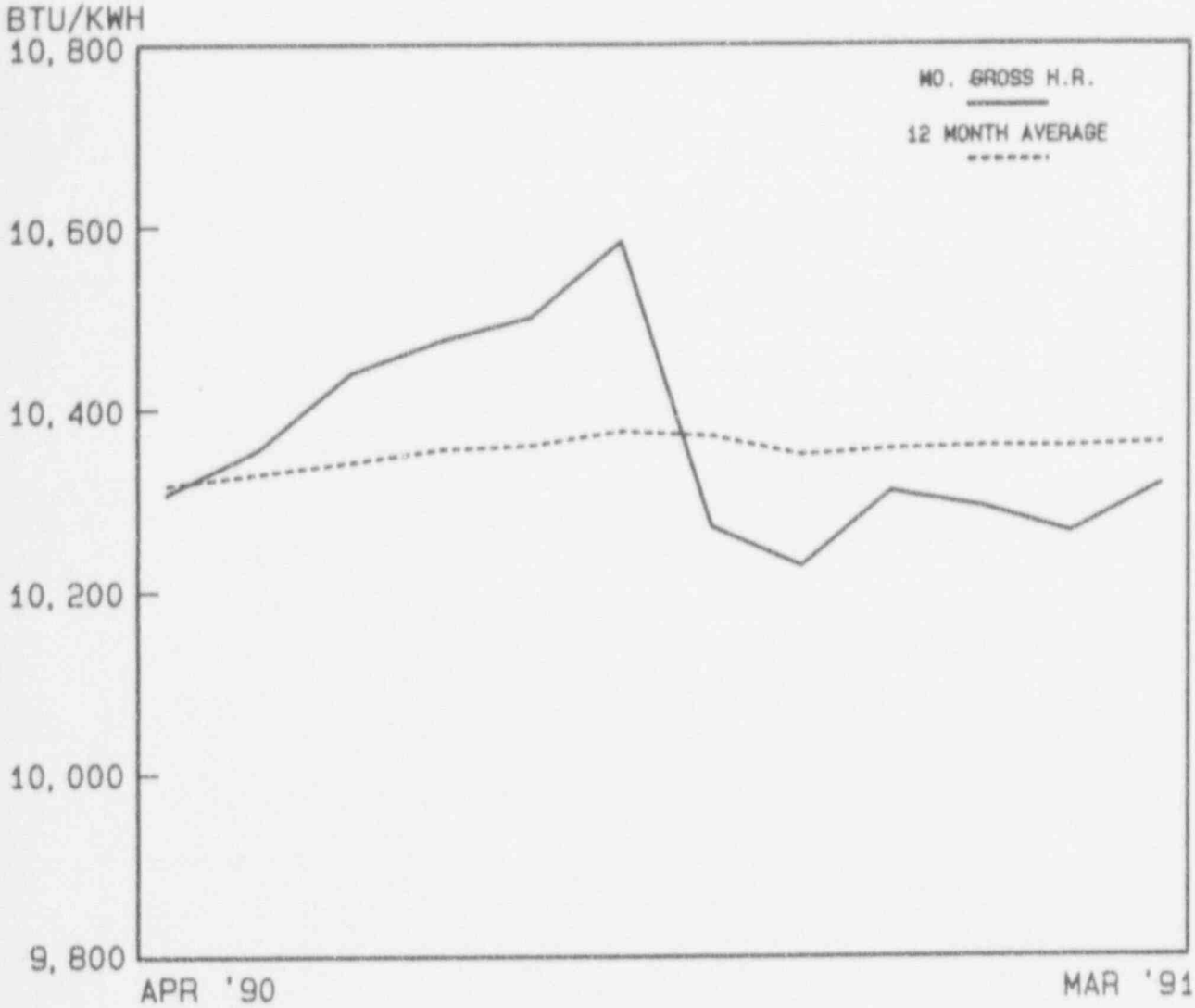
UNIT EQUIVALENT AVAILABILITY MEASURES THE FRACTION OF A PERIOD THAT A UNIT WAS AVAILABLE FOR RATED GENERATION SERVICE.

HIGH EQUIVALENT AVAILABILITY INDICATES POSITIVE PERFORMANCE.

GRAPH 17

Entergy Operations, Inc.

# System Nuclear Unit
# Gross Heat Rate




MONTHLY AVERAGE 10317

ROLLING 12 MONTH AVERAGE 10362

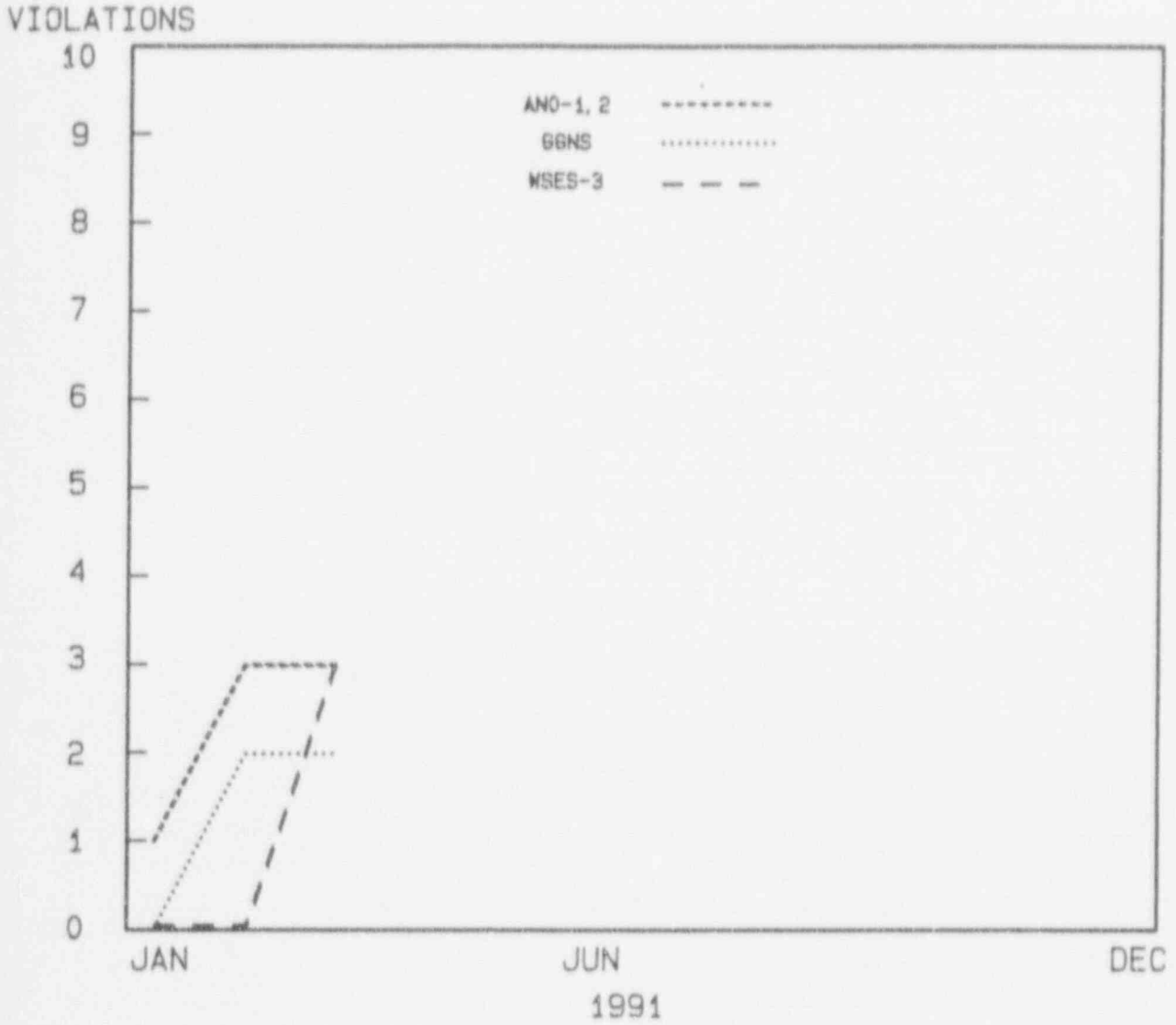
1989 INDUSTRY AVERAGE 10218 BTU/KWH

GROSS HEAT RATE (BTU/KWH) MEASURES THE EFFICIENCY OF THE UNIT IN CONVERTING THERMAL ENERGY INTO ELECTRICAL ENERGY.

LOW GROSS HEAT RATES INDICATE POSITIVE PERFORMANCE.

GRAPH 18

# NRC VIOLATIONS
# YEAR-TO-DATE




GRAPH 19

## NRC VIOLATIONS

## RECEIVED IN MARCH, 1991

ANO: None

GGNS: None

WSES-3: Level III — Civil penalty of $37,500 assessed for combination of three related violations pertaining to the control room air conditioning system (CRHVAC). Enforcement was mitigated due to self identification, prompt reporting, and prompt and comprehensive corrective actions by WSES.

1. Failure to properly specify design basis in test acceptance criteria.
2. Existence of condition in which CRHVAC did not meet design basis requirements.
3. Inadequate control over fire/air seal work activities.

Level IV — Failure of plant operations review committee to review a portion of acceptance testing involving a modification to the plant protection system.

Level IV — Failure to provide appropriate Fitness-for-Duty supervisory training to some contract supervisors whose work fell within the scope covered by the FFD program.

### 1991 YEAR-TO-DATE-TOTAL

| | Severity Level | | | | | |
|---|---|---|---|---|---|---|
| Plant: | I | II | III | IV | V | TOTAL |
| ANO: | | | | 2 | 1 | 3 |
| GGNS: | | | | 2 | 0 | 2 |
| WSES-3 | | | 1 | 2 | 0 | 3 |

### 1990 TOTAL VIOLATIONS

| | Severity Level | | | | | |
|---|---|---|---|---|---|---|
| | I | II | III | IV | V | TOTAL |
| | | | 2 | 21 | 2 | 25 |
| | | | | 9 | | 9 |
| | | | | 13 | | 13 |

ARKANSAS NUCLEAR ONE
STATUS REPORT
(Status as of 04/09/91)

A. PLANT STATUS

UNIT 1

The plant is shutdown for Modification Outage.

| | 4/9/91 YTD |
|---|---|
| o Equivalent Availability (%) | 79.9 |
| o Unit Capability Factor (%) | 79.3 |
| o Capacity Factor (%) | 80.0 |
| o Gross Heat Rate (BTUs/KWH) | 10005 |
| o Consecutive Days On-Line - 0 | |

UNIT 2

The plant is shutdown for refueling.

| | 4/9/91 YTD |
|---|---|
| o Equivalent Availability (%) | 51.3 |
| o Unit Capability Factor (%) | 51.7 |
| o Capacity Factor (%) | 50.4 |
| o Gross Heat Rate (BTUs/KWH) | 10184 |
| o Consecutive Days On-Line - 0 | |

ARKANSAS NUCLEAR ONE
STATUS REPORT
(Status as of 04/09/91)

## B. OPERATING SUMMARY SINCE LAST REPORT

### UNIT 1

- Unit 1 was shutdown 4/9/91 at 7:38 a.m. for a scheduled two-week modification outage.

### UNIT 2

- The eighth refueling outage of Unit 2 began on 02/22/91 at 11:04 p.m.

## C. SIGNIFICANT EVENTS

- There were seven NRC inspections initiated during March. Zero violations and zero deviations were received and two inspector follow-up items were received. During the current SALP period beginning December 1, 1990 through March 1991, seven violations have been received. This represents seven fewer violations received for the same period during the previous SALP period.

- Inspection activities included:

  1) Modification, calibration and snubbers
  2) Resident inspector's inspection
  3) Radiation Protection Program
  4) Verification of Isolation Containment Exemptions (VOICE)
  5) NRC Measurements Van (Chemistry Program)
  6) Eddy Current Testing

- A shutdown Operations Information visit was conducted by NRR.

- A response to the NRC SALP report was submitted 3/19/91.

- Copper dusting was found in the Unit 2 main generator resulting in the need for rotor rewinding. This activity is critical path and will continue until after heatup of the primary system.

- Eddy current testing of the steam generators resulted in the need to plug 76 tubes.

# LICENSEE EVENT REPORT

Arkansas Nuclear One

Unit 1

Arkansas Nuclear One
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 1

**Title:** Inadvertent Start of a Low Pressure Injection Pump During Post Maintenance Testing Due to Personnel Error

**LER No:** 50-313/90-017-00

**Date Submitted:** January 7, 1991

On December 7, 1990, at approximately 0125, an inadvertent automatic start of the 'B' Decay Heat Removal/Low Pressure Injection pump (P-34B) occurred during testing. On November 4, 1990, a time delay relay (162-405) which enables the Low Pressure Injection low flow alarm was replaced using an approved work plan. The prerequisites for the workplan required the circuit breaker for P-34B to be tagged open because the method of testing relay 162-405 would also initiate a pump start. Due to plant conditions at that time, the post maintenance test could not be performed and the hold card was released. On December 7, the electrician assigned to perform the test checked that the prerequisites were signed off, but did not physically reverify them. When a jumper was installed to test relay 162-405, P-34B started. The pump ran for approximately 5 seconds before being secured by an operator. The root cause of this event was personnel error. This event was discussed with Electrical Maintenance personnel. The importance of reverifying prerequisites prior to resuming jobs which are unexpectedly halted was stressed. Additionally, the 'Conduct of Maintenance' procedure will be revised to include appropriate guidance in this area.

**Title:** Inadvertent Actuation of the Control Room Emergency Ventilation System Due to A Spurious Trip of a Radiation Monitor Caused By An Unsoldered Connection Which Resulted From a Manufacturing/Production Defect

**LER No:** 50-313/90-019-00

**Date Submitted:** January 8, 1991

On December 9, 1990, at approximately 2228, an inadvertent actuation of the Control Room Emergency Ventilation System (CREVS) occurred. At the time of the actuation, Operations personnel observed that the indication of the ANO-2 Control Room (CR) ventilation radiation monitor (2RE-8750-1) failed low, then increased to the trip setpoint and actuated the CREVS. 2RE-8750-1 was reset and the ventilation lineup was returned to normal. However, at 2235, the monitor was

LER 50-313/90-019-00 (Cont'd)

declared inoperable and the CR was isolated and ventilation was placed in the recirculation mode. The immediate cause of this event was determined to be an unsoldered electrical connection on the radiation monitor operation selector switch. The switch was repaired, the monitor was returned to service and the ventilation system was returned to normal at 0855 on December 12, 1990. A review of the maintenance records was conducted and no documentation was found to indicate that previous maintenance had been performed on the switch. Therefore, it was concluded that the most likely root cause of this event was a manufacturing/production defect. Other ANO-2 Technical Specifications radiation monitors were visually inspected. Since only one additional unsoldered connection was identified, it was concluded that this condition was not a generic problem.

**Title:** Design Deficiency Results in Potential for Structural Damage or Failure of Containment Polar Crane During Design Basis Accident Conditions

**LER No:** 50-313/90-020-00

**Date Submitted:** January 10, 1991

On October 30, 1990 during a refueling outage, Design Engineering personnel determined that an analysis had never been performed to ensure that certain structural components of the ANO-1 containment building polar crane could withstand the effects of a rapid increase in containment pressure during a loss of coolant accident without sustaining structural damage. The potential concern was that inadequate venting could result in a large differential pressure across the crane's girders which might cause the girders to yield or collapse allowing the crane or a part of the crane structure to fall from its stored position. Further investigations determined that the ANO-2 polar crane, which is similarly designed, had been modified during the construction phase of the unit to address the same concern. The ANO-1 polar crane vendor was consulted and an analysis was performed which indicated that modifications were necessary to ensure the crane components were adequately vented. The crane was modified by cutting vent holes in the bridge girders, trolley sides and end trucks. The root cause was determined to be an oversight by the ANO-1 architect engineer during the construction phase of ANO-1.

Arkansas Nuclear One
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 1

**Title:** Reactor Shutdown Required By Technical Specification Due To Unisolable Leak In A Pressurizer Nozzle Which Was Caused by Pure Water Stress Corrosion Cracking

**LER No:** 50-313/90-021-00

**Date Submitted:** January 21, 1991

On December 22, 1990, maintenance personnel identified a potential Reactor Coolant System leak in the area of a pressurizer upper level instrumentation nozzle. An inspection was conducted which verified the existence of a very small leak at the nozzle. A Notification of Unusual Event was declared at 1011, and the plant was taken to cold shutdown. Subsequent inspection using Nondestructive Examination methods confirmed the existence of a small axial crack in the nozzle inner surface which extended to the annulus between the nozzle and the pressurizer shell and breached the outside diameter (OD) of the nozzle at the toe of the nozzle to vessel weld. Based on the location and orientation of the flaw, and industry experience, the most probable root cause was determined to be Pure Water Stress Corrosion Cracking. A temporary repair was completed which consisted of establishing the nozzle pressure boundary at the outside surface of the pressurizer and installing a new nozzle into the penetration from the shell OD. A Design Change Package will be developed and implemented during the next refueling outage (1R10) to provide a permanent repair to the nozzle.

**Title:** Reactor Trip During Plant Heatup Due to Personnel Error While Shifting Reactor Coolant Pumps

**LER No:** 50-313/90-022-00

**Date Submitted:** January 17, 1991

On December 18, 1990, while conducting a plant heatup in preparation for startup, an automatic reactor trip was initiated by the Reactor Protection System (RPS) upon sensing no reactor coolant pumps (RCPs) running in the "B" Reactor Coolant System (RCS) loop. At the time of the trip, RCPs P-32C and P-32D were running in RCS loop 'A' and P-32A was running in loop 'B'. RCPs were being balanced to reduce vibration in accordance with an approved procedure. The operators were requested to shift from P-32A to P-32B in RCS loop 'B'. After

LER 50-313/90-022-00 (Cont'd)

reviewing the RCP operating procedure, the involved operators asked the Shift Supervisor (SS) if he wished to stop P-32A and start P-32B. The SS gave an affirmative response. At that time, a trainee under the supervision of a senior reactor operator, stopped P-32A. A reactor trip then occurred due to zero pumps running in the 'B' RCS loop. The root cause of this event was personnel error. An inadequate procedure was a contributing factor. The RCP operating procedure contained no cautions regarding the possibility of initiating trips when stopping RCPs. A crew briefing was held with the crew involved to discuss this event and its significance. The RCP operating procedure will be revised to include additional guidance regarding shifting RCPs.

**Title:** Automatic Reactor Trip Due To A Main Turbine Trip Which Was Caused By Failure Of The Turbine Generator Exciter

**LER No:** 50-313/91-001-00

**Date Submitted:** February 11, 1991

On January 10, 1991, at approximately 2326, with the plant at 100 percent of rated power, a reactor trip occurred as a result of the main turbine tripping due to loss of field excitation to the main generator. An anticipatory Reactor Protection System (RPS) trip was initiated, as designed, when the main turbine tripped while reactor power was greater than 43 percent. Plant response to the trip was as expected. Reactor Coolant System (RCS) pressure decreased to 1828 psig and was quickly recovered into the post trip window. Minimum post trip RCS temperature was 553 degrees. A temporary exciter was installed while the plant remained in the hot shutdown condition and the reactor was returned to power on January 17, 1991. The temporary exciter will be replaced with a permanent exciter during mid cycle outage 1M91, which is scheduled to begin in April, 1991. An investigation to determine the root cause of the exciter failure is being conducted by the vendor (Westinghouse). The results of the completed investigation and the subsequent corrective actions to prevent recurrence of similar events will be included in a supplement to this report which will be submitted by April 30, 1991.

Arkansas Nuclear One
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 1

**Title:** Inadvertent Actuation of the Control Room Emergency Ventilation System Initiated by a Trip of a Chlorine Monitor Most Likely Caused by Radio Frequency Interference

**LER No:** 50-313/90-011-01

**Date Submitted:** March 29, 1991

On September 30, 1990, at approximately 0050, an unexpected actuation of the Control Room Emergency Ventilation System (CREVS) occurred. Investigation into the cause of the actuation revealed that chlorine monitor 2CLS-8762-2 was tripped. However, the immediate cause of the monitor trip could not be positively determined. Since no actual high chlorine condition existed, the monitor was reset and the Control Room ventilation lineup was returned to normal at 0058 hours. The most likely cause of the actuation was radio frequency interference (RFI) caused by the keying of a hand held radio in the vicinity of the monitor. However, the root cause of this event is directly related to system design. The extreme sensitivity of the chlorine monitors coupled with the actuation logic configuration, which requires only one monitor to trip in order to initiate the CREVS, makes the system highly susceptible to inadvertent actuations. Action has been completed to better mark areas in the plant where radio usage is prohibited. Additionally, an evaluation is being conducted to determine the feasibility of amending the Technical Specifications to delete requirements for the system chlorine monitors.

# LICENSEE EVENT REPORT

Arkansas Nuclear One

Unit 2

Arkansas Nuclear One
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 2

**Title:** Inadequate Preventive Maintenance Program For Steam Turbine Driven Emergency Feedwater Pump Results In Degraded Turbine Governor System And Subsequent Overspeed Trips Of Turbine.

**LER No:** 50-368/90-024-00

**Date Submitted:** January 4, 1991

On December 5, 1990 based on evaluations of two previous events involving overspeed trips of the steam turbine driven emergency feedwater pump, it was concluded that the cause of the trips had been water slugging of the turbine on startup due to condensate accumulation in the steam supply line to the turbine. Following another overspeed trip on December 6, 1990, the actual cause for the turbine trips was found to be sluggish response of the turbine governor valve due to a contaminated control oil system. The root cause was considered to be inadequacies in the preventive maintenance program. The program did not appropriately address and minimize the potential effects of oil contamination and degradation of governor components over time. Following the last overspeed trip, the oil and oil filter assembly were changed, a hydraulic actuator was replaced and a remote servo valve and control oil tubing were cleaned. The turbine is being tested on an increased frequency and the oil quality is being monitored to ensure it is not degrading. Long term actions include procedure revisions to include periodic cleaning and/or replacement of control oil system components. Additionally, the turbine oil system will be cleaned to remove varnish and hardened oil deposits during the next refueling outage.

**Title:** Inoperable Fire Dampers Result in Technical Specification Violation Due to Failure to Perform Functional Testing Following Installation

**LER No:** 50-368/86-003-02

**Date Submitted:** February 15, 1991

During the initial performance of periodic functional testing of fire dampers, a total of 19 fire dampers were identified as inoperable. The testing involves removal of the fire damper fusible link and verifying that the fire damper completely closes in the presence of normal ventilation air flow. Of the 19 inoperable fire dampers, 9

LER 50-368/86-003-02 (Cont'd)

failures were attributed to mechanical interference and 10 were attributed to a design deficiency of the fire damper. The cause of this event was inadequate functional testing of installed fire dampers in that the ability of the fire dampers to completely close with normal ventilation air flow had not been previously verified. As a result of this event, the fire dampers that failed to completely close due to mechanical interference were repaired and successfully tested. A plant modification has been implemented to replace the fire dampers that failed to completely close under normal ventilation air flow. Periodic functional testing will be discontinued to eliminate the potential for personnel injury or equipment damage. The performance of functional tests following maintenance or modification activities combined with the Technical Specification required visual inspections will ensure the continued operability of the fire dampers.

**Title:** Degraded Plant Fire Barriers Which Were Not Properly Identified During Routine Inspections Due To Inadequate Communications Between Different Plant Departments

**LER No:** 50-368/91-001-00

**Date Submitted:** February 13, 1991

In January 1991, while performing additional inspections of plant fire barriers following the recent completion of a routine eighteen month surveillance of the barriers, fire protection personnel discovered several deficiencies which had not been identified during performance of the surveillance activity. Based on evaluations of the deficiencies, it was determined that three fire barriers separating safety related areas were inoperable. Upon discovery of the conditions roving fire watches were established in the affected areas. The root cause of the failure to identify the deficiencies during the surveillance activity was attributed to inadequate communication between Fire Protection personnel and electrical maintenance during a prejob briefing conducted prior to performing the surveillance. Appropriate actions have been initiated to improve the procedures used for inspections and to provide additional training of inspection personnel. Based on the availability of fire detection and fire suppression systems for the affected plant areas and fire brigade personnel, there was no safety significance to these conditions.

Arkansas Nuclear One
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 2

**Title:** Design Deficiency Results In Potential For Failure Of Emergency Diesel Generators Due To Inoperability Of Room Exhaust Fans

**LER No:** 50-368/91-002-00

**Date Submitted:** February 19, 1991

On January 18, 1991, while performing inspections of vital, 480 volt AC, Motor Control Center breakers, the control power transformers (CPTs) in the breakers for the emergency diesel generator (EDG) rooms exhaust fans were found to be undersized. Evaluation of the design deficiency concluded that proper operation of the exhaust fans could not be assured under certain conditions of degraded offsite power voltage. Since the exhaust fans are needed to ensure proper operation of the EDGs, both ANO-2 EDGs were declared to be inoperable. Operating handswitches in the Control Room for one exhaust fan in each EDG room were placed in the 'off' position to prevent automatic starting of the fans. It was determined that the fans could be started manually if required and would not be susceptible to failure if the EDGs were supplying power to the vital electrical busses. Based on this action, both EDGs were declared operable. The undersized CPTs were replaced. The cause of this event appears to be the failure to identify the incorrect CPT size during previous design reviews due to an incorrect vendor drawing for the equipment. A plant electrical design drawing has been revised to indicate the correct CPT size for each applicable breaker.

**Title:** Procedural Inadequacy Results In A Failure To Adhere To The Reduced Power Requirements Of Technical Specifications During Recovery From A Dropped Control Element Assembly

**LER No:** 50-368/91-003-00

**Date Submitted:** February 19, 1991

On January 18, 1991 at 1415, while performing Control Element Assembly (CEA) current traces, Group 6 CEA 46 dropped to its fully inserted position. Operations personnel reduced power to 97.5 percent in accordance with the "CEA Misalignment" procedure (AOP 2203.03). At 1429, after determining that the cause of the dropped rod had been the inadvertent opening of its circuit breaker, the

LER 50-368/91-003-00 (Cont'd)

operators commenced withdrawing CEA 46 to realign it with the rest of the Group 6 CEAs. Reactor power was held constant (97.5 percent) during the recovery effort, as directed by AOP 2203.03. At 1443, CEA 46 reached realignment with Group 6. A subsequent evaluation determined that the time dependent reduced power requirements of Technical Specifications (TS) regarding dropped CEAs was not adhered to during the recovery. TS required that during the period of recovery, power should have been reduced by 5.8 percent. The root cause of this event was determined to be inadequate procedural guidance. AOP 2203.03 was ambiguous with respect to reduced power requirements. This event was discussed with the operations crews. A procedure change was implemented to remove the ambiguities regarding reduced power requirements.

**Title:** Failure To Maintain Control Room Ventilation System Radiation Monitor Alarm/Trip Setpoint Value Within Technical Specifications Due To Personnel Error

**LER No:** 50-368/91-004-00

**Date Submitted:** February 27, 1991

On January 28. 1991 it was discovered that the Unit 2 control room radiation monitor had an alarm/trip setpoint greater than two times background and the control room emergency ventilation system had not been placed in the recirculation mode as required by Technical Specifications. On January 25, 1991 the average background was determined to be 83 counts per minute (CPM) and the alarm/trip setpoint at that time was set to 200 CPM. A job request was initiated to have the setpoint adjusted. It was not recognized until January 28, 1991 that Technical Specification 3.3.3.1 required the setpoint be adjusted within 4 hours. The control room ventilation system was placed in the recirculation mode, and subsequently, the setpoint was adjusted to 150 CPM. The event had no adverse impact on control room habitability; the capability of the monitor to perform its intended function was maintained. The root cause of this event was personnel error. The Operations Manager has issued a night order to remind operations personnel of the requirements of Technical Specification 3.3.3.1, and this event will be discussed during the training cycle following the 2R8 refueling outage.

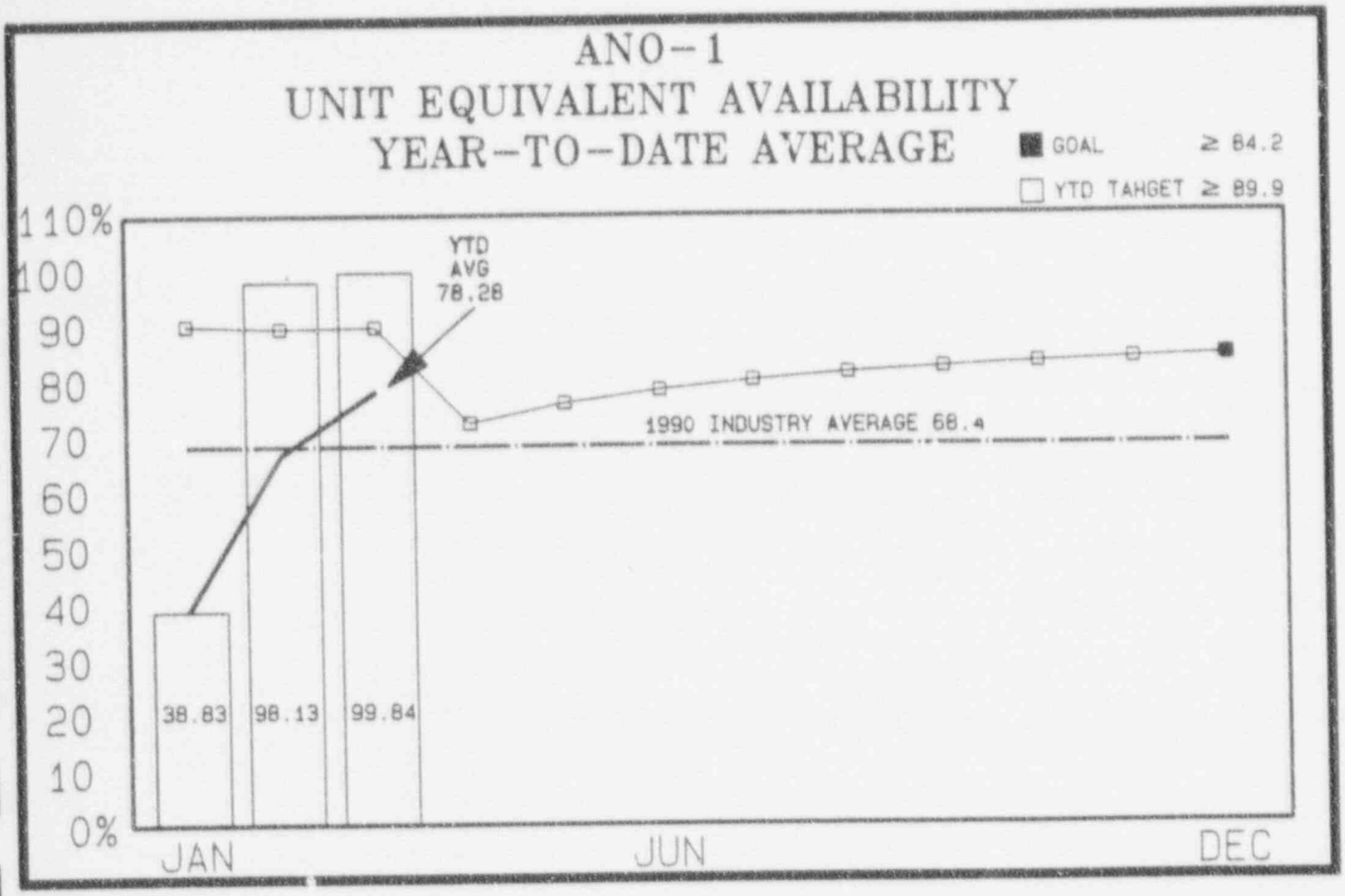
## LEGEND

Graphs with a double line border denote indicators which are within the established goal.

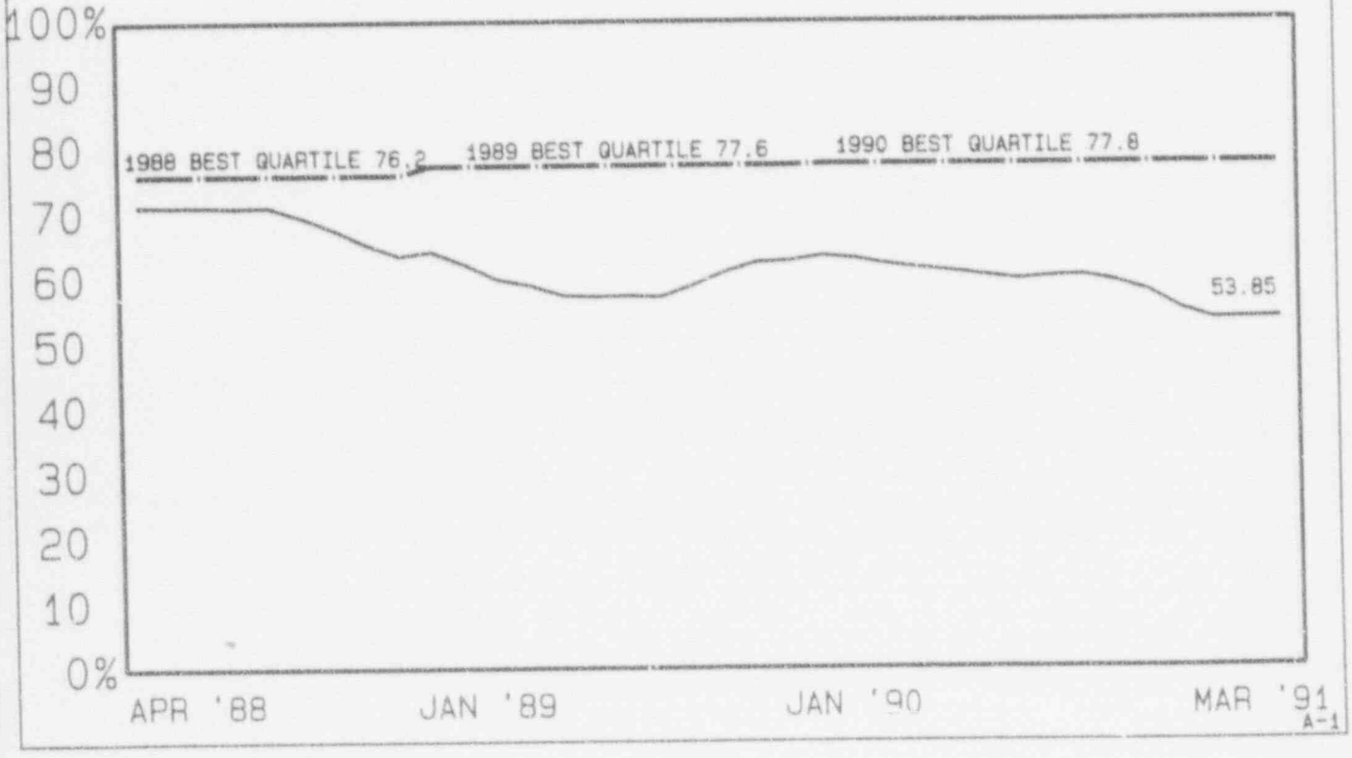
Graphs with a thick, single line border denote indicators which are outside the established goal.

Graphs with a thin, single line border denote indicators for which no goal has been established or for which a goal is not applicable.

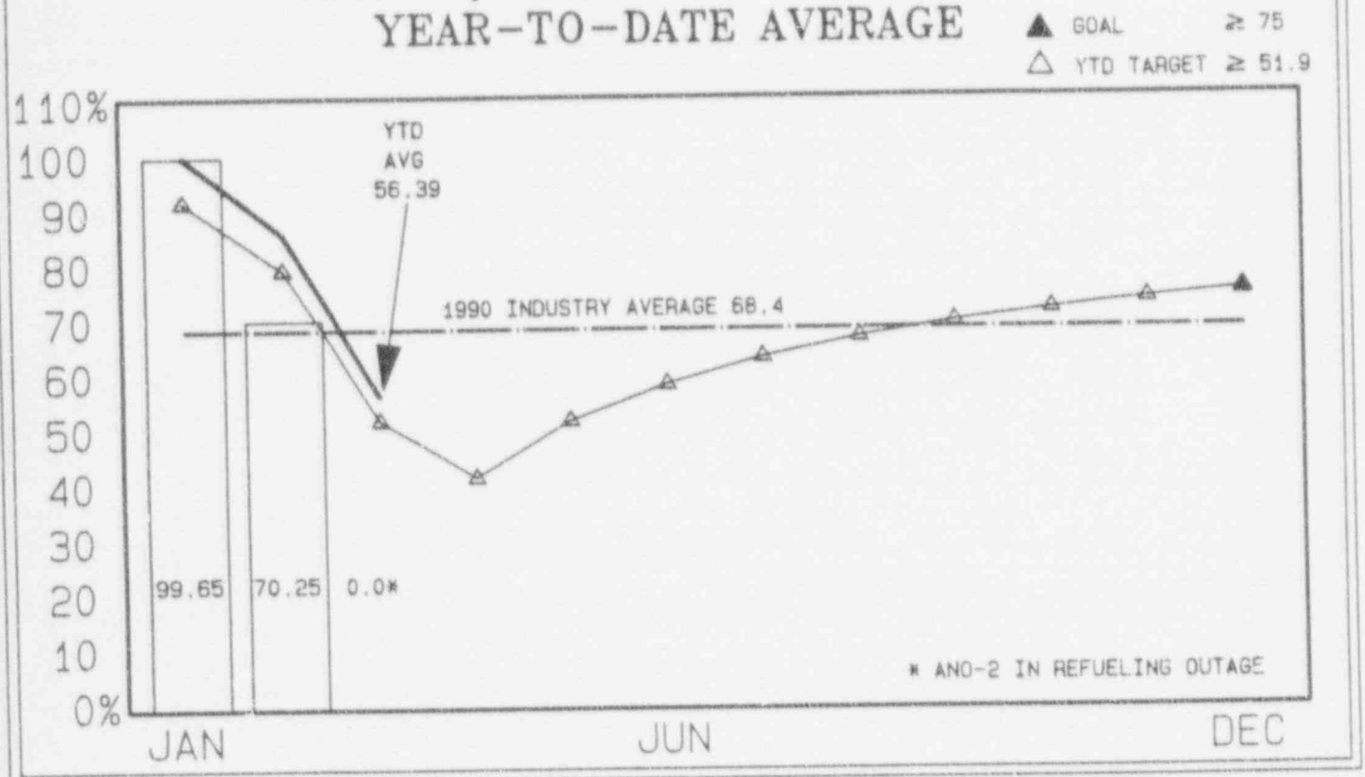
# ANO-1
## UNIT EQUIVALENT AVAILABILITY
## YEAR-TO-DATE AVERAGE



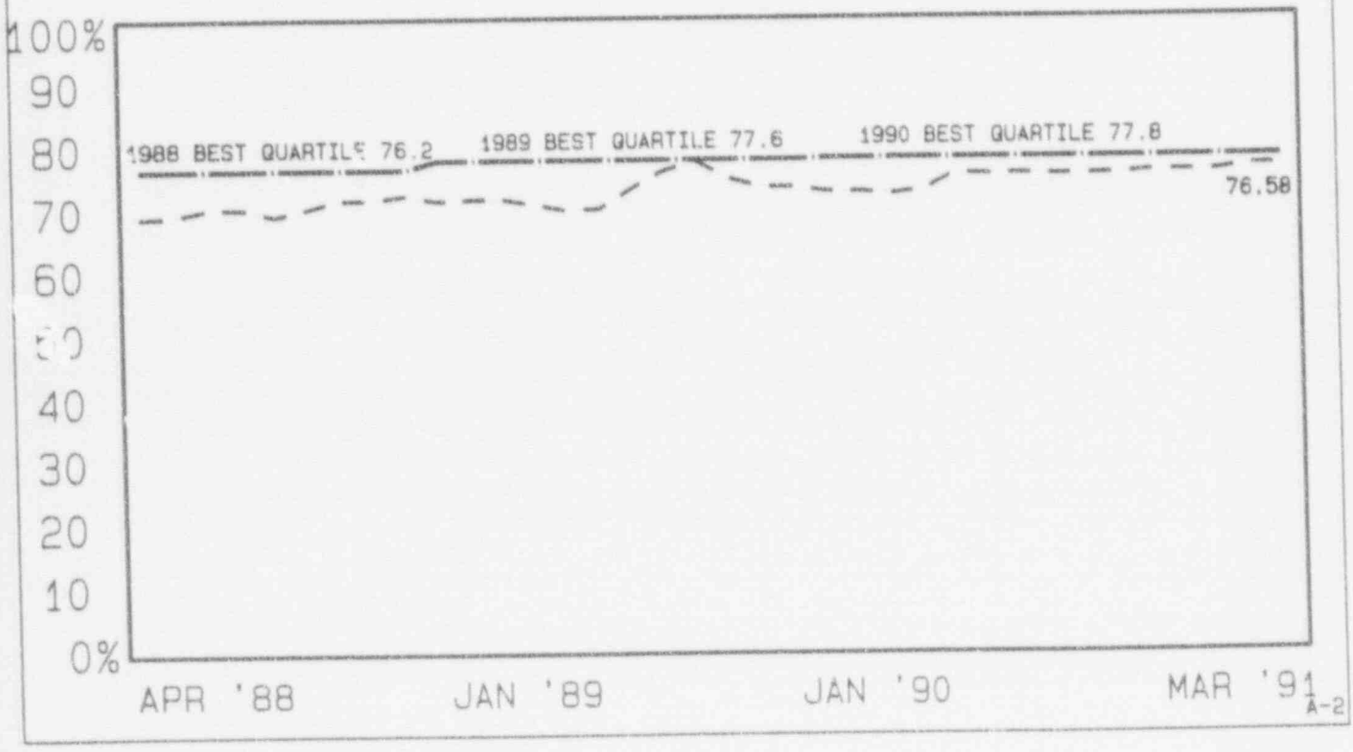

## THREE YEAR MOVING AVERAGE



# ANO-2
## UNIT EQUIVALENT AVAILABILITY
## YEAR-TO-DATE AVERAGE

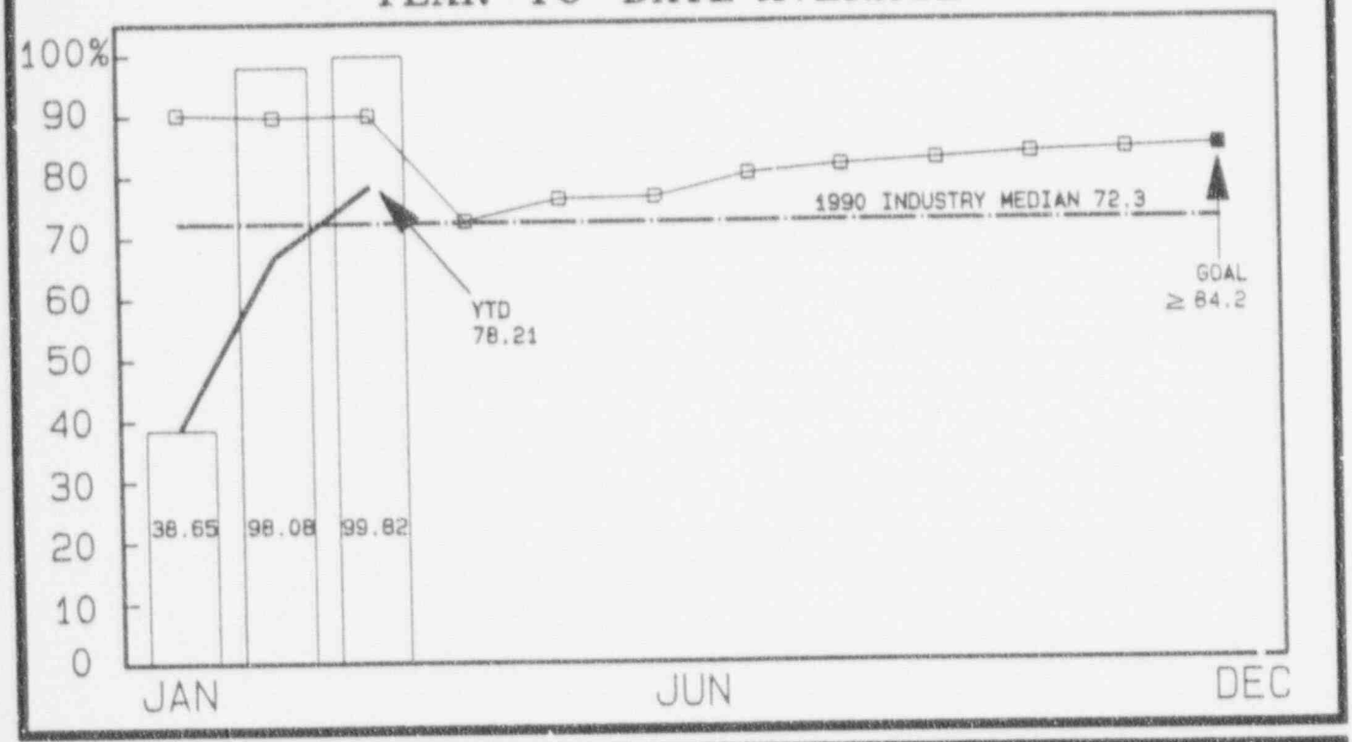



## THREE YEAR MOVING AVERAGE

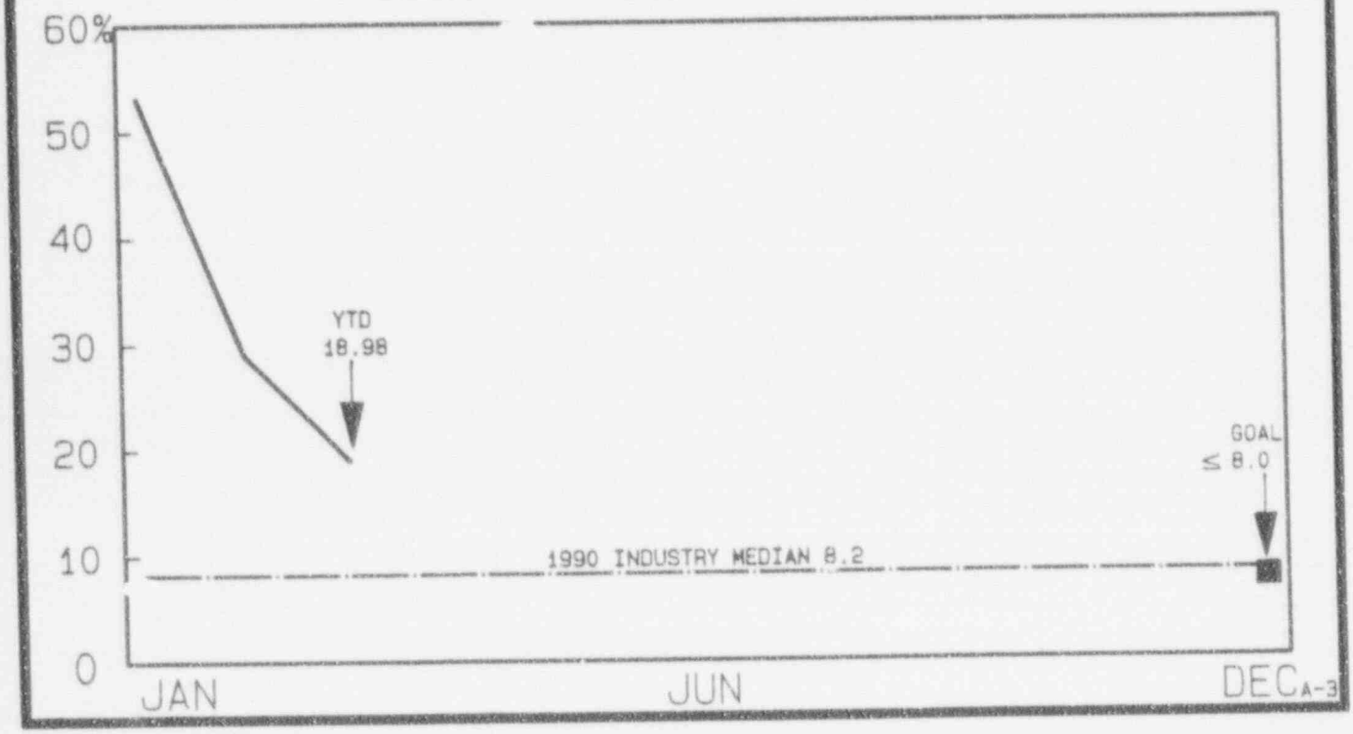


# ANO-1

## UNIT CAPABILITY FACTOR
## YEAR-TO-DATE AVERAGE




## UNPLANNED CAPABILITY LOSS FACTOR
## YEAR-TO-DATE AVERAGE



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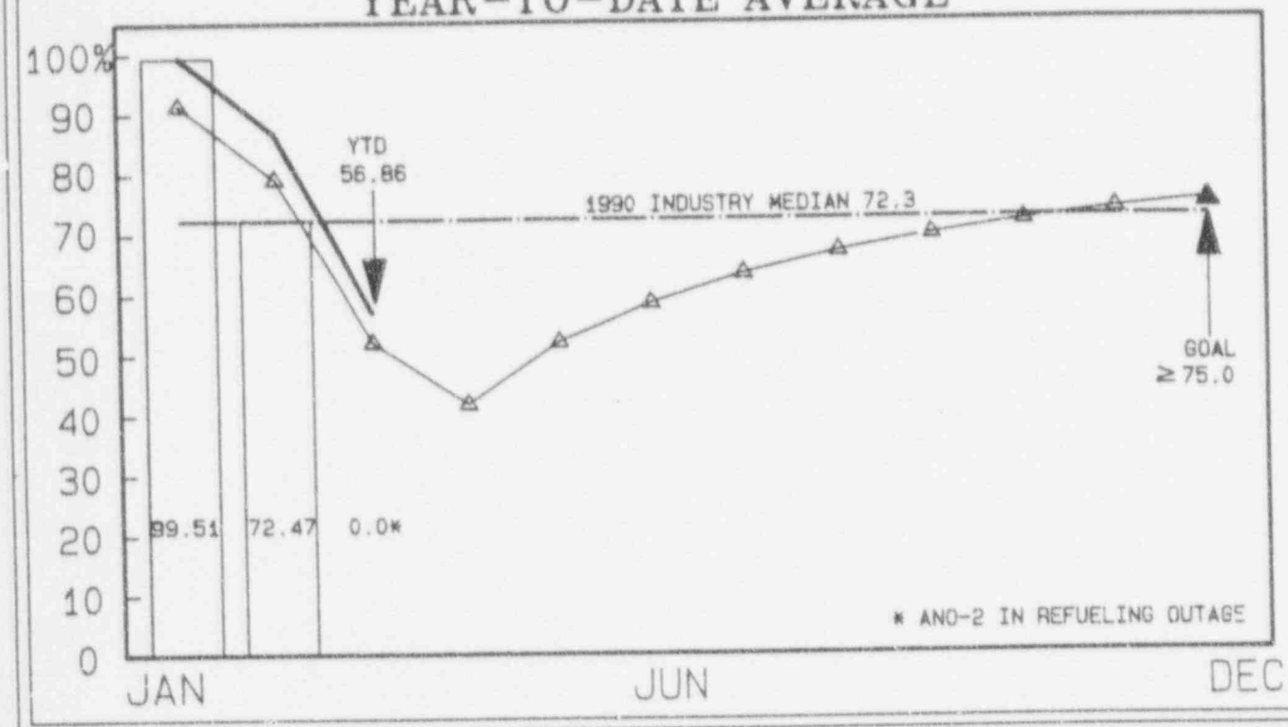
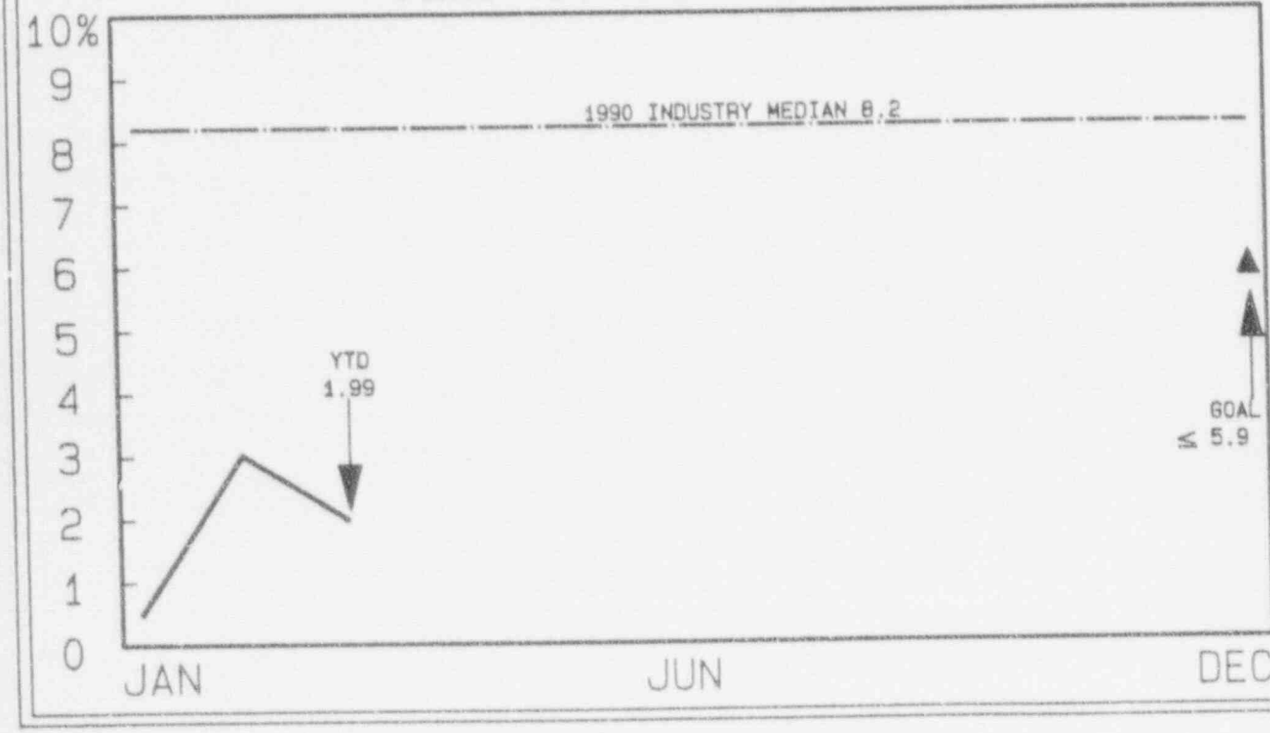
## UNIT CAPABILITY FACTOR
## YEAR-TO-DATE AVERAGE



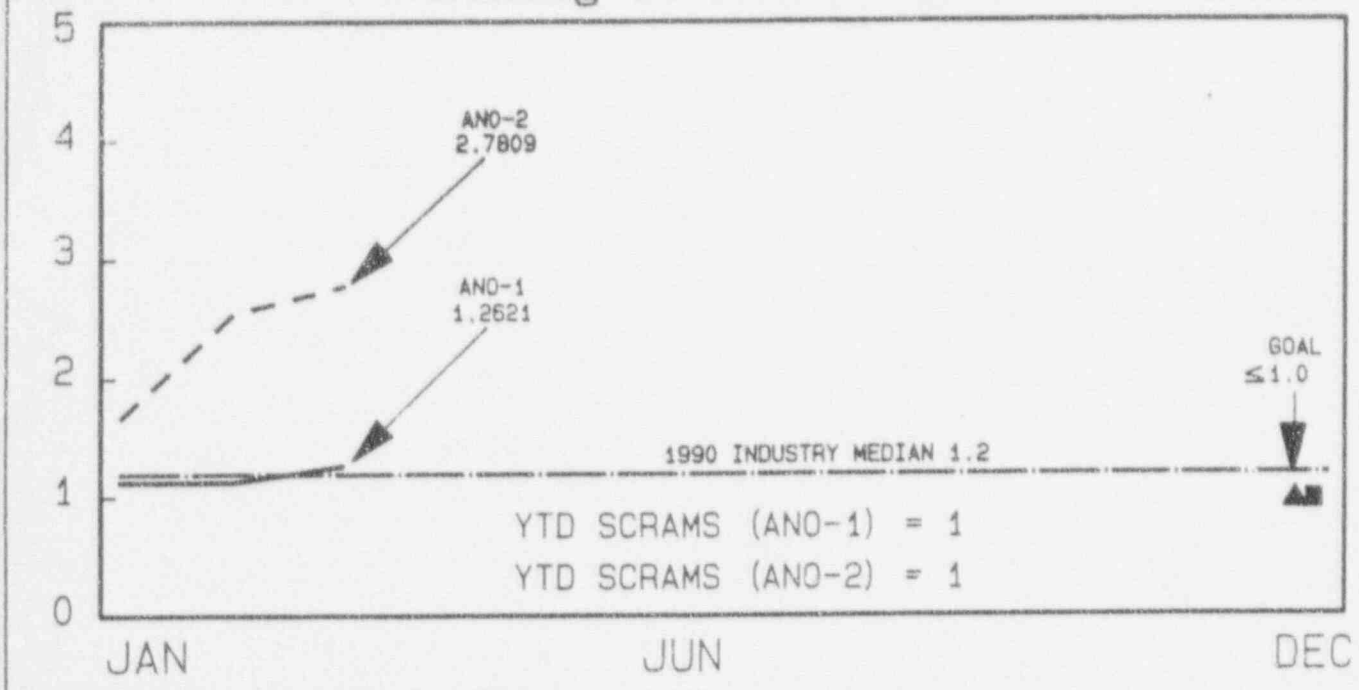

## UNPLANNED CAPABILITY LOSS FACTOR
## YEAR-TO-DATE AVERAGE



# ANO

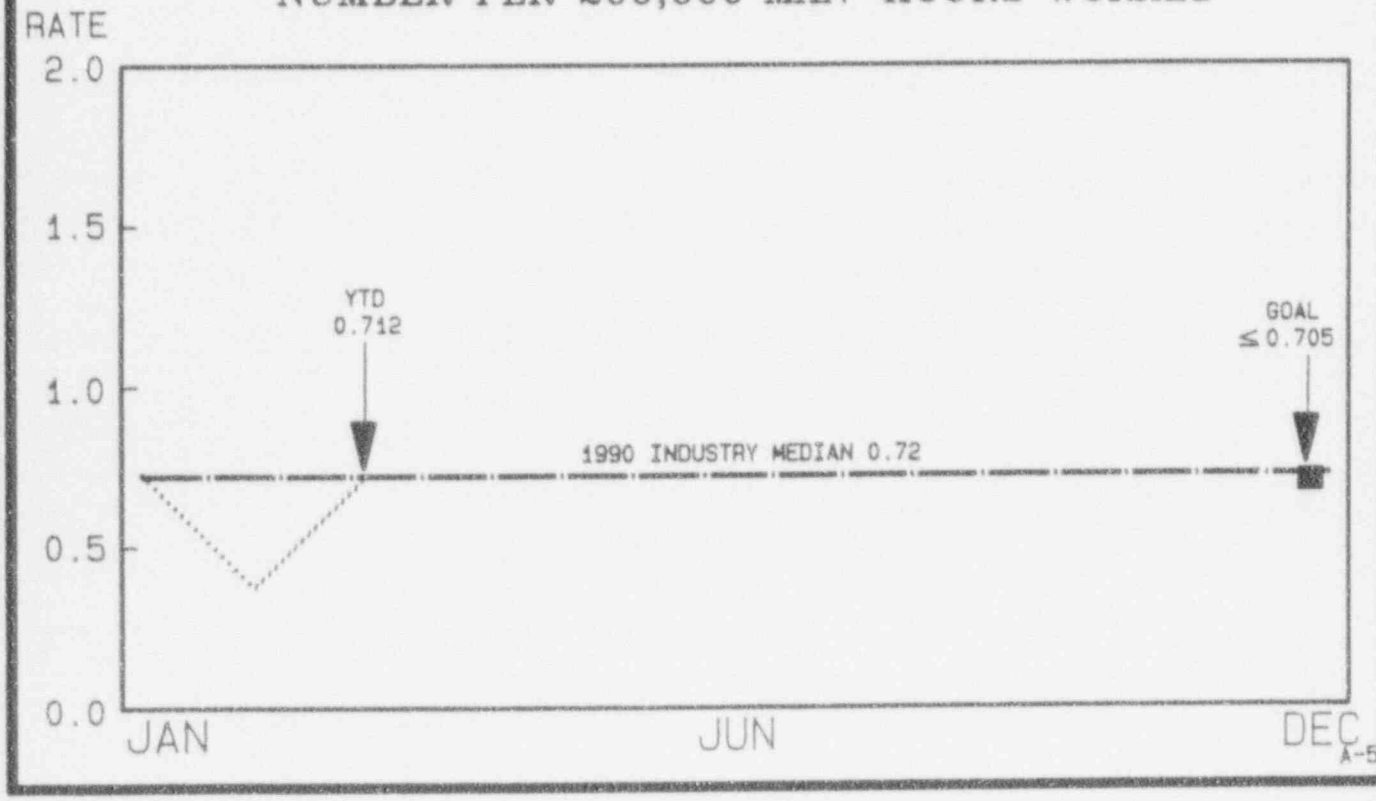
## SCRAMS PER 7000 CRITICAL HOURS

## Rolling 12 Month Value

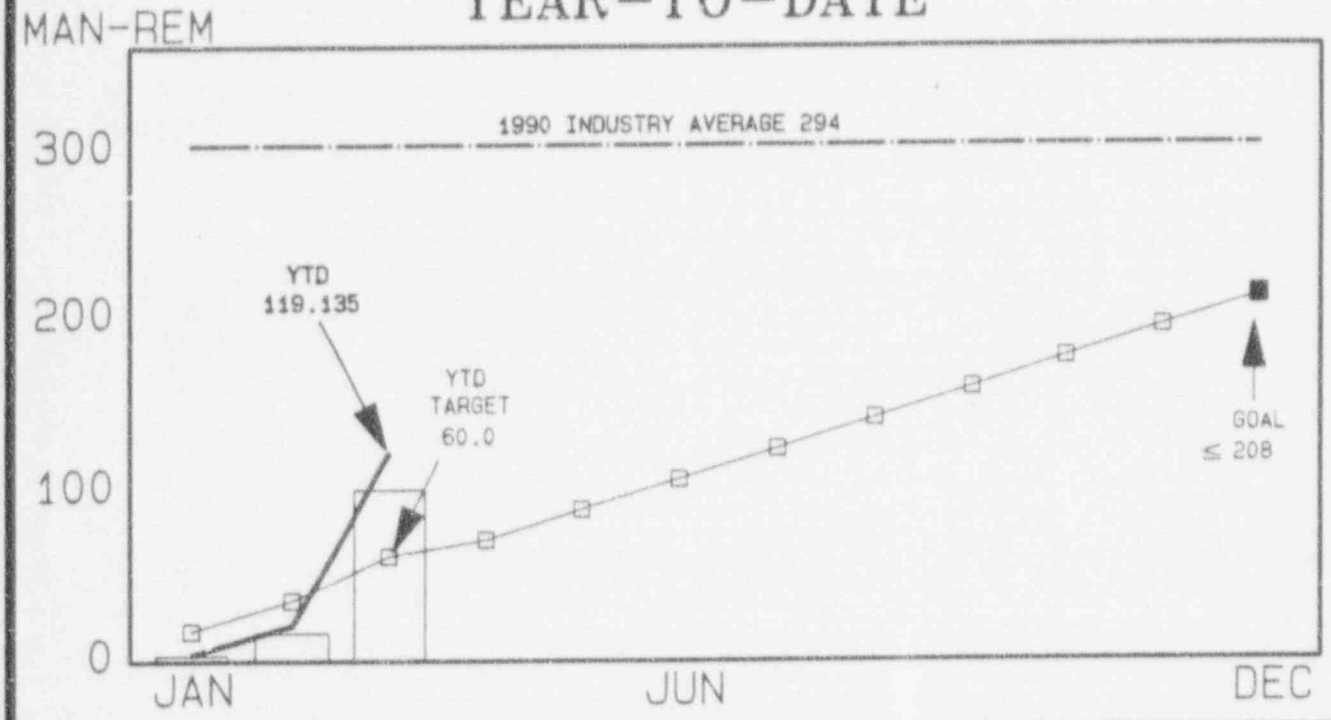



## YTD INDUSTRIAL SAFETY ACCIDENT RATE

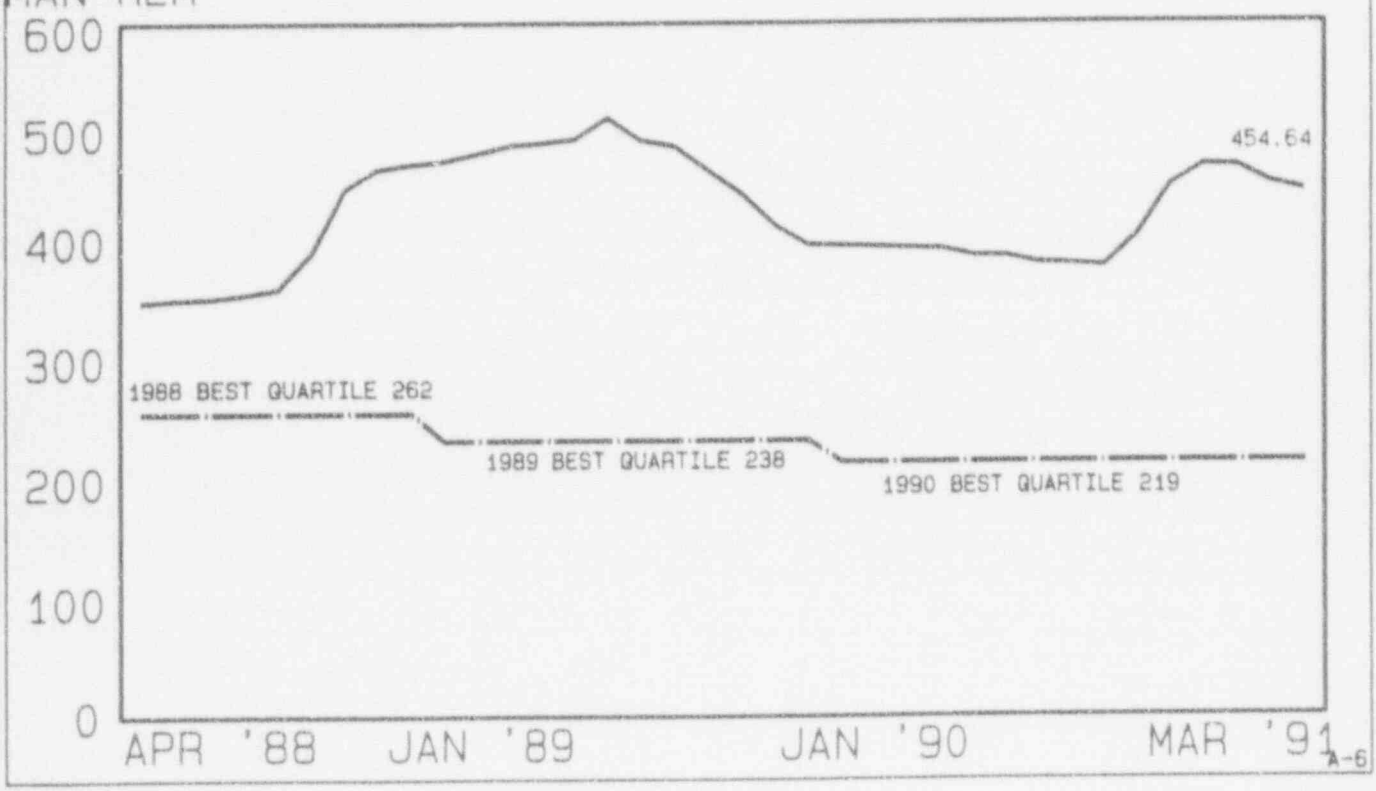
## NUMBER PER 200,000 MAN-HOURS WORKED



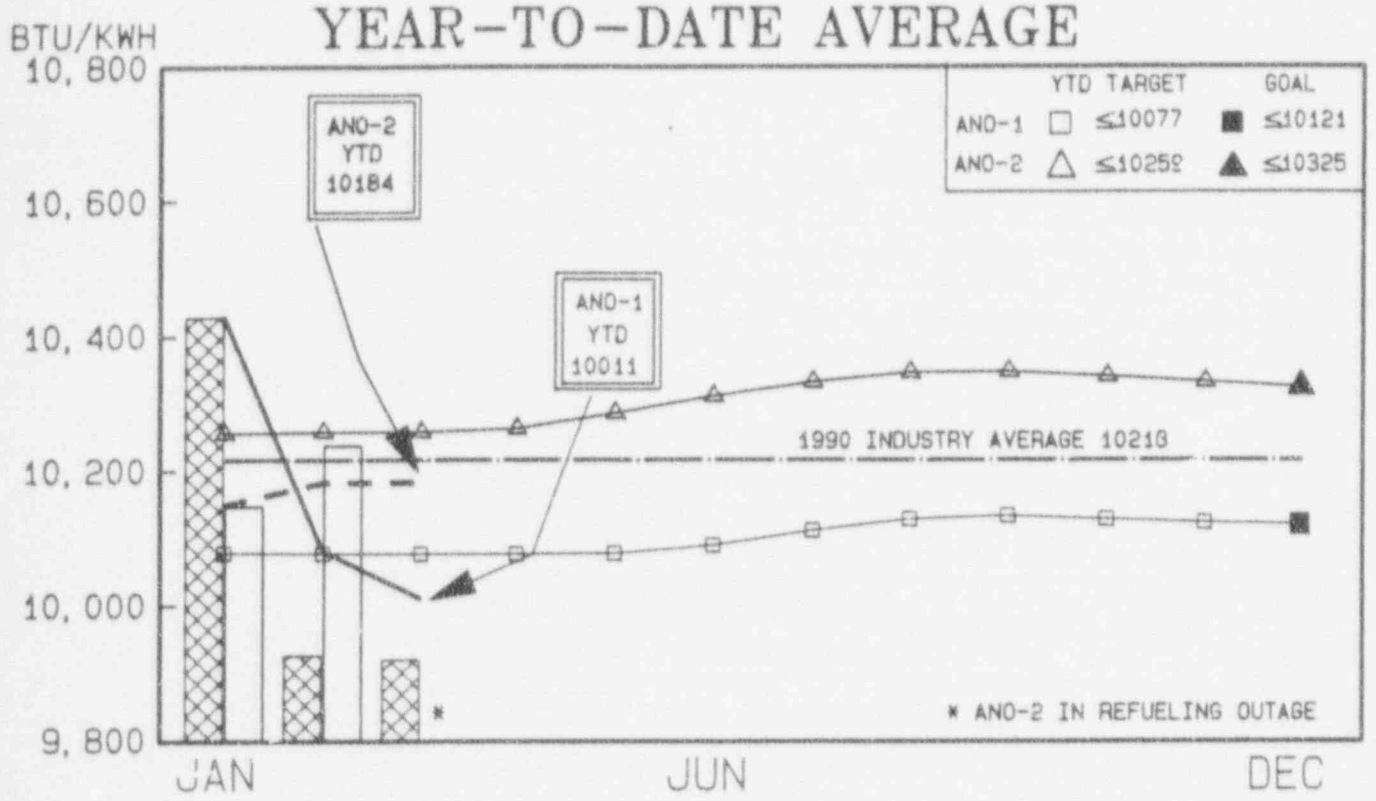
# ANO
# RADIATION DOSE PER UNIT
# YEAR-TO-DATE



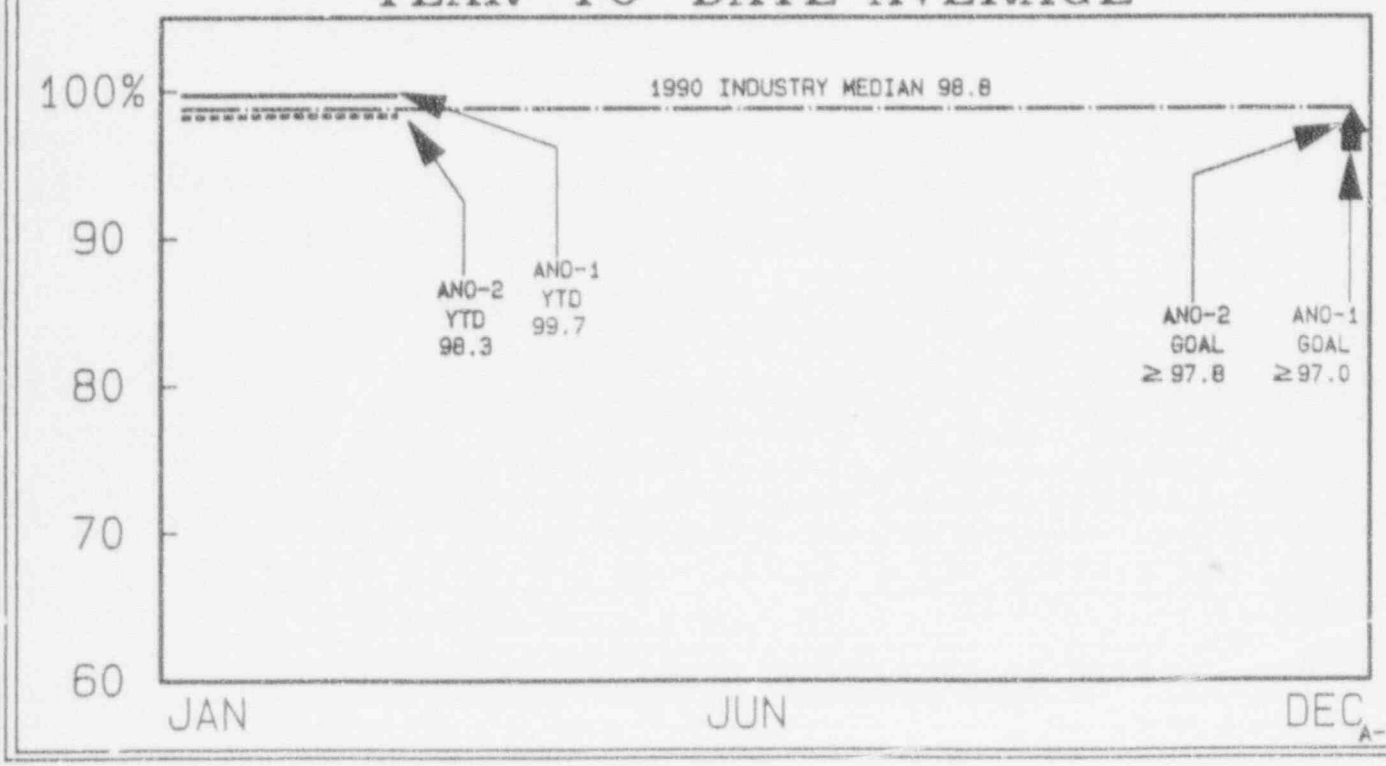

# THREE YEAR MOVING AVERAGE



# ANO
# UNIT GROSS HEAT RATE
# YEAR-TO-DATE AVERAGE



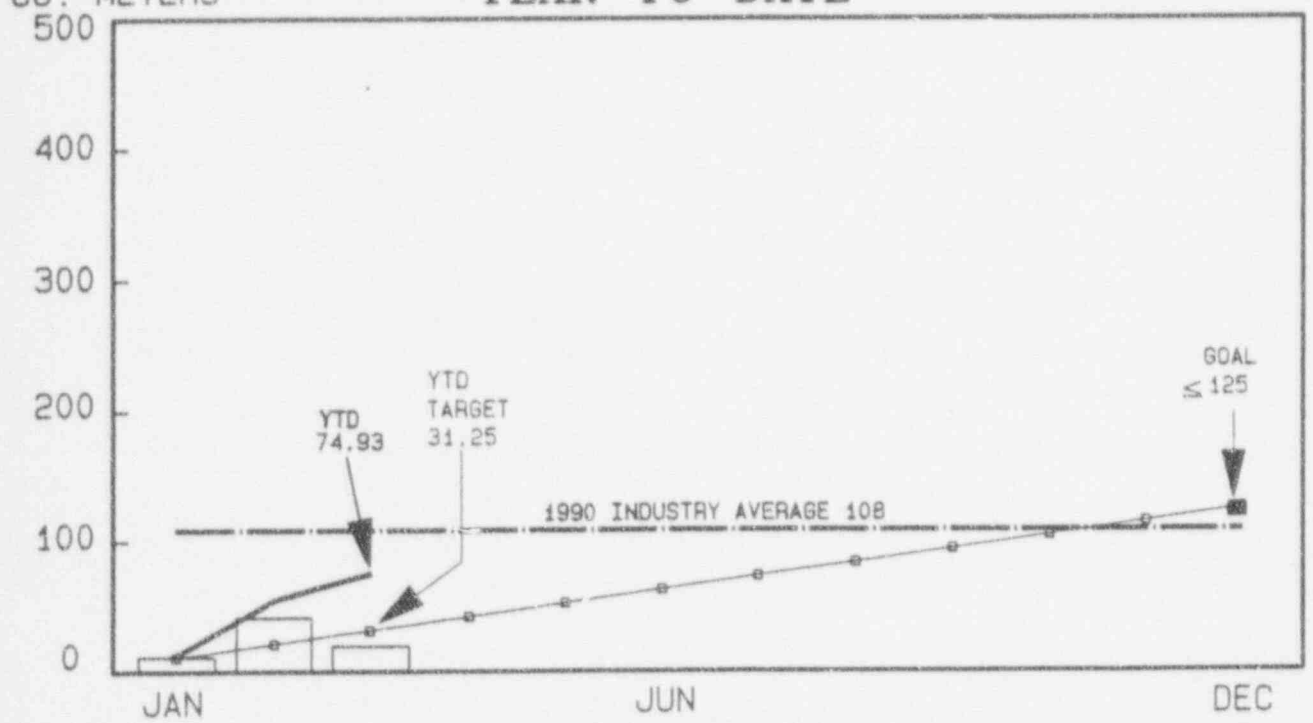

# THERMAL PERFORMANCE INDEX
# YEAR-TO-DATE AVERAGE



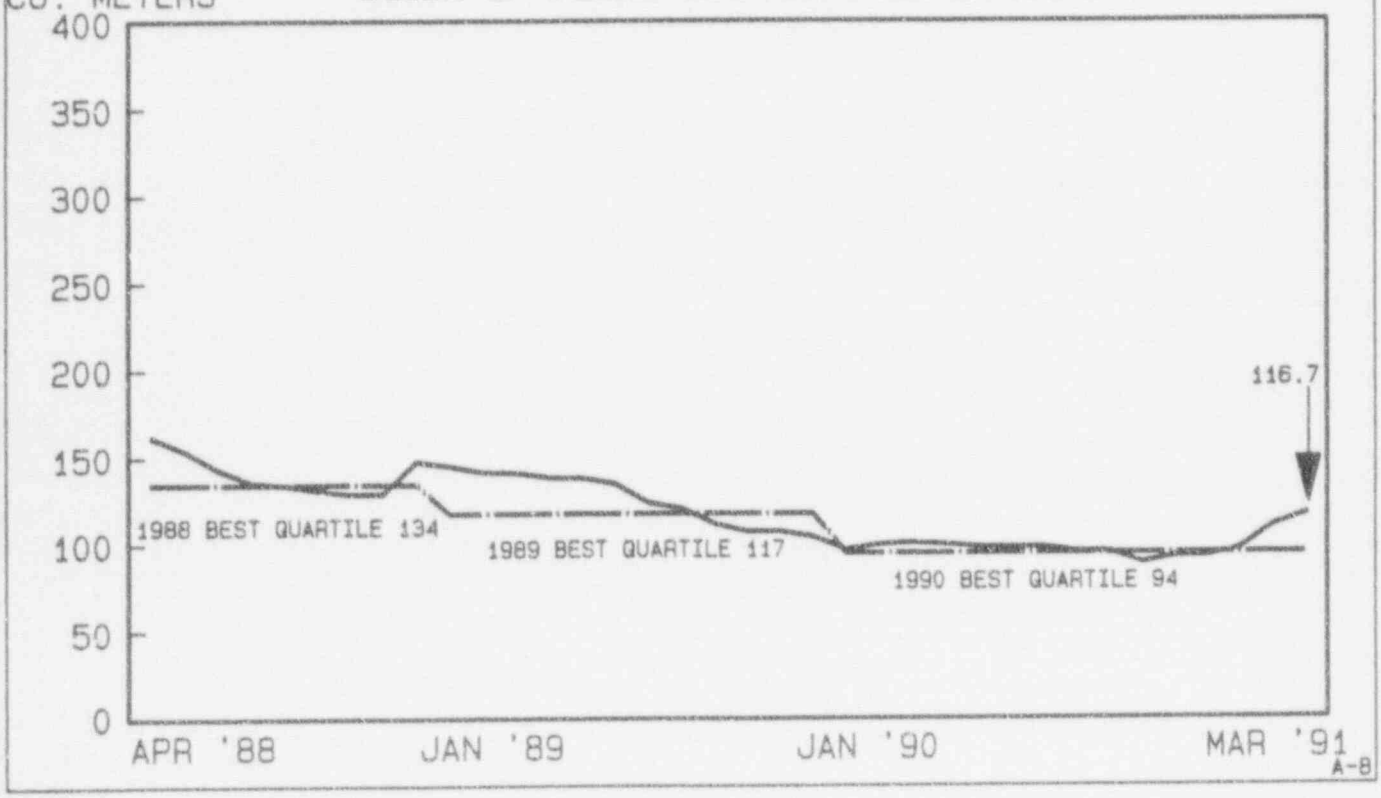
# ANO

## VOLUME OF LOW LEVEL WASTE PER UNIT

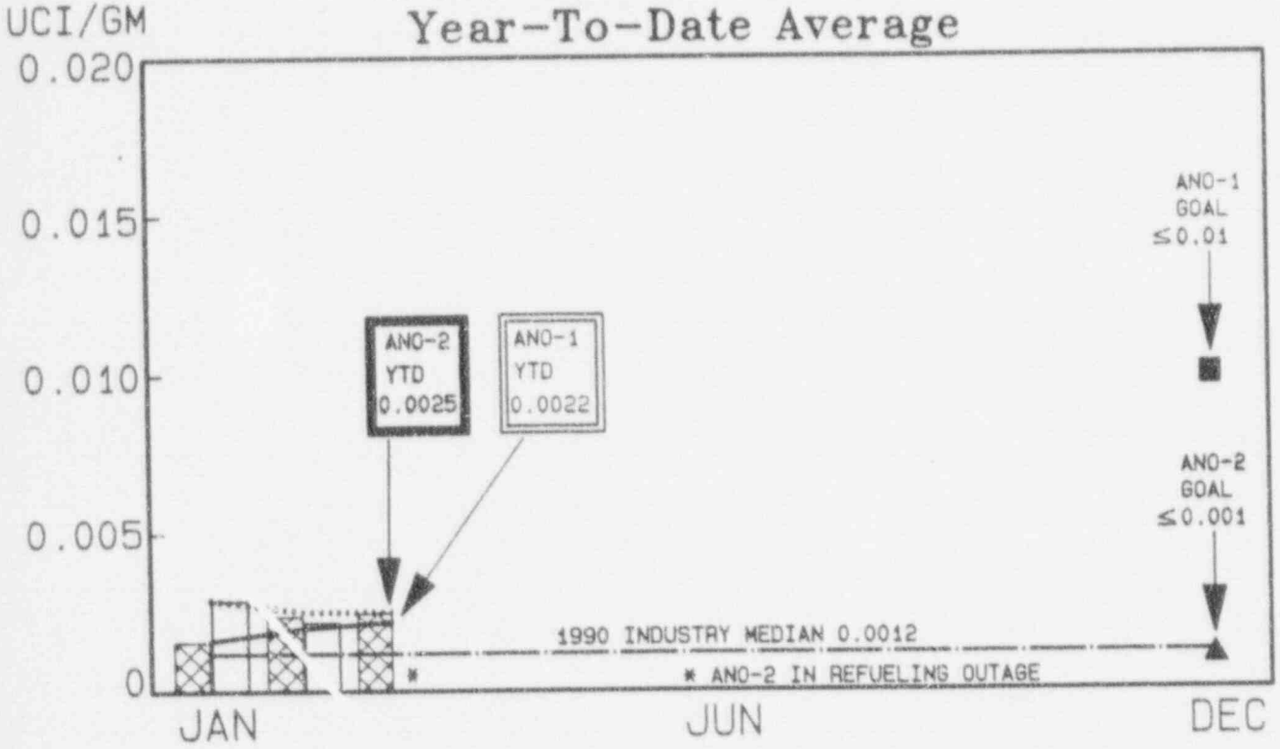
## YEAR-TO-DATE



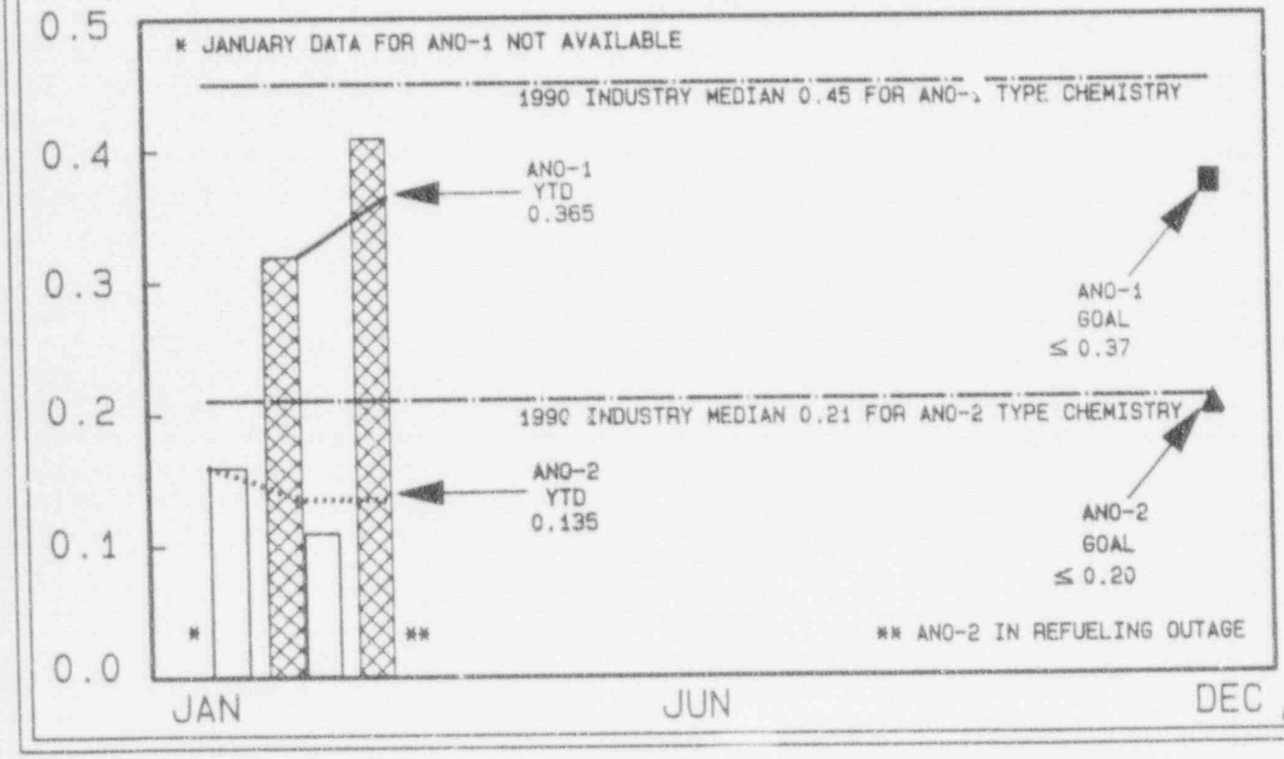

## THRFE YEAR MOVING AVERAGE



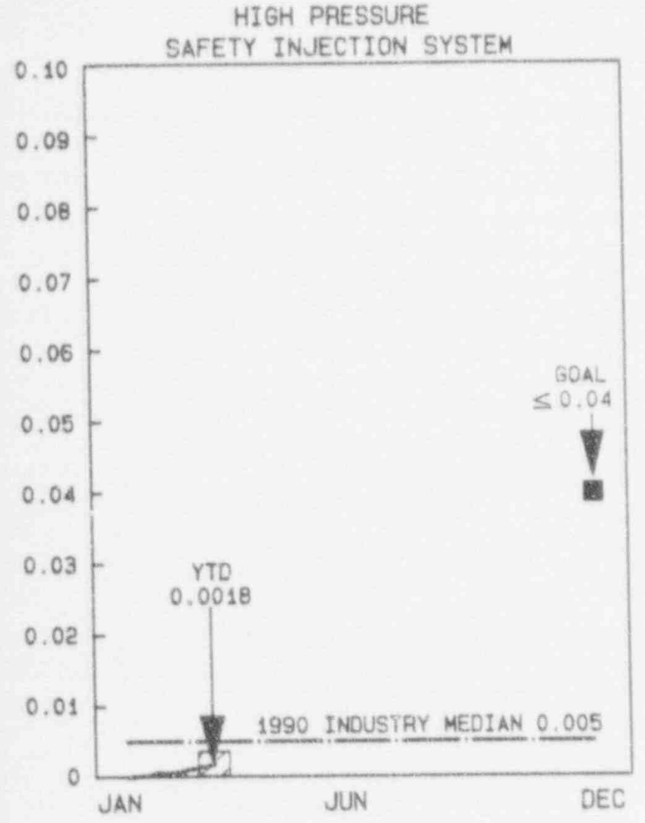
# ANO
## FUEL RELIABILITY INDEX
### Year-To-Date Average

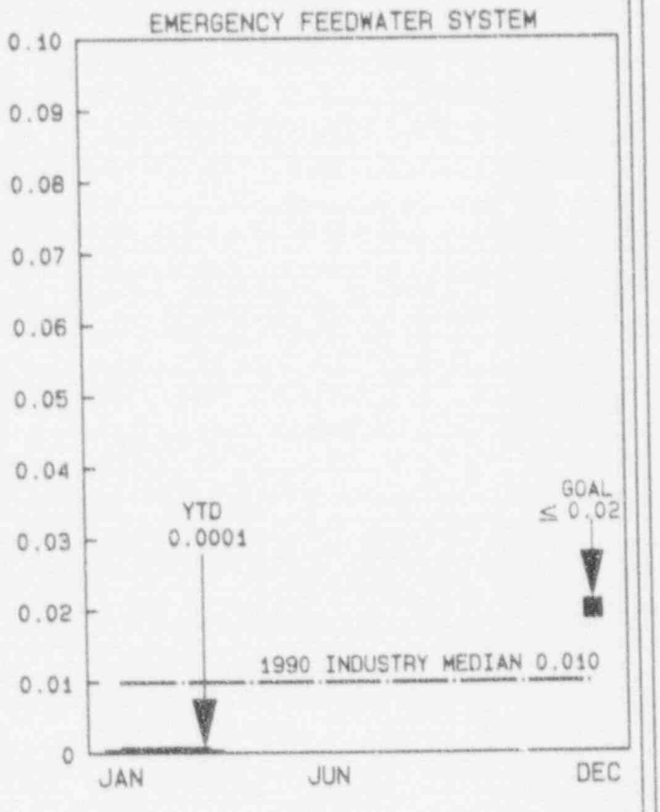


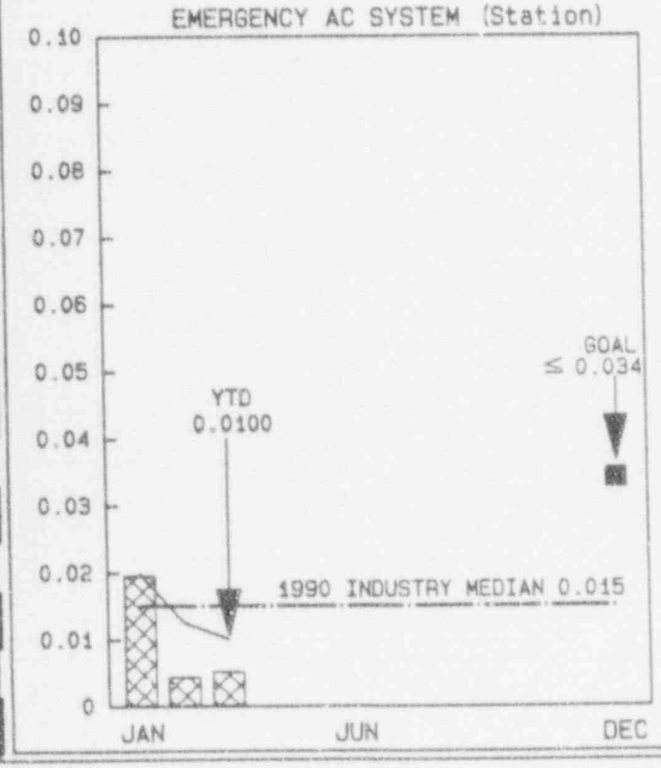

## CHEMISTRY INDEX
### Year-To-Date Average



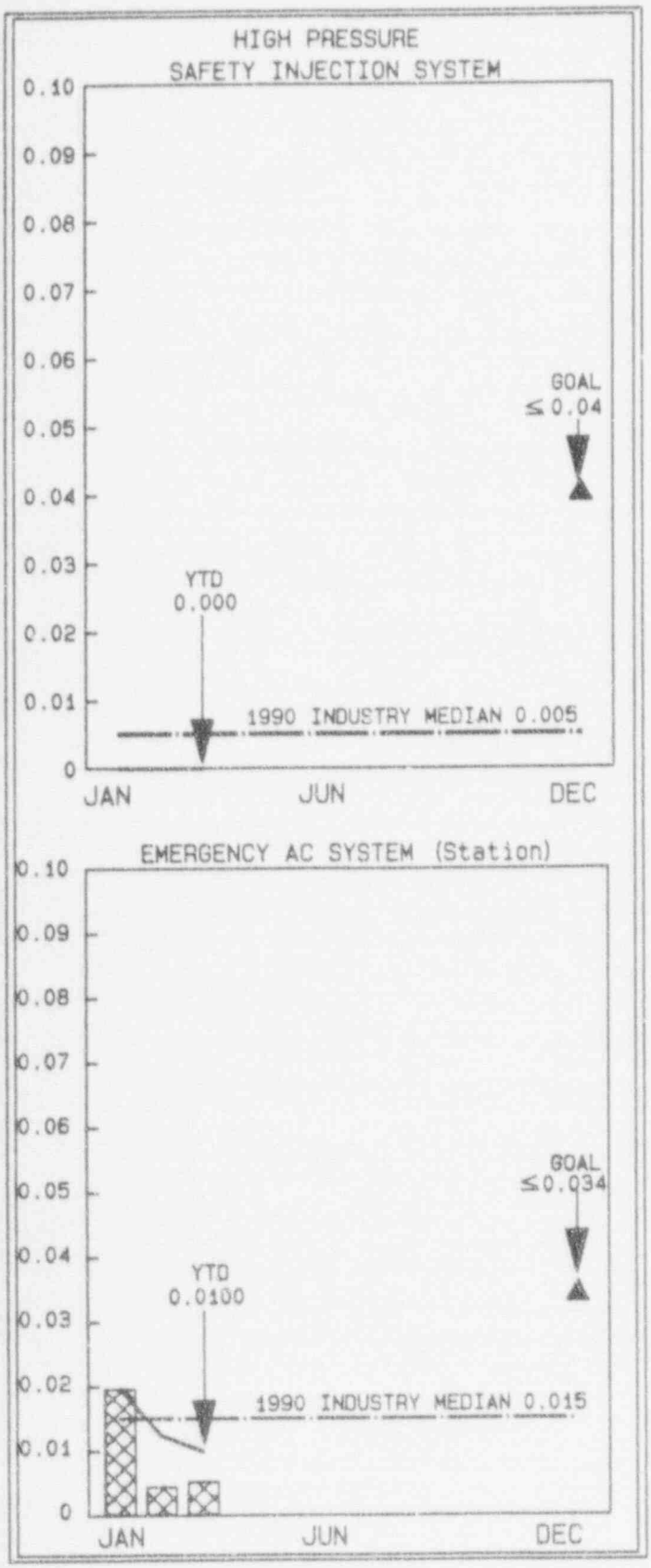
# ANO-1
# SAFETY SYSTEM PERFORMANCE
# Y-T-D

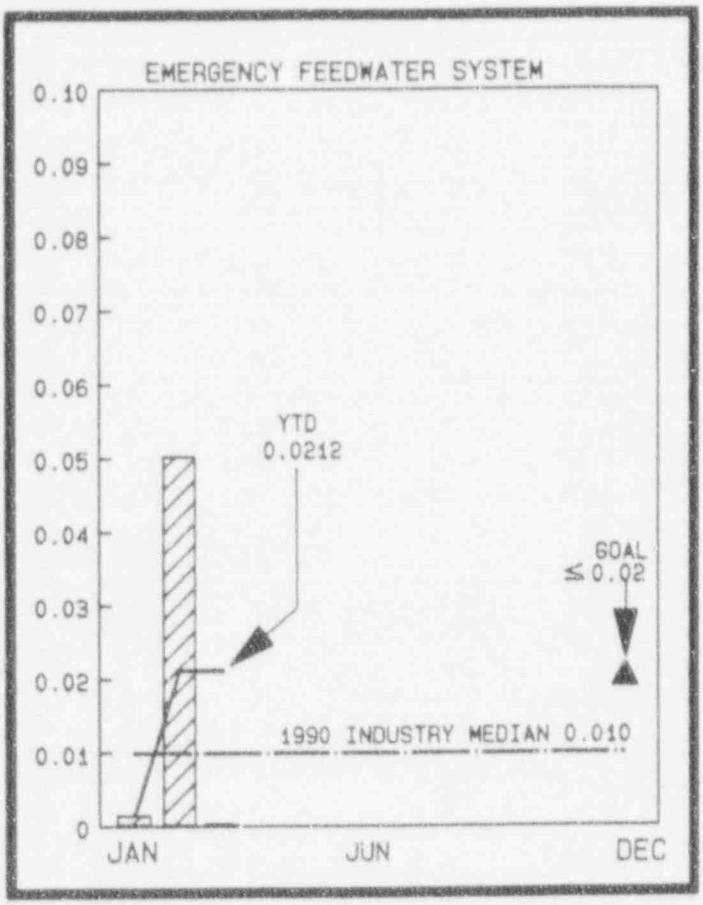


# ANO-2
# SAFETY SYSTEM PERFORMANCE
# Y-T-D



GRAND GULF NUCLEAR STATION
STATUS REPORT
(PLANT STATUS AS OF 04/10/91)

UNIT 1

## A. PLANT STATUS

Operating at approximately 100% power in Mode 1.

| | | 1991 YTD |
|---|---|---|
| o | Equivalent Availability (%) | 96.4 |
| o | Unit Capability Factor (%) | 96.2 |
| o | Capacity Factor (%) | 100.8 |
| o | Gross Heat Rate (BTUs/KWH) | 10505 |

o Consecutive Days On-Line - 5 (04/07/91 - 04/12/91)

## B. OPERATING SUMMARY SINCE LAST REPORT

o The plant had been on-line 96 consecutive days prior to a reactor scram which occurred on April 6, 1991, during the Automatic Turbine Testing Overspeed Trip Test Surveillance.

The scram was caused by a failure in a test circuit module. The module was replaced and the generator was synchronized at 1603 hours on April 7, 1991.

## C. SIGNIFICANT EVENTS

o None

## D. EXECUTIVE SUMMARY

Operator actions in response to the reactor scram and the subsequent recovery were well executed.

The INPO plant evaluation concluded on March 29. The formal exit meeting is scheduled for April 19. The NRC Region II Regional Administrator toured the plant on March 27. The Region II SALP Board Chairman for GGNS toured the plant on April 4 and 5. Comments on both visits were positive.

GRAND GULF NUCLEAR STATION
STATUS REPORT
(Status as of 04/08/91)

UNIT 2

## A. ACTIVITIES RELATED TO UNIT 2

- A request to the NRC for cancellation of Unit 2 Construction Permit was submitted.

# LICENSEE EVENT REPORT

Grand Gulf Nuclear Station

Grand Gulf Nuclear Station
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 1

**Title:** Delinquent LCO Action For Diesel Generator 11 Due To Personnel Error

**LER No:** 91-001

**Date Submitted:** February 13, 1991

A Technical Specification limiting condition of operation (LCO) action was not satisfied after the Division 1 diesel generator (DG 11) was removed from standby service to perform preventive maintenance. A Technical Specification LCO report had been generated on the previous shift when DG 11 was made inoperable. The replacement Shift Superintendent and Shift Supervisor were made aware of the required actions. The missed surveillance was due to personnel error by plant licensed operators. Inattention to detail was determined to be the cause of this event.

Station procedure 06-OP-1000-D-0001 was amended to incorporate a surveillance requirement trigger for the diesel generators. The procedure provides a method of completing and tracking surveillances required on a daily or more frequent schedule. The procedure change should prevent recurrence of similar delinquent diesel generator LCO actions.

The late verification of offsite A.C. power sources did not result in a compromise to plant safety. All emergency core cooling systems were operable and available to perform the required safety functions.

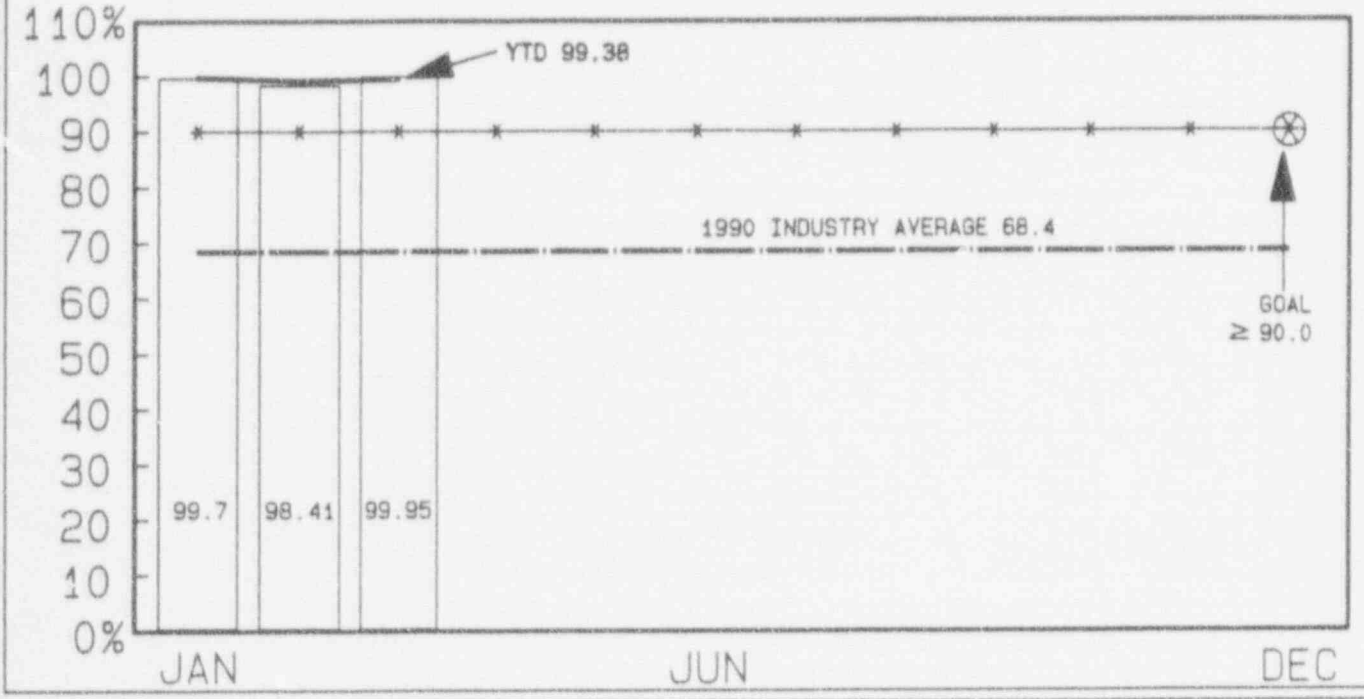
LEGEND

Graphs with a double line border denote indicators which are within the established goal.

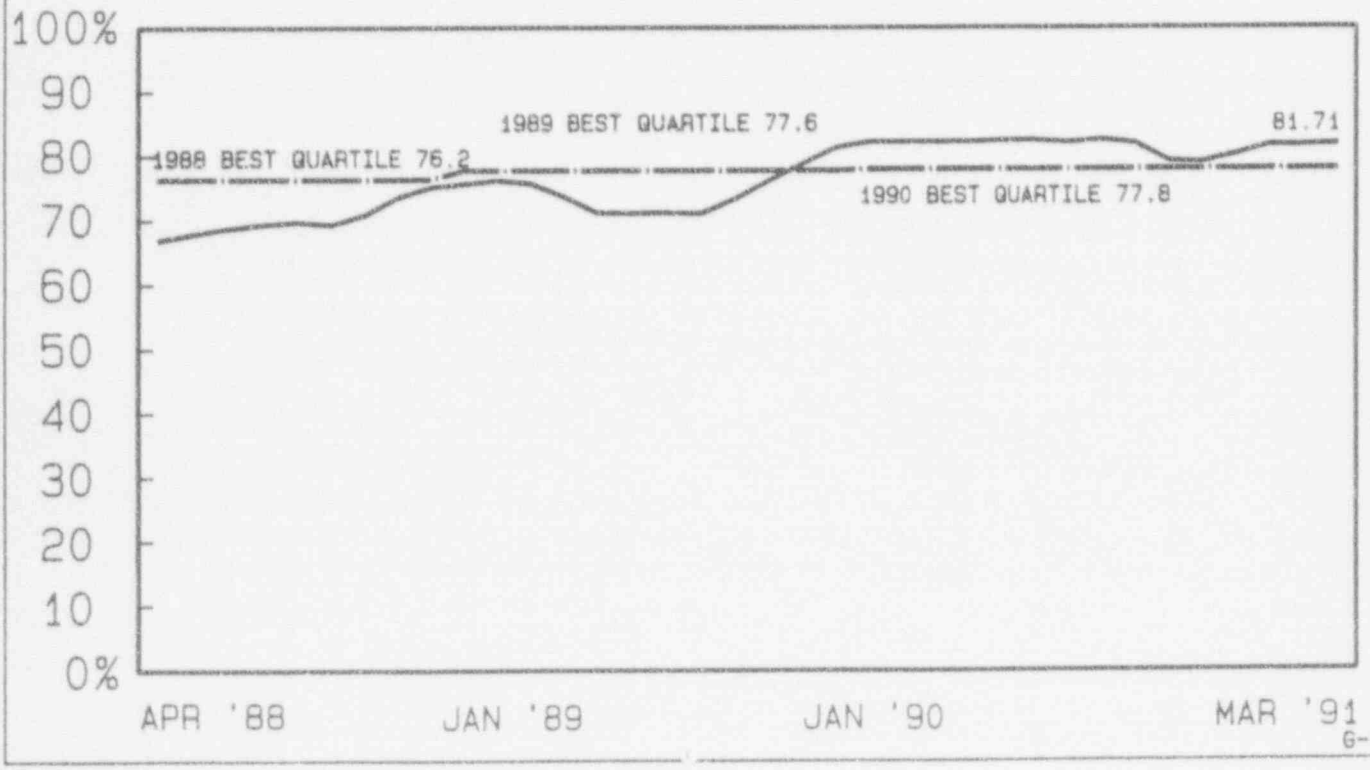
Graphs with a thick, single line border denote indicators which are outside the established goal.

Graphs with a thin, single line border denote indicators for which no goal has been established or for which a goal is not applicable.

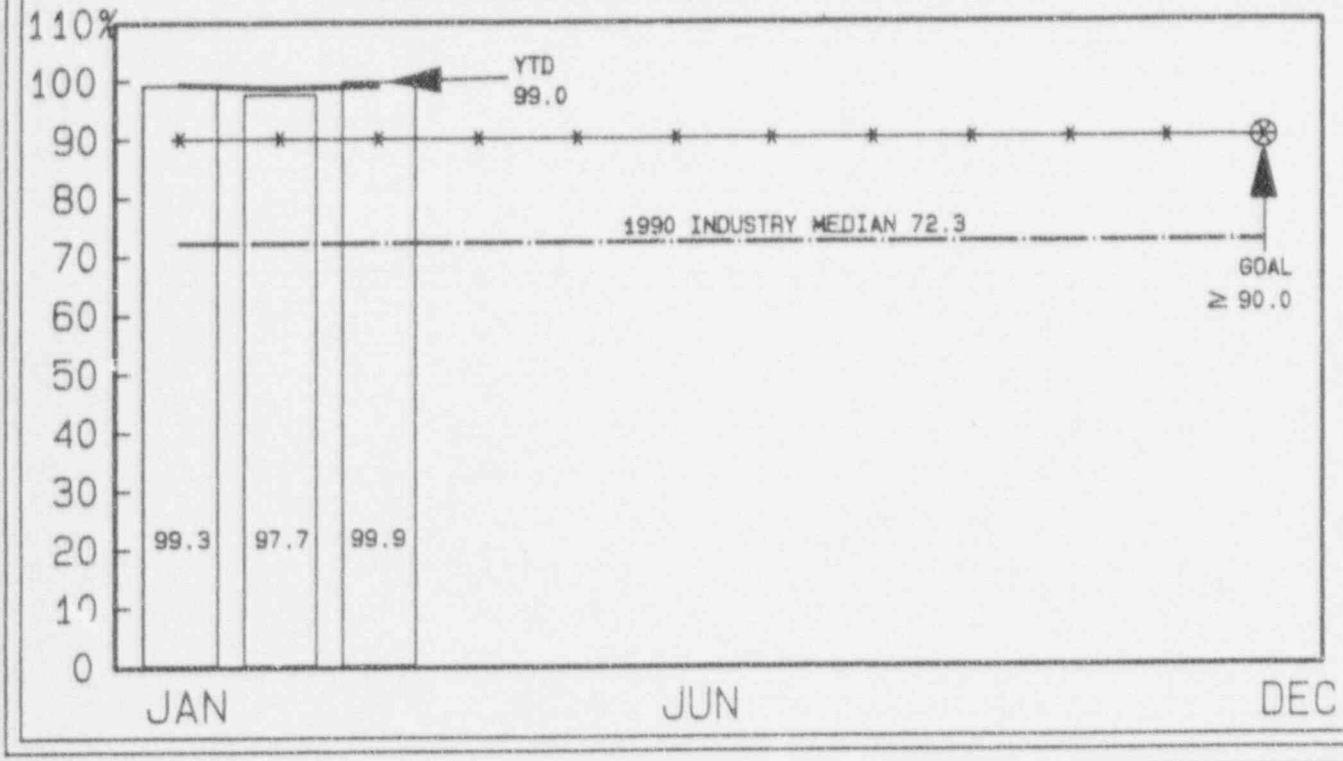
# GGNS
# UNIT EQUIVALENT AVAILABILITY
# YEAR-TO-DATE AVERAGE



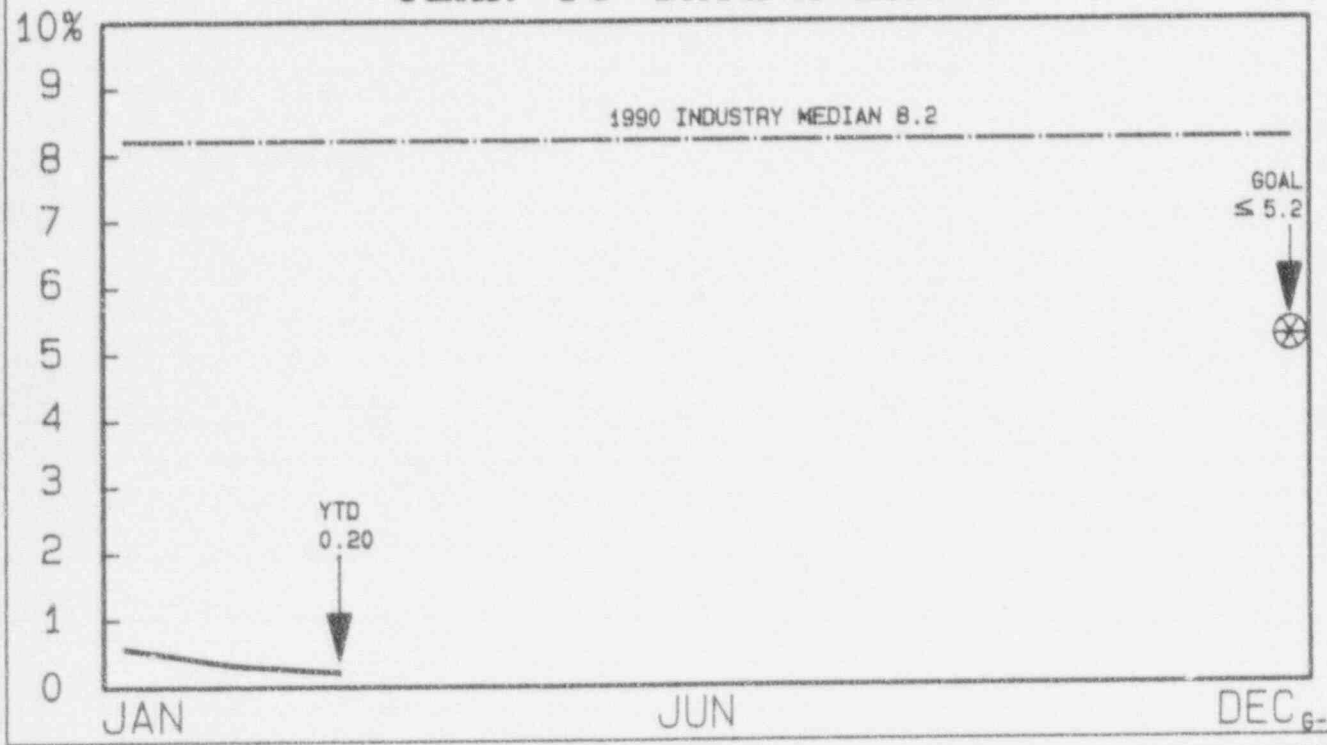

## THREE YEAR MOVING AVERAGE



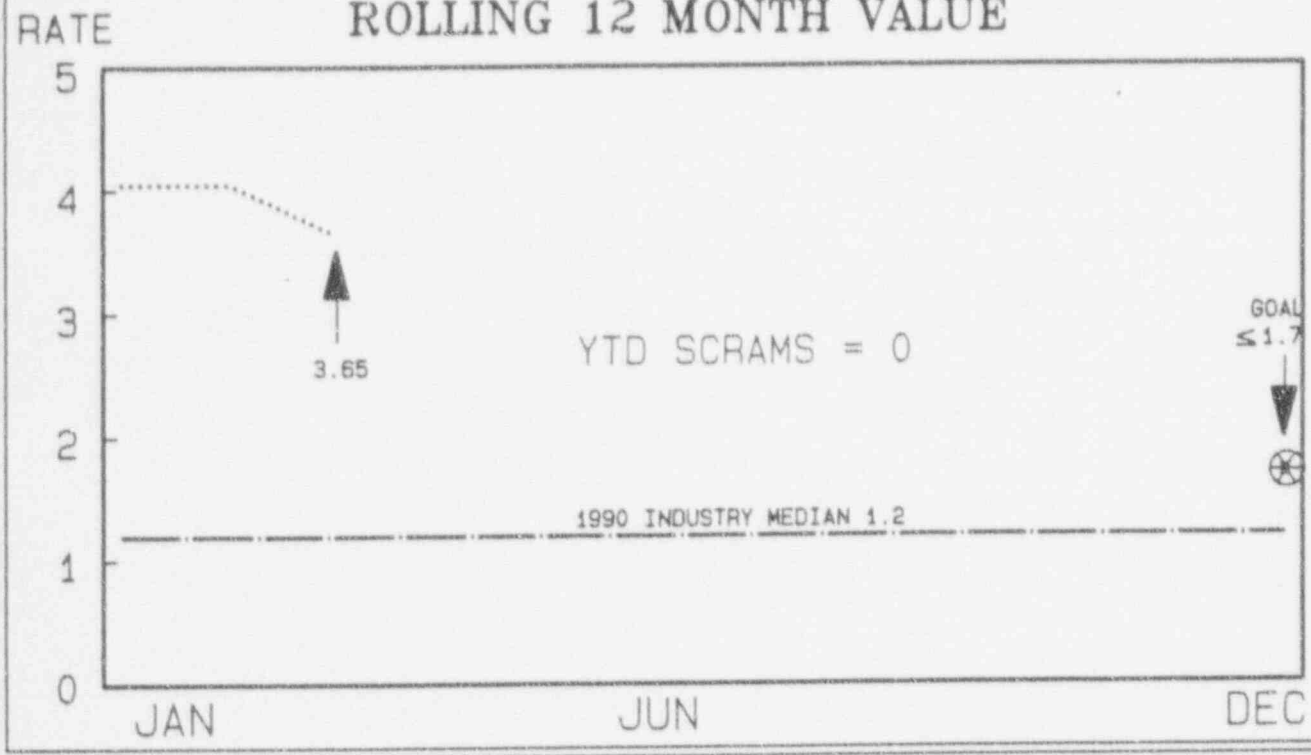
# GGNS
# UNIT CAPABILITY FACTOR
# YEAR-TO-DATE AVERAGE



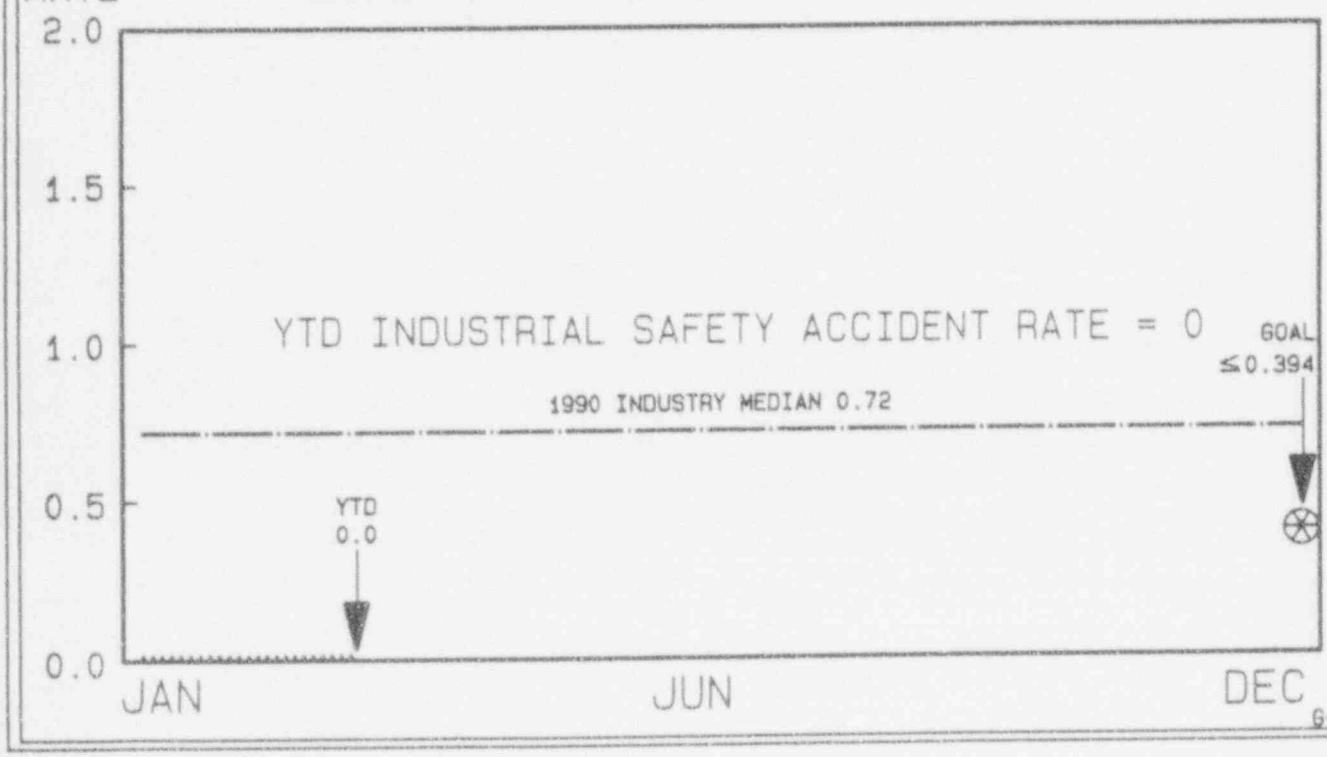

# UNPLANNED CAPABILITY LOSS FACTOR
# YEAR-TO--DATE AVERAGE



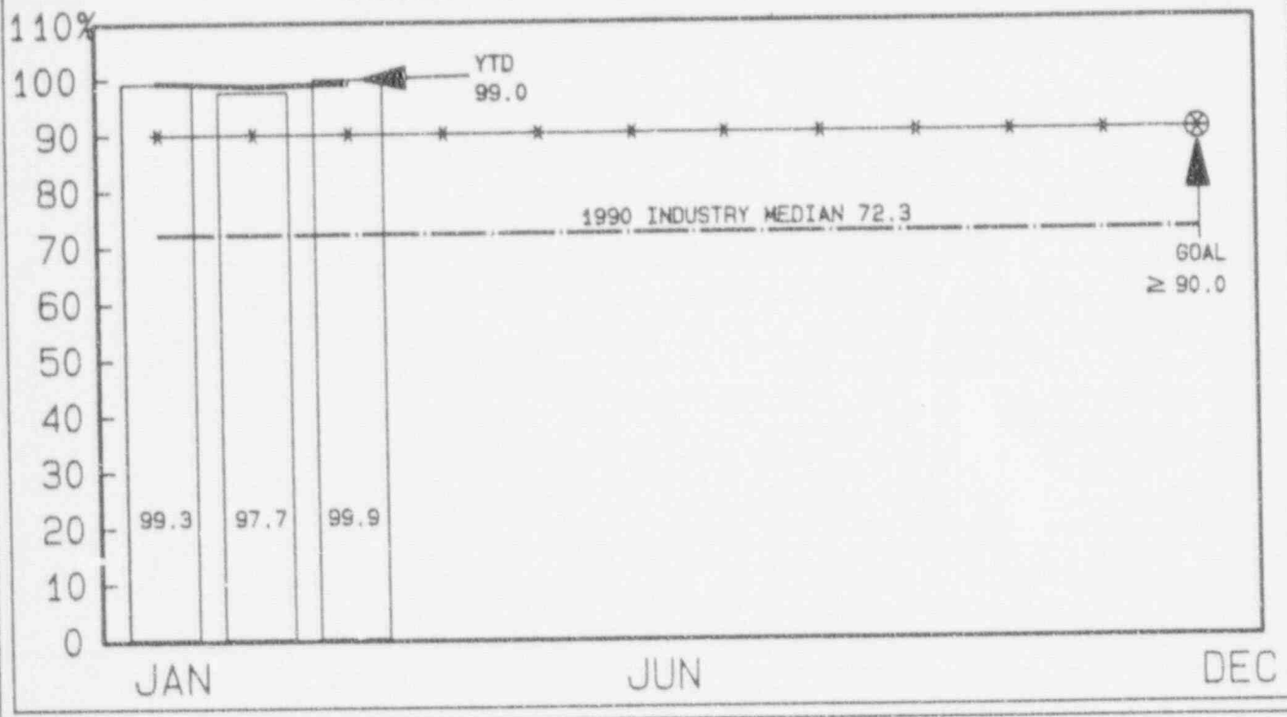
# GGNS
## SCRAMS PER 7000 CRITICAL HOURS
## ROLLING 12 MONTH VALUE



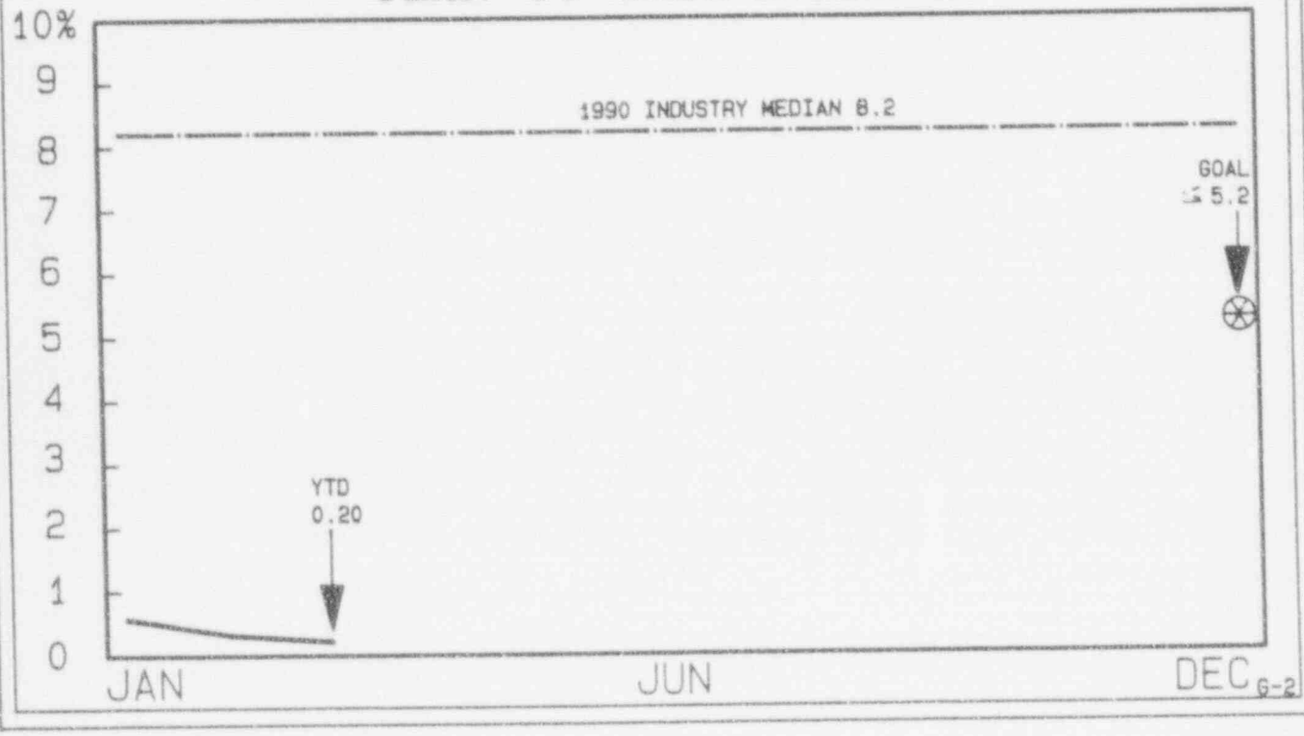

## YTD INDUSTRIAL SAFETY ACCIDENT RATE
### NUMBER PER 200,000 MAN-HOURS WORKED



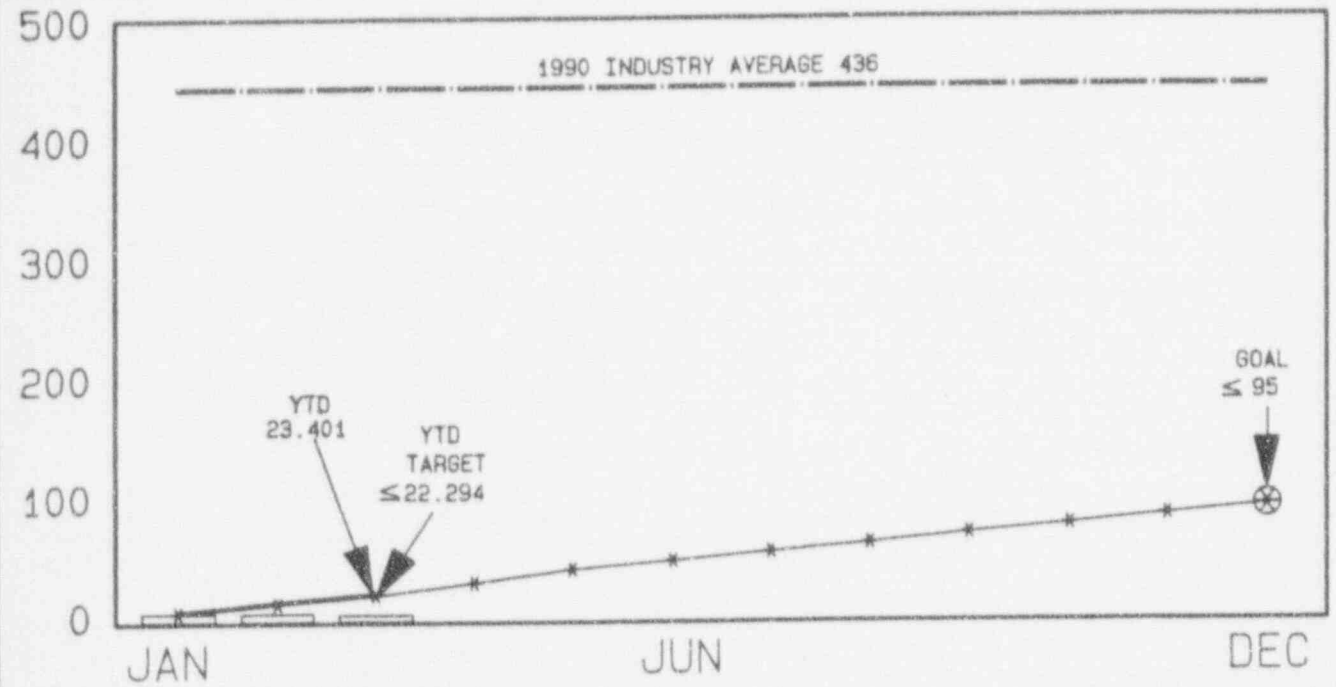
# GGNS
# UNIT CAPABILITY FACTOR
# YEAR-TO-DATE AVERAGE



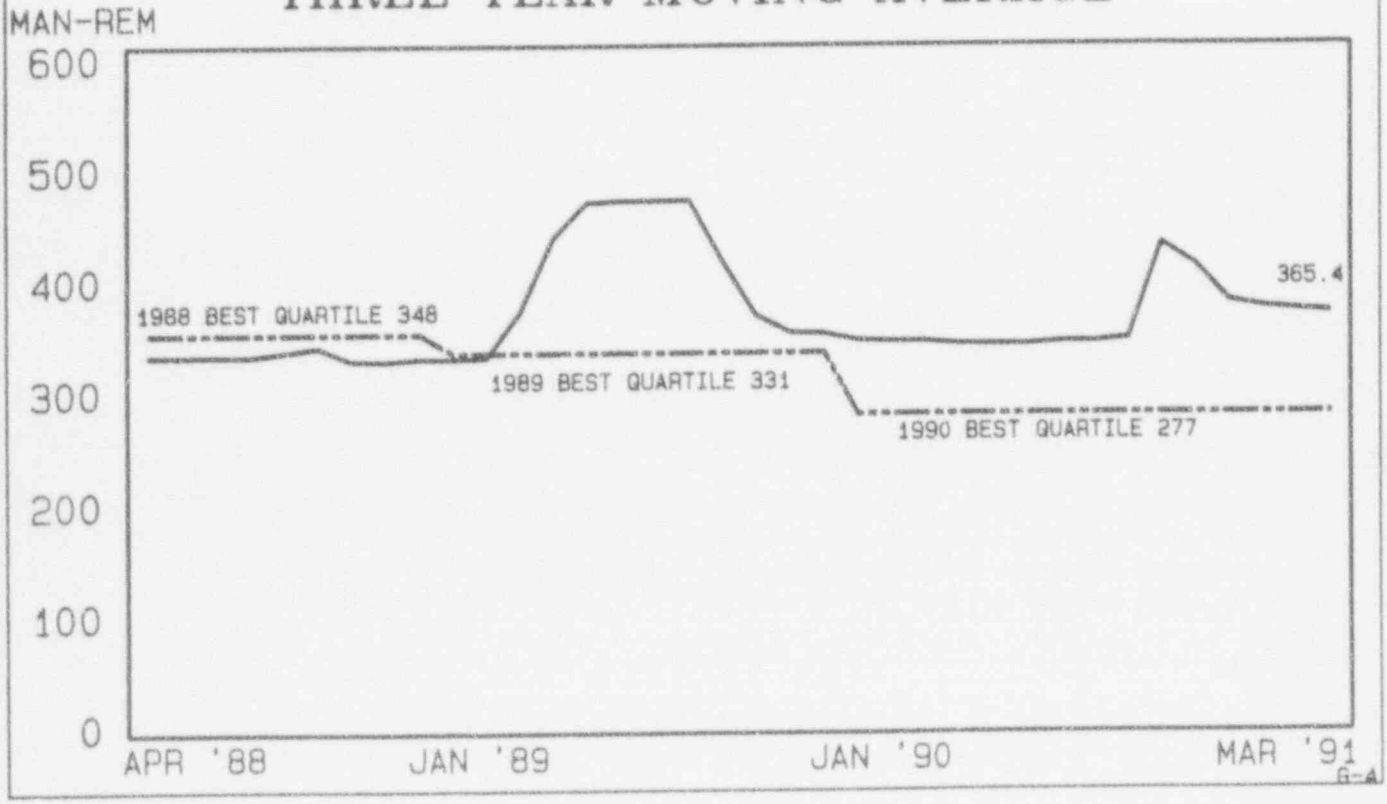

# UNPLANNED CAPABILITY LOSS FACTOR
# YEAR-TO-DATE AVERAGE



# GGNS
# RADIATION DOSE
# YEAR–TO–DATE

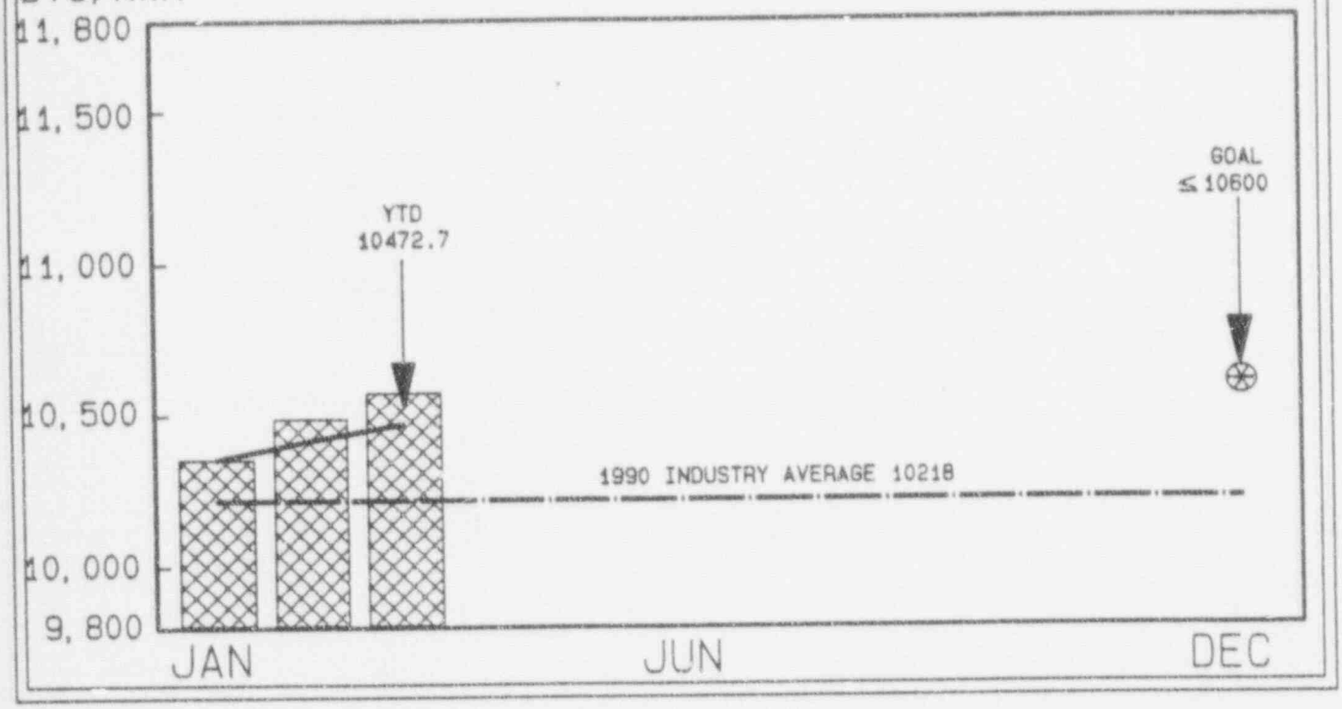



# THREE YEAR MOVING AVERAGE

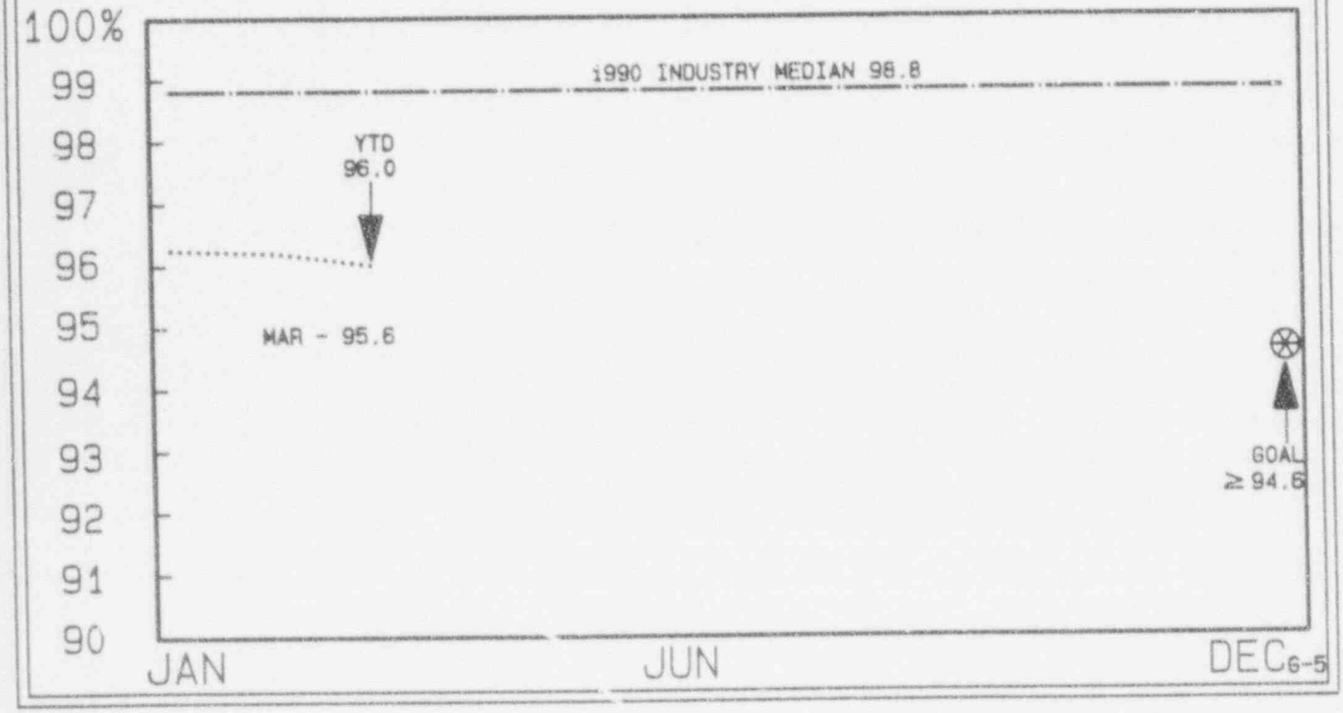



G-4

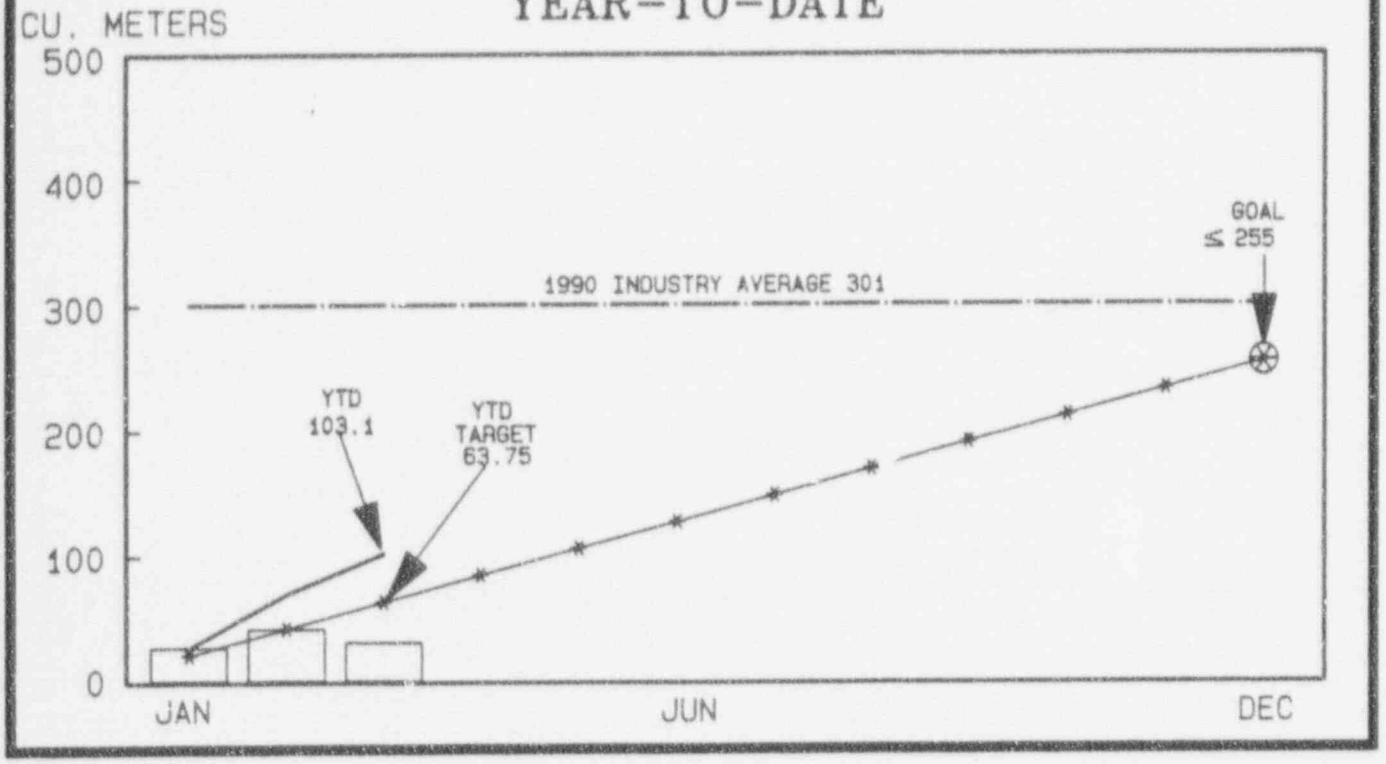
# GGNS
# UNIT GROSS HEAT RATE
# YEAR-TO-DATE AVERAGE



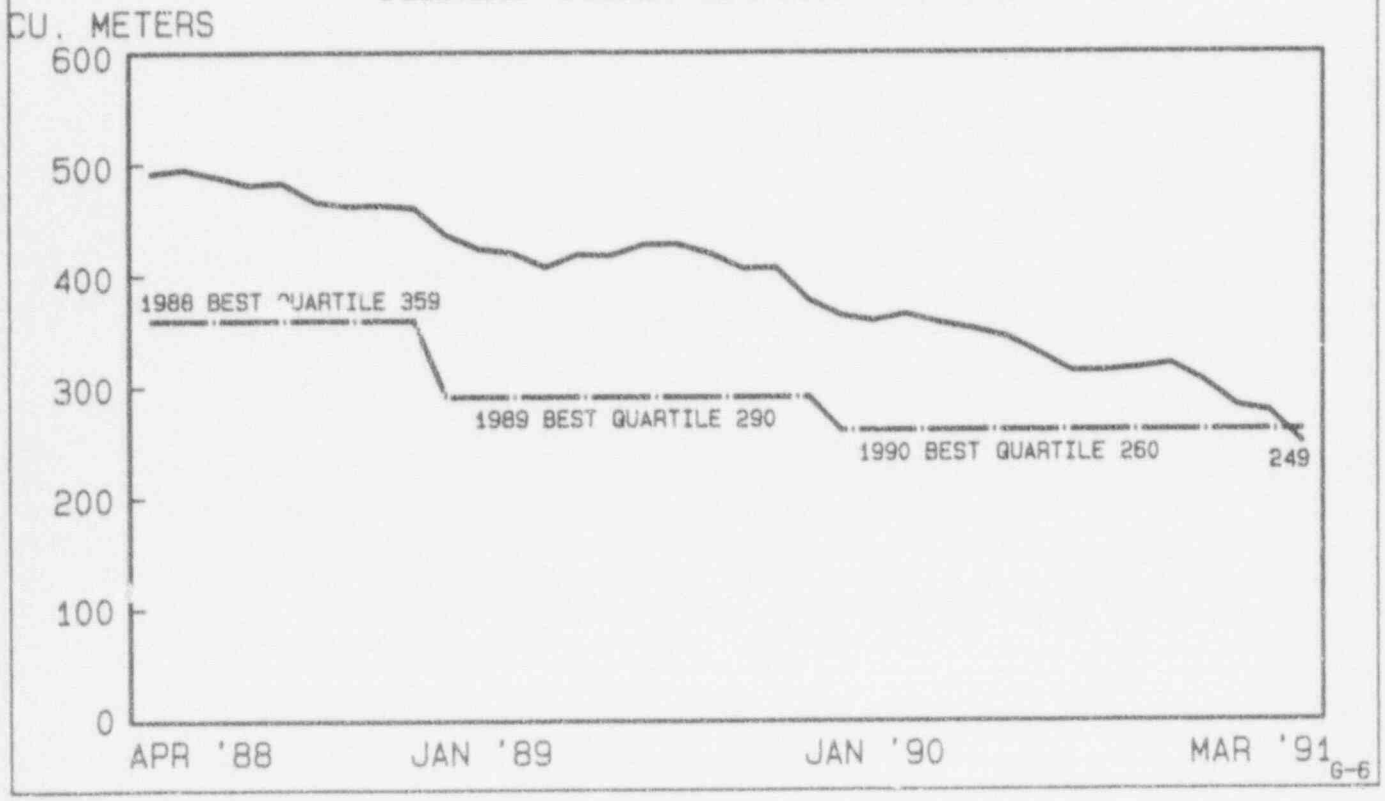

# THERMAL PERFORMANCE INDEX
# YEAR-TO-DATE AVERAGE

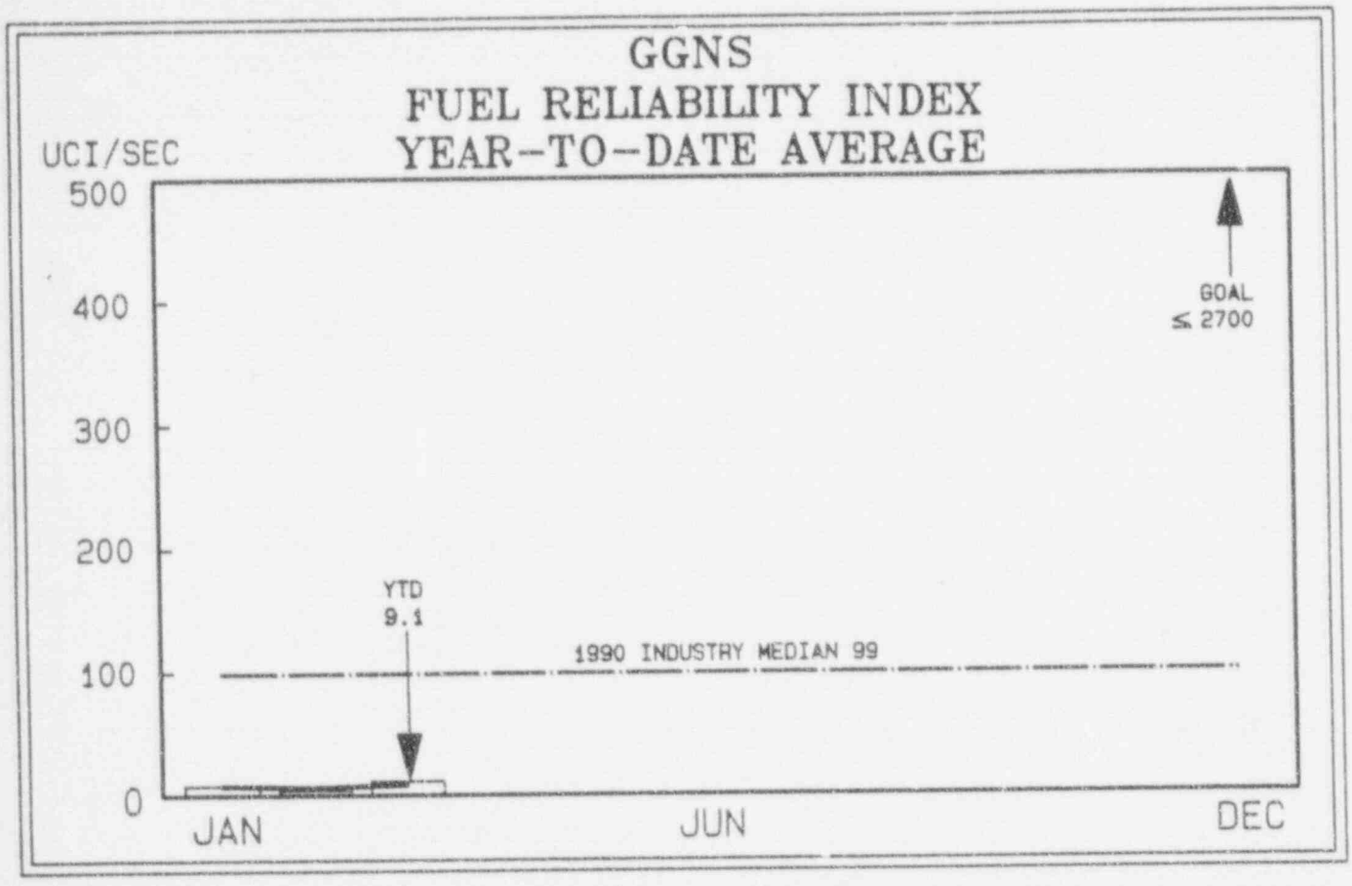


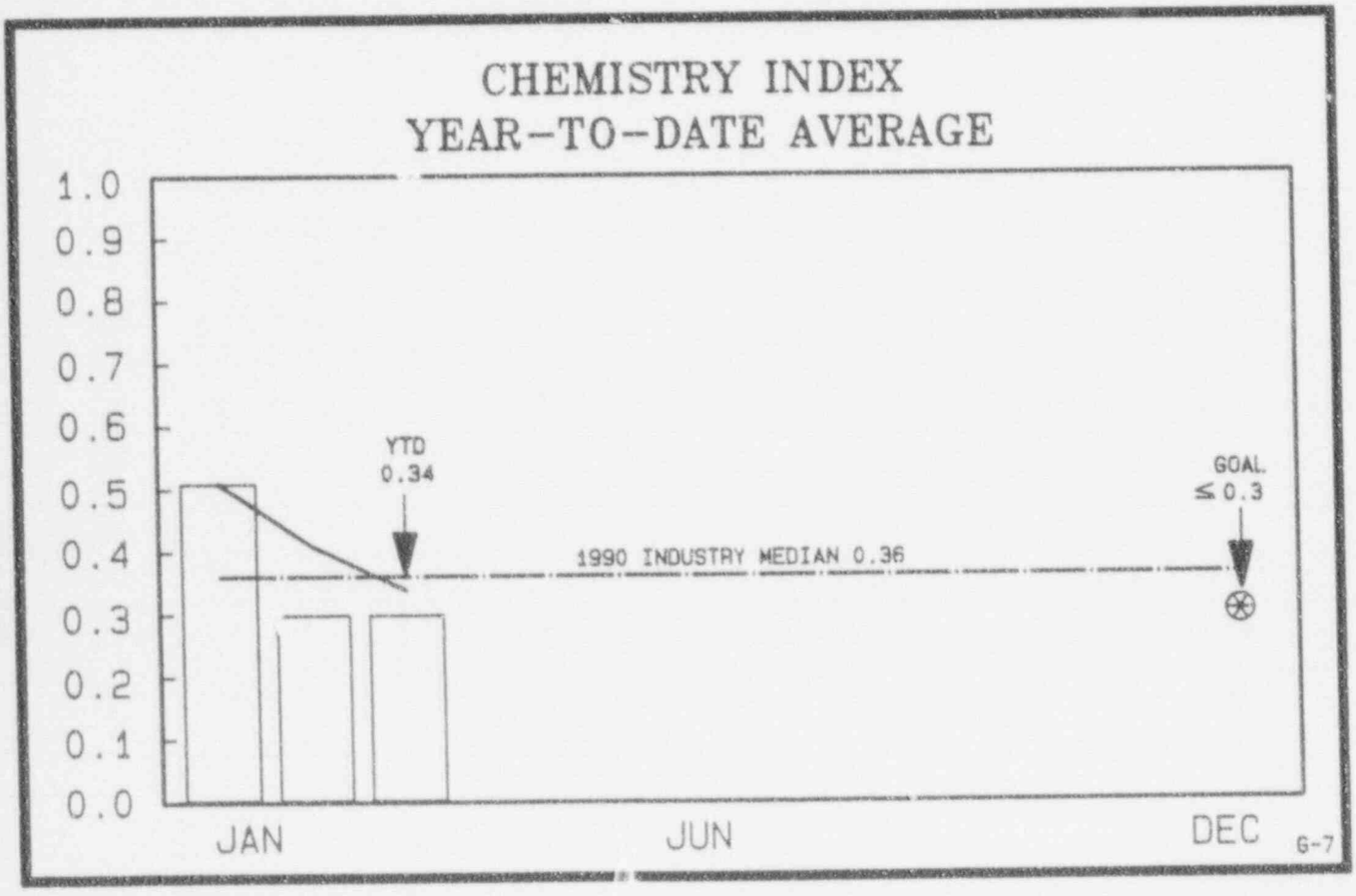
# GGNS
# VOLUME OF LOW LEVEL WASTE
# YEAR-TO-DATE




## THREE YEAR MOVING AVERAGE

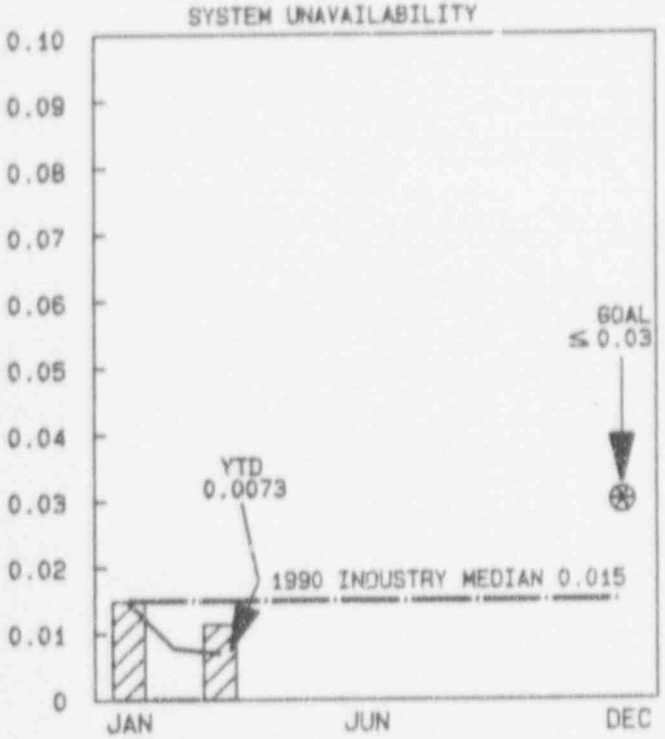



GGNS
FUEL RELIABILITY INDEX
YEAR-TO-DATE AVERAGE
UCI/SEC
500
400
300
200
100
0
GOAL
≤ 2700
YTD
9.1
1990 INDUSTRY MEDIAN 99
JAN
JUN
DEC

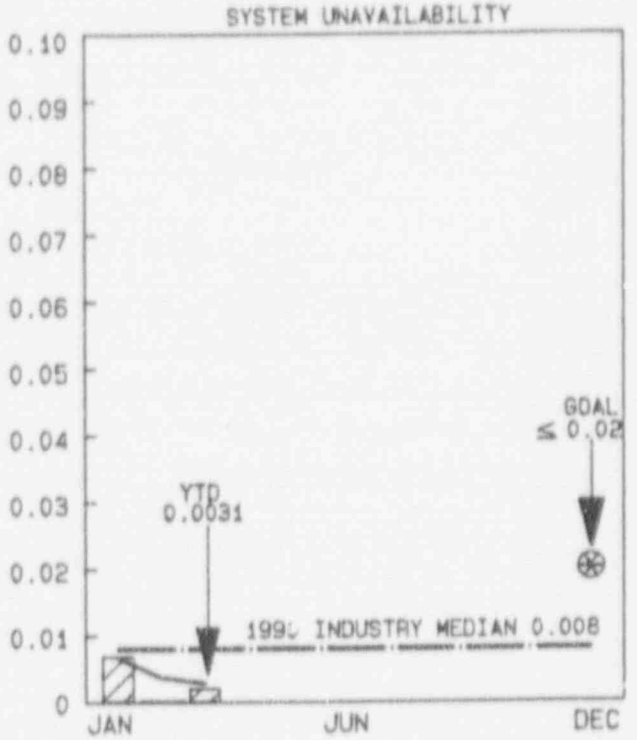

CHEMISTRY INDEX
YEAR-TO-DATE AVERAGE
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
YTD
0.34
GOAL
≤ 0.3
1990 INDUSTRY MEDIAN 0.36
JAN
JUN
DEC
G-7

# CGNS
# SAFETY SYSTEM PERFORMANCE
## Y-T-D

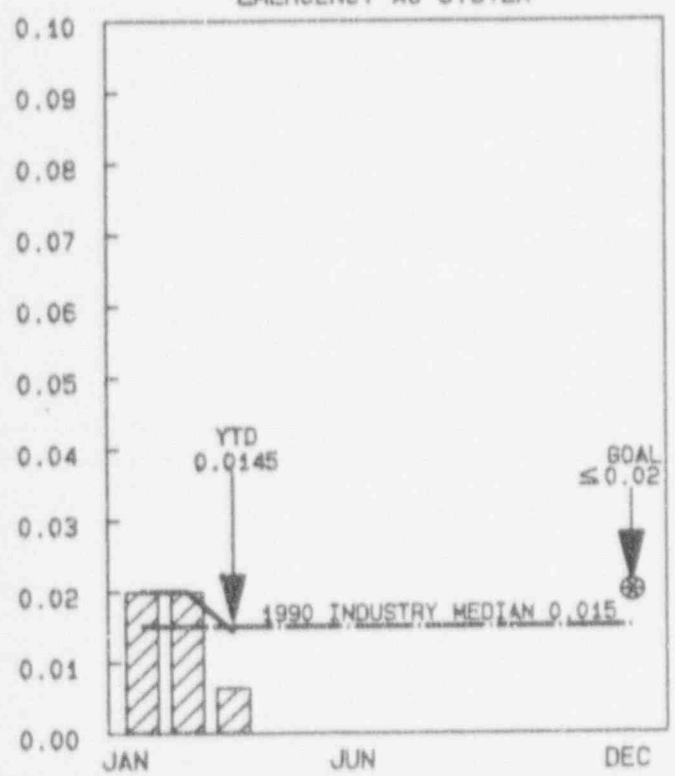
HIGH PRESSURE INJECTION
SYSTEM UNAVAILABILITY




RESIDUAL HEAT REMOVAL
SYSTEM UNAVAILABILITY




EMERGENCY AC SYSTEM



WATERFORD 3 STEAM ELECTRIC STATION
STATUS REPORT

A. PLANT STATUS (AS OF 3/31/91)

The plant is currently in mode 6 (refueling).

| | 3/31/91 YTD |
|---|---|
| o Equivalent Availability (%) | 81.8 |
| o Unit Capability Factor (%) | 82.2 |
| o Capacity Factor (%) | 80.6 |
| o Gross Heat Rate (BTUs/KWH) | 10332 |
| o Consecutive Days On-Line - 0 (Operated 151 consecutive days prior to shutdown for refueling) | |

B. OPERATING SUMMARY SINCE LAST REPORT (AS OF 3/30/91)

- The plant entered Mode 6 (Refueling) on March 29, 1991 at 0847 and the reactor head was removed and secured the following Tuesday, 72 hours behind schedule. The lost time was due to difficulty in removing Reactor Coolant Pump 2A motor mount for gasket changeout.

- Currently, the reactor core off-load is 50% complete with 32 hours recovered on the original outage length. The time recovery can be attributed to cooperation exhibited in the commencement of the system outage work and the successful implementation of the modification to install the Refueling Machine Console. The new console has enhanced outage personnel's ability to move fuel in a faster and safer manner.

- During the second and third weeks of the outage, containment activities included: the removal of the RCP 2A motor mount, removing and cutting in-core instrumentation, eddy current testing of both steam generator's primary tubes and the sludge lance cleaning of Steam Generator #2's secondary.

B. OPERATING SUMMARY SINCE LAST REPORT (Cont'd.)

- Currently the balance of plant work consists of maintenance and repairs on EDG "A", inspection and repairs of the extraction steam bellows and piping, dredging the intake structure, and maintenance on the circulating water pumps.

- Main turbine work is proceeding slightly behind schedule and is currently vying with refueling activity for critical path due to additional work scope. In particular, the high pressure turbine seal work has added approximately 8 days to the original outage scope, bringing it within 10 hours of refueling activities.

- Plans are in place to recover additional time on the outage schedule. Completion on the original schedule still appears feasible.

# LICENSEE EVENT REPORT

Waterford-3 Nuclear Station

Waterford-3 SES
Licensee Event Reports for the Quarter Ending March 31, 1991
Unit 3

**Title:** Chlorine Gas Release From Local Chemical Plant

**LER No:** 90-020-00

**Date Submitted:** January 28, 1991

On December 27, 1990, an alert was declared at 2116 hours at Waterford Steam Electric Station Unit 3, due to a release of chlorine gas from a nearby chemical plant. Waterford 3 was notified of a site area emergency at the chemical plant and entered procedures for Toxic Chemical Release (Operating Procedure 901-047) and Toxic Chemical Emergency (Emergency Plan 004-010). At 2144, the event was terminated. This event is being reported as an item of potential industry interest although this event did not present a hazard to plant equipment or the health and safety of the general public.

**Title:** Both Trains of Control Room Air Conditioning Inoperable Due to Breach in the Control Room Envelope

**LER No:** 90-019-01

**Date Submitted:** February 28, 1991

On December 12, 1990, with Waterford Steam Electric Station Unit 3 at 100% power, Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.3 was entered when both trains of the Control Room Heating Ventilation and Air Conditioning (HVAC) System were declared inoperable due to a breach in the Control Room envelope. The breach in the Control Room envelope existed since December 5, 1990, when a Control Room penetration fire seal was removed in accordance with an approved Design Change. The root cause of this event is lack of sufficient documentation and details of the Control Room Envelope boundary seals.

On December 14, 1990, the plant was operated in a TS LCO 3.0.3 condition for approximately one and one half hours while investigating a problem with the Control Room HVAC System. The root cause of this event is the combination of two failures. The Control Room recirculation damper failed in the intermediate position and the Control Room envelope had excessive leakage. As a result, the operability requirements of the Control Room HVAC System could not be met.

LER 90-019-01 (Cont'd)

Operation in a TS LCO 3.0.3 condition is reportable as operation prohibited by plant TS. Calculations have shown that during each of these events the habitability of the Control Room would have been preserved during a high radiation or toxic chemical scenario; therefore, this event did not threaten the health and safety of the general public or plant personnel.

**Title:** Failure to Place Plant Protection Channel in Bypass during Surveillance Testing

**LER No:** 91-001-00

**Date Submitted:** March 4, 1991

On February 3, 1991, during the performance of Operating Procedure OP-903-107, "Plant Protection System Channel B Functional Test," Steam Generator 2 low pressure setpoint was determined to be out of tolerance. Contrary to Technical Specification 3.3.1 Action Statement 2, channel B was not placed in the bypassed or tripped condition within 1 hour. This event is reportable as a condition prohibited by Technical Specifications.

This event was caused by an inappropriate action resulting from a failure to recognize that the channel was required to be placed in bypass during the surveillance test. The inoperable channel bistables were placed in bypass when the error was realized. Since the other three Steam Generator 2 low pressure channels were within specification, this event did not result in an increased risk to the health and safety of the public or plant personnel.

**Title:** Inadequate Design of Air Accumulators Due to Incomplete Review of Post-TMI Action Plan

**LER No:** 89-007-01

**Date Submitted:** March 8, 1991

At 1100 hours on March 31, 1989, Waterford Steam Electric Unit 3 was Operating at 100% power when the issue of reportability was raised on

LER 89-007-01 (Cont'd)

the sizing of the Instrument Air (IA) accumulators which supply the Safety Injection (SI) Recirculation Sump Outlet Isolation Valves, SI-602A&B. Design requirements did not consider certain accident scenarios, with a postulated loss of IA where operation of the valves may be required. Manual operation of the valves was not considered an adequate backup due to potential radiation levels at the valve location. Therefore, the plant was operated in an unanalyzed condition since initial startup.

On February 6, 1991, a review of Surveillance Procedures revealed that the plant was operated with a nitrogen accumulator IV leakage rate of 57.6 psi/hr vice the 55 psi/hr required by Design Basis Documentation (DBD). Accumulator IV supplies nitrogen to operate the SI pump suction valves to the refueling water storage pool on loss of IA. This condition existed from November 23, 1990 through February 7, 1991.

The root cause of this event was an inadequate review of design requirements implemented as part of the post-TMI action plan. Phase one of DC 3195 has been implemented to provide a nitrogen source of gas to provide remote operation of SI-602A&B and a review of the DBD is being conducted.

**Title:** Inadvertent Control Room Outside Air Isolation Due to Equipment Malfunction

**LER No:** 91-002-00

**Date Submitted:** March 25, 1991

On February 21, 1991, Waterford Steam Electric Station Unit 3 experienced an unplanned actuation of the Engineered Safety Feature (ESF) portion of the Control Room ventilation system. The actuation was initiated by a high alarm from one of the four normal Control Room Outside Air Intake (CROAI) radiation monitors, resulting in a Control Room isolation. Due to the ongoing performance of Operating Procedure 903-051, Control Room Emergency Filtration Unit Operability Check, a Control Room Emergency Filtration Unit was already in service. All other CROAI radiation monitors were indicating normal radiation levels and air samples taken in the area of the alarming radiation monitor showed no detectable activity. This event is reportable as an unplanned ESF actuation.

LER 91-002-00 (Cont'd)

The root cause of this actuation was equipment malfunction of the CROAI radiation monitor caused by a perforation in the beta radiation window light shield. The beta radiation window light shield was replaced and the CROAI was returned to service. A unit availability investigation will be conducted to evaluate beta radiation light shield failures. In this event, the Control Room Emergency Filtration System functioned as designed and there was no actual release of radioactive material; therefore, this event did not result in an increased risk to the health and safety of the public or plant personnel.

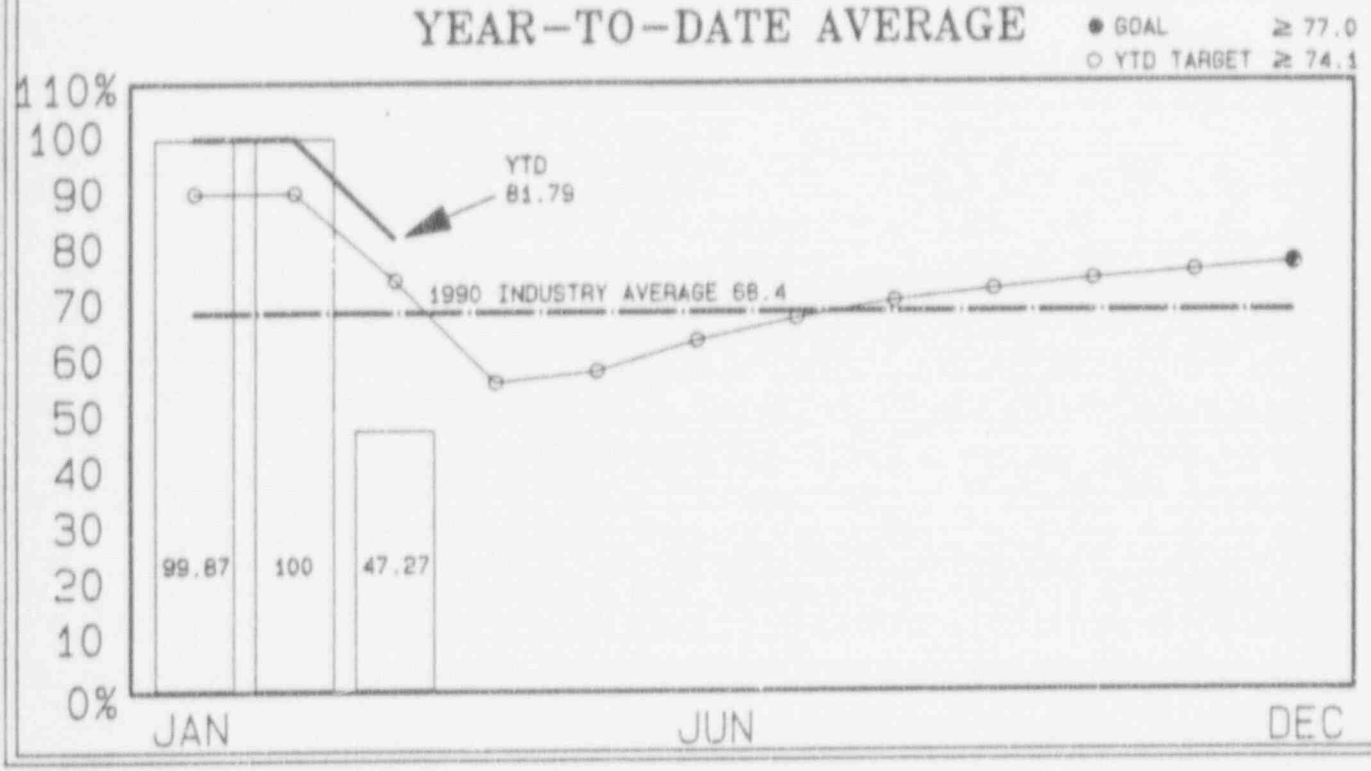
## LEGEND

Graphs with a double line border denote indicators which are within the established goal.

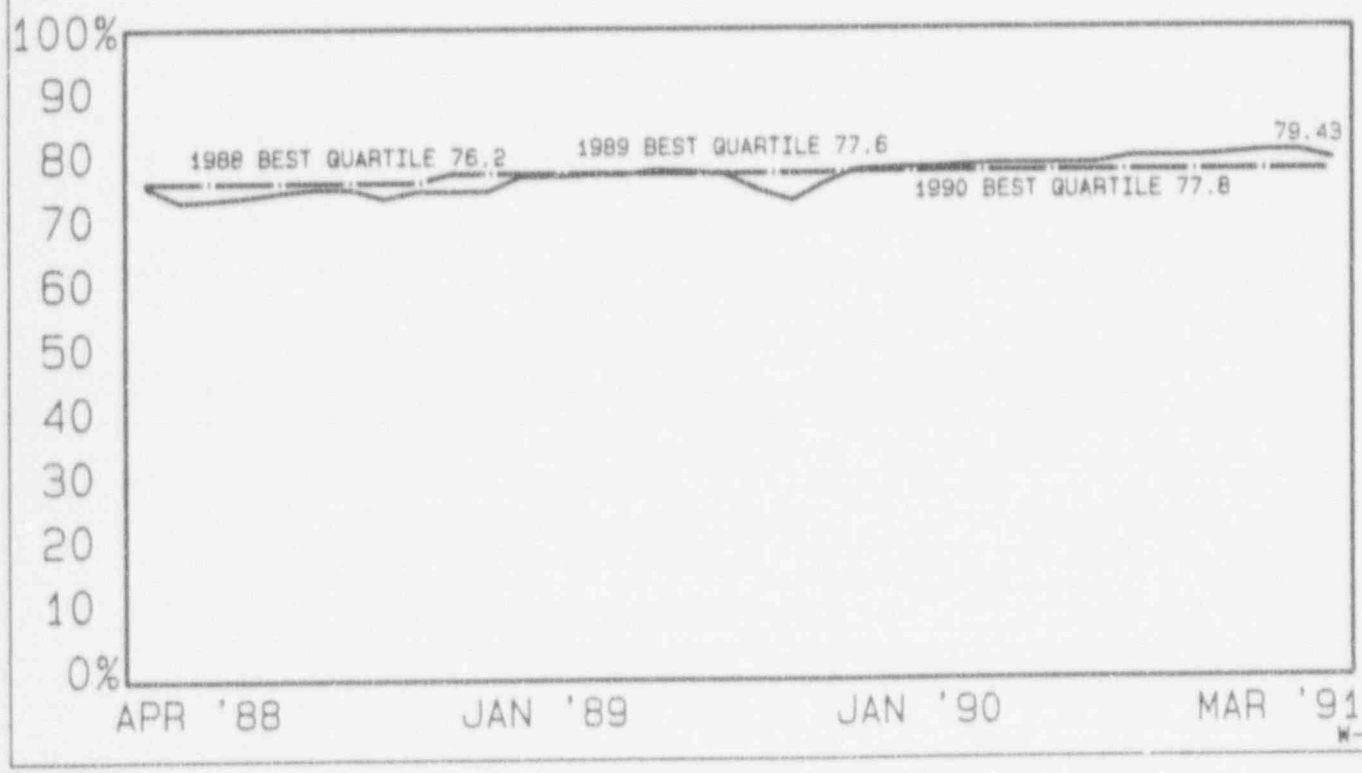
Graphs with a thick, single line border denote indicators which are outside the established goal.

Graphs with a thin, single line border denote indicators for which no goal has been established or for which a goal is not applicable.

# WSES-3
## UNIT EQUIVALENT AVAILABILITY
## YEAR-TO-DATE AVERAGE

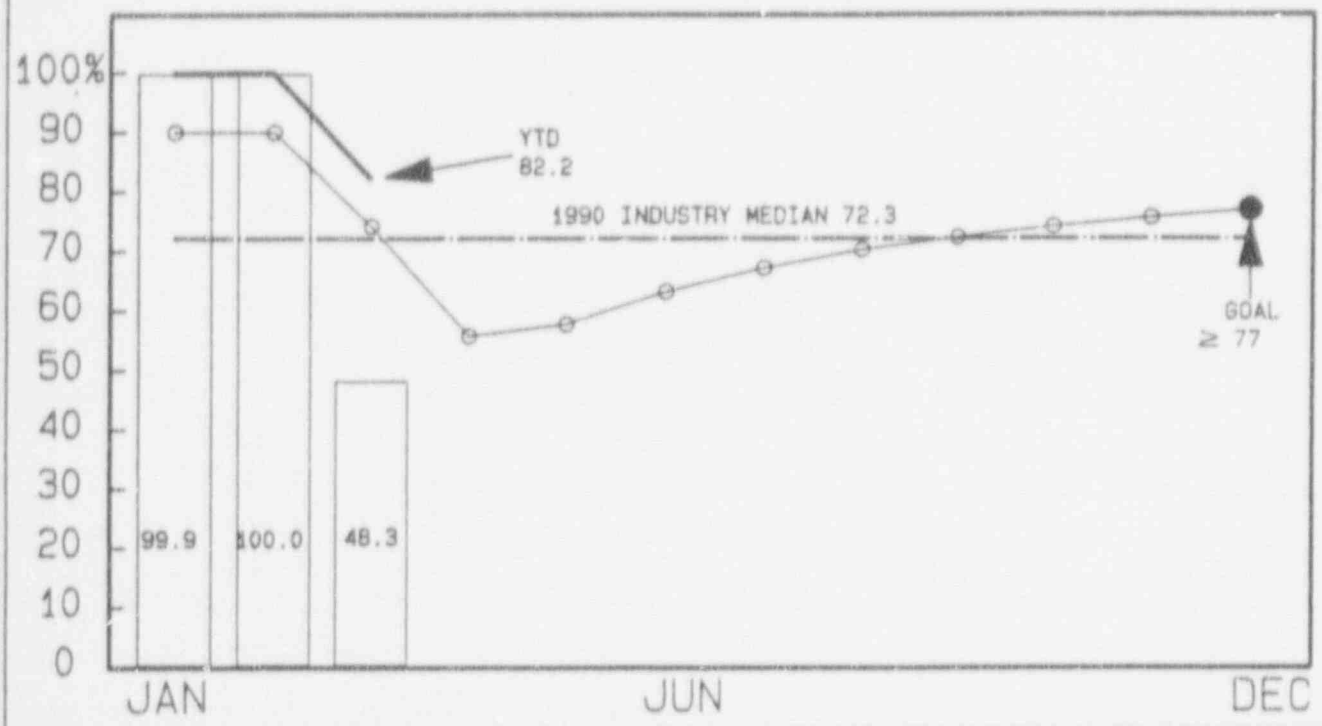



## THREE YEAR MOVING AVERAGE

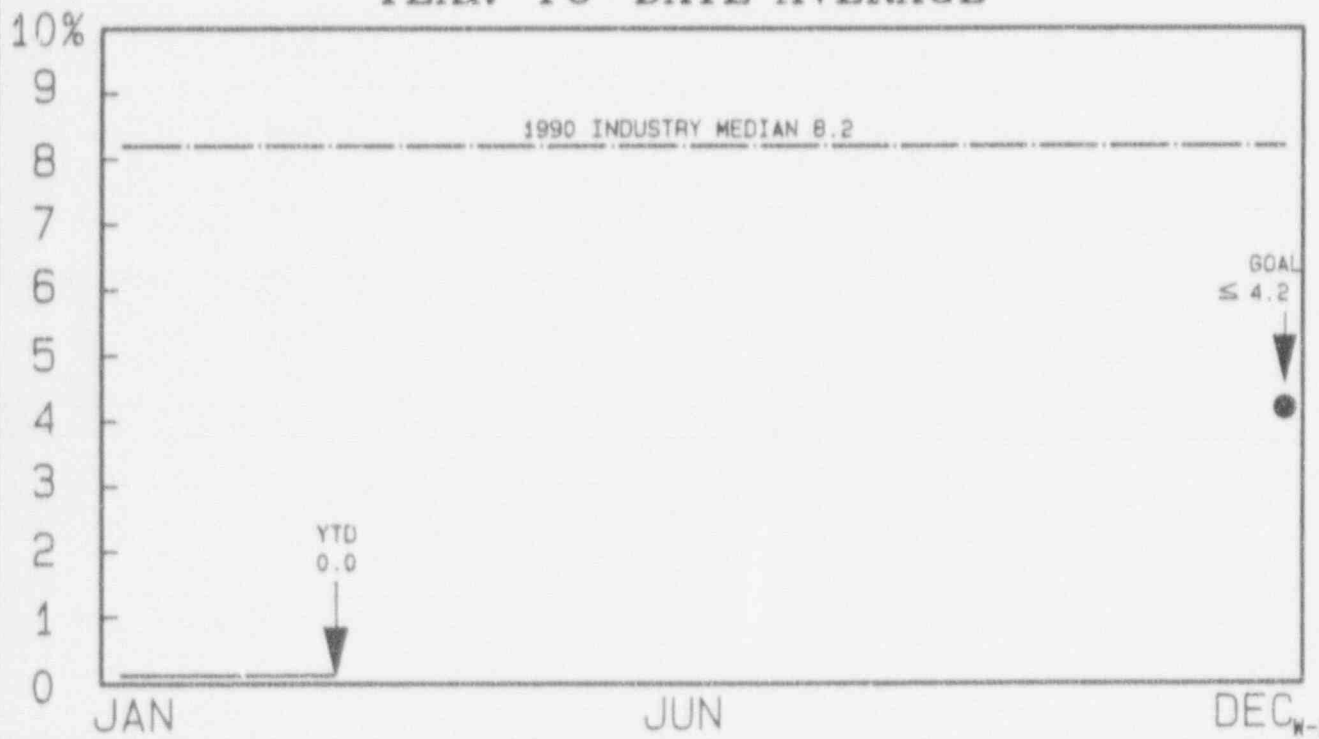



W-1

# WSES-3
## UNIT CAPABILITY FACTOR
## YEAR-TO-DATE AVERAGE




## UNPLANNED CAPABILITY LOSS FACTOR
## YEAR-TO-DATE AVERAGE

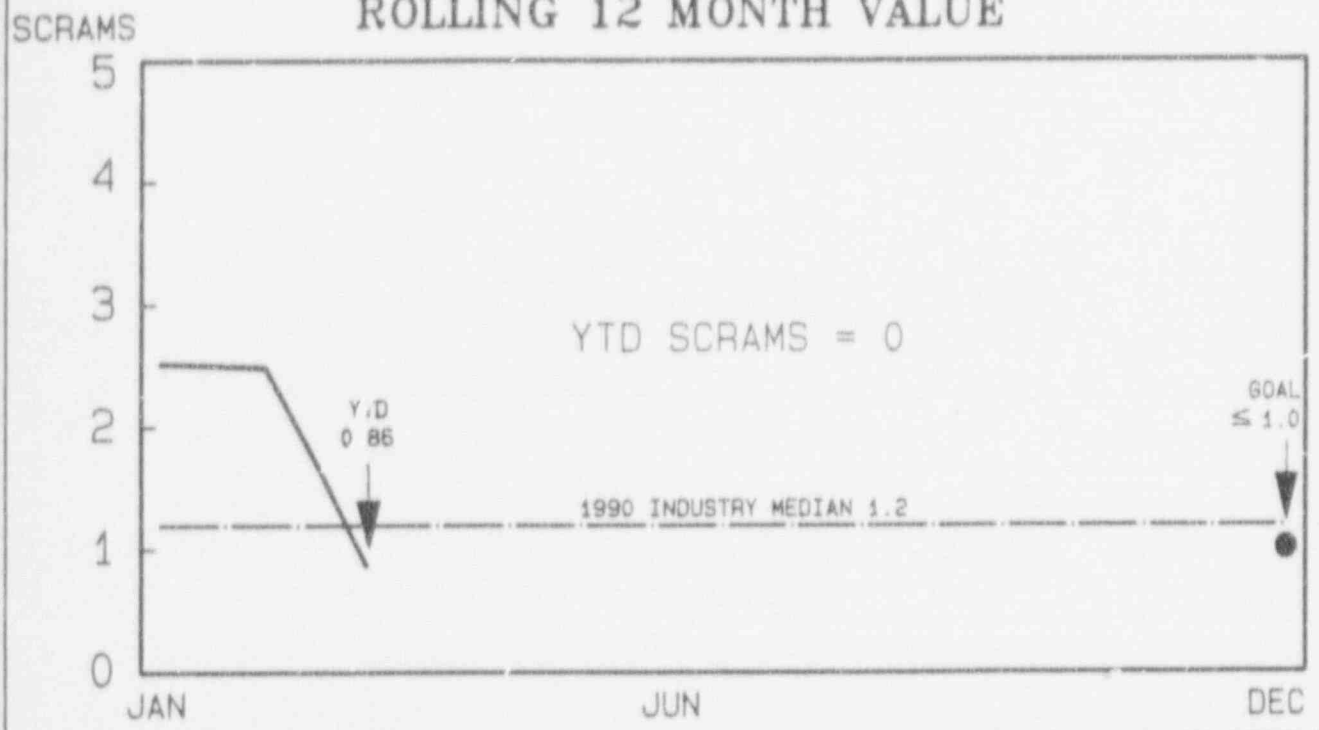



W-2

# WSES-3

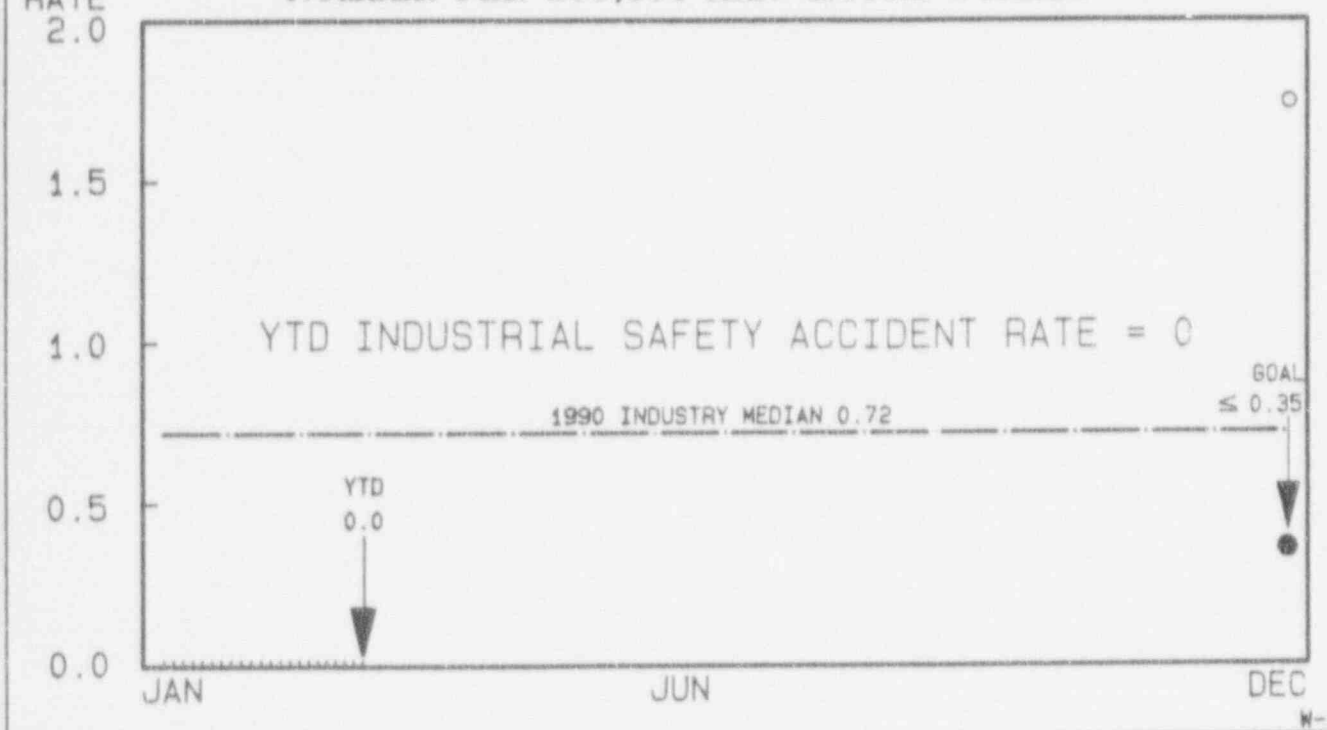
## SCRAMS PER 7000 CRITICAL HOURS

## ROLLING 12 MONTH VALUE



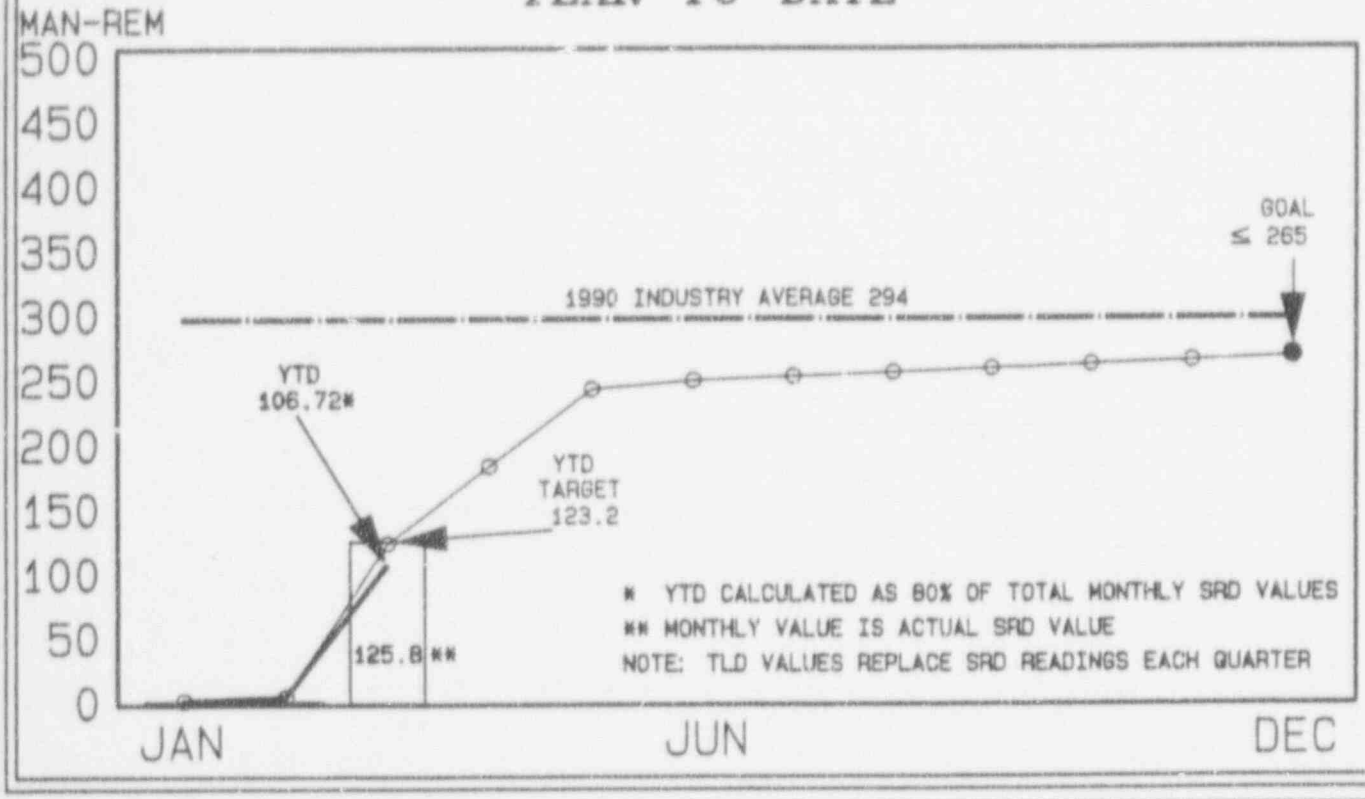

## Y-T-D INDUSTRIAL SAFETY ACCIDENT RATE

### NUMBER PER 200,000 MAN-HOURS WORKED

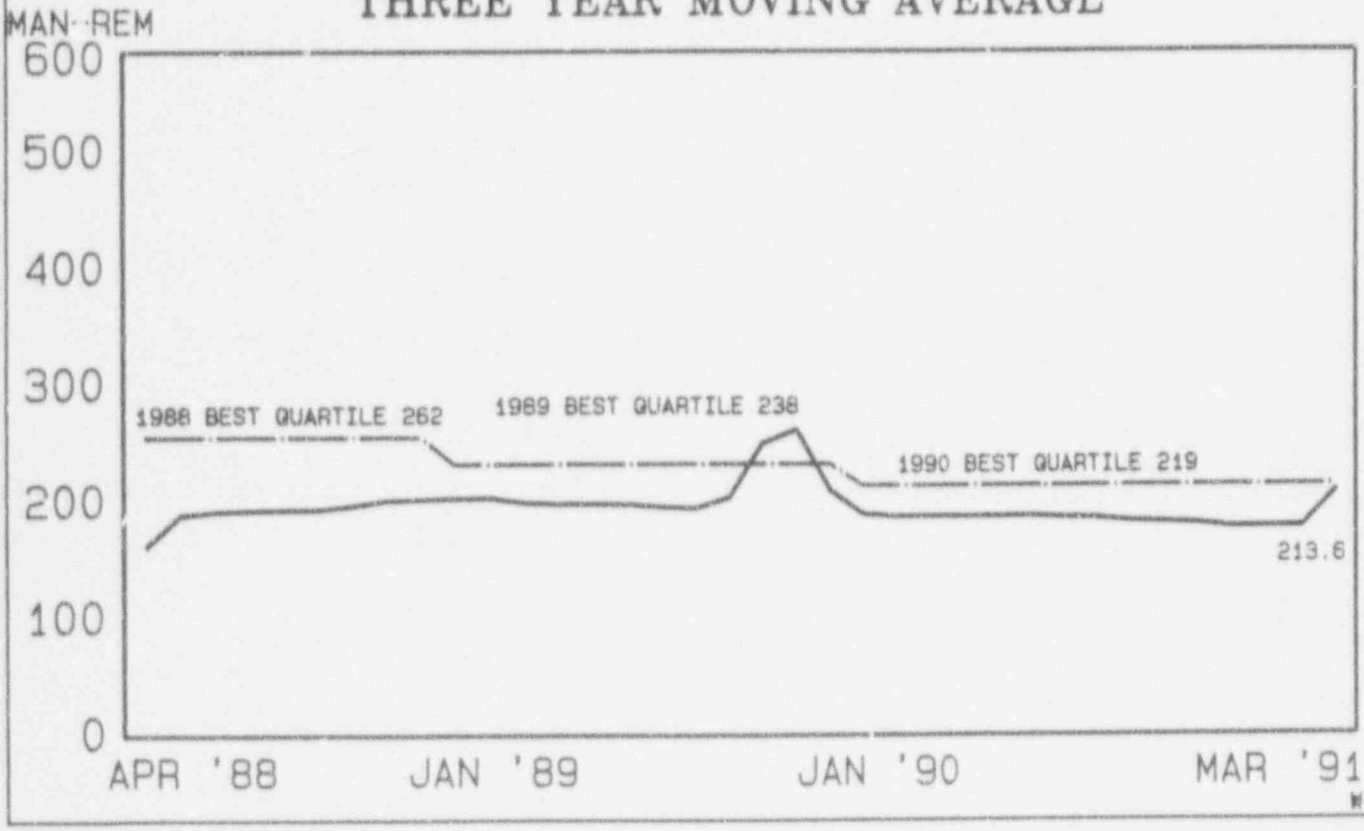



W-3

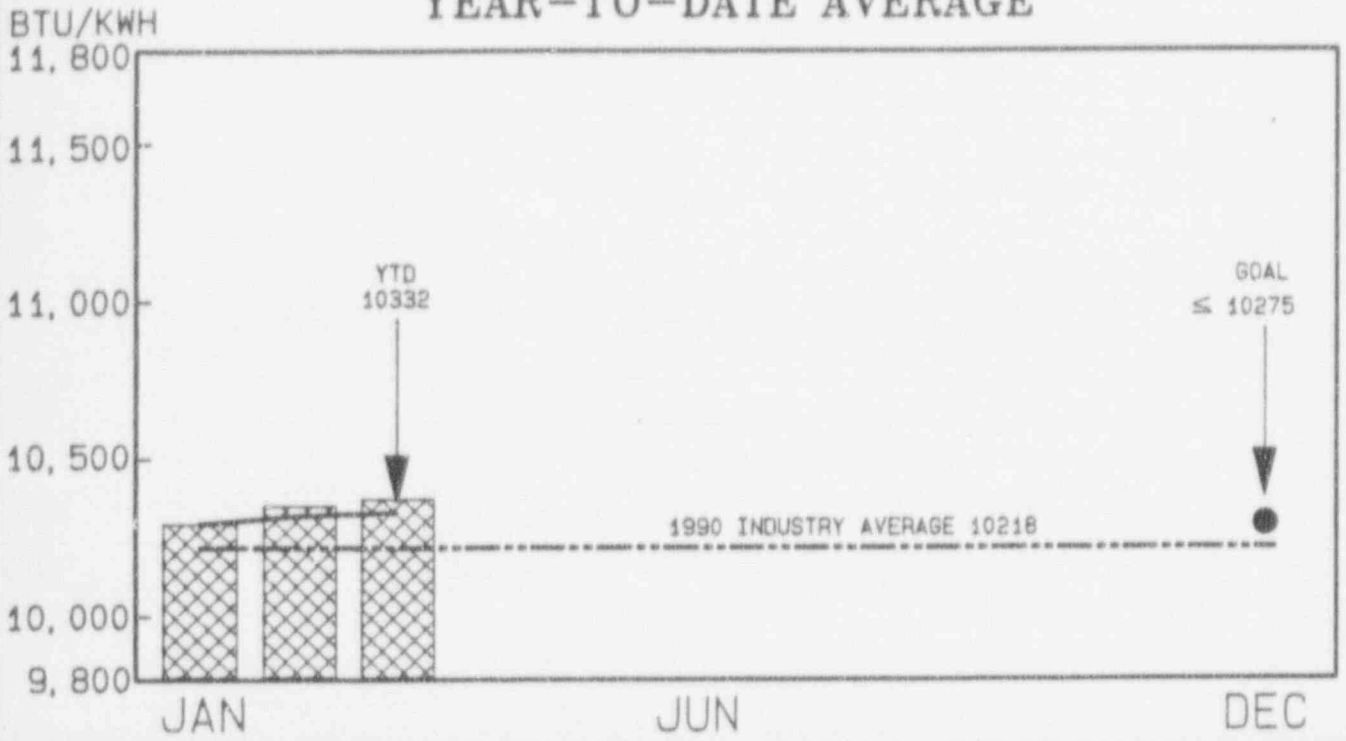
# WSES-3
# RADIATION DOSE
# YEAR-TO-DATE



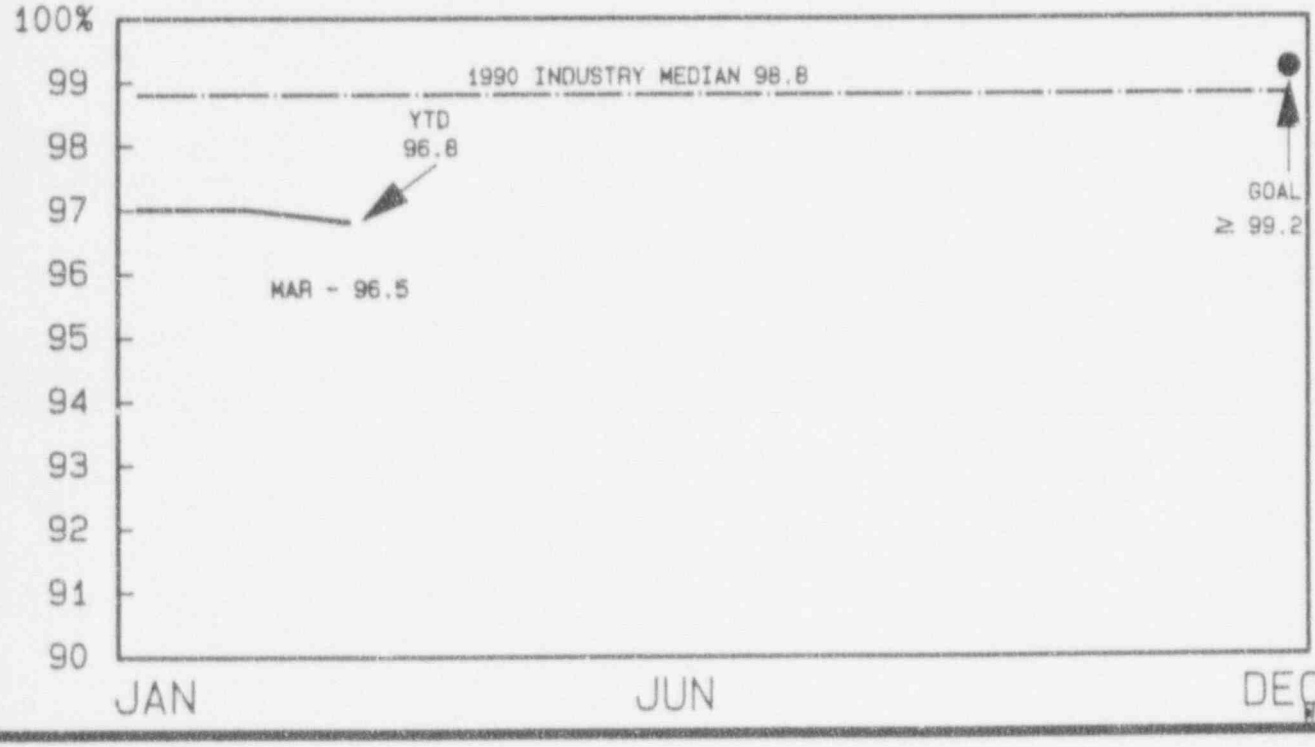

## THREE YEAR MOVING AVERAGE



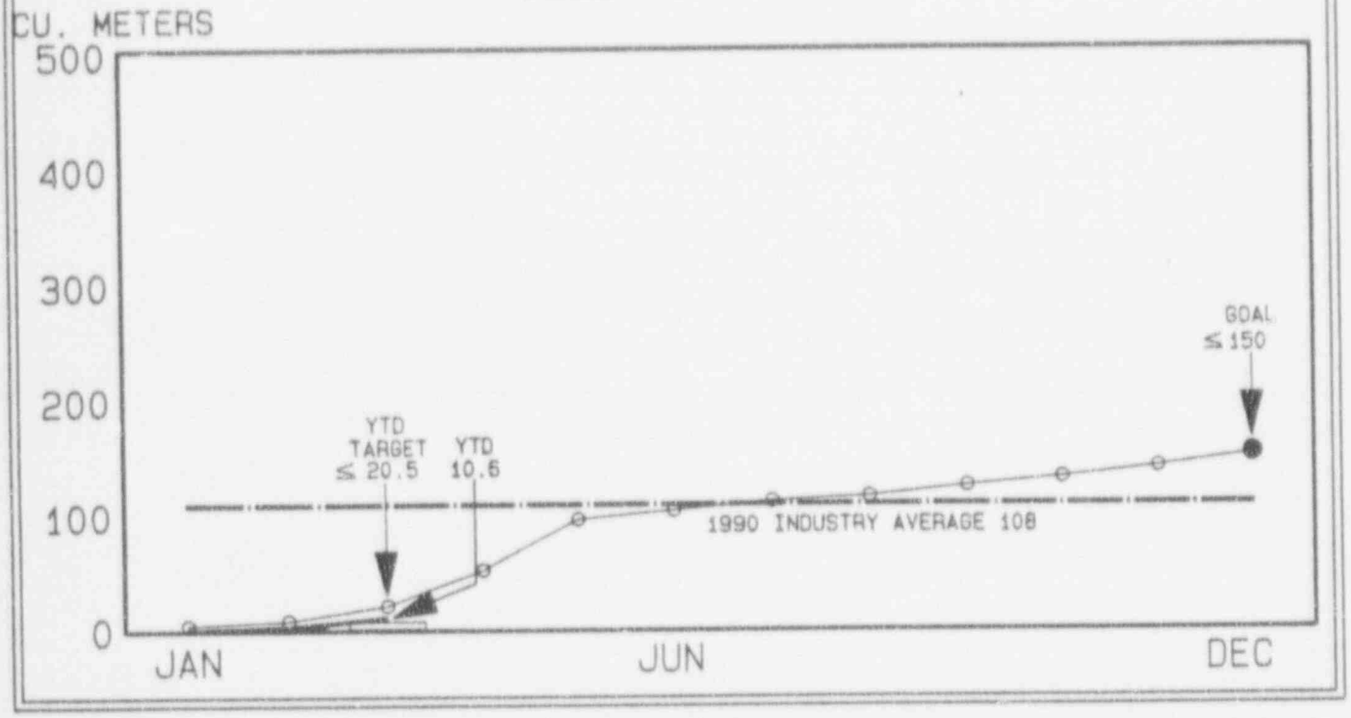
# WSES-3
## UNIT GROSS HEAT RATE
## YEAR-TO-DATE AVERAGE



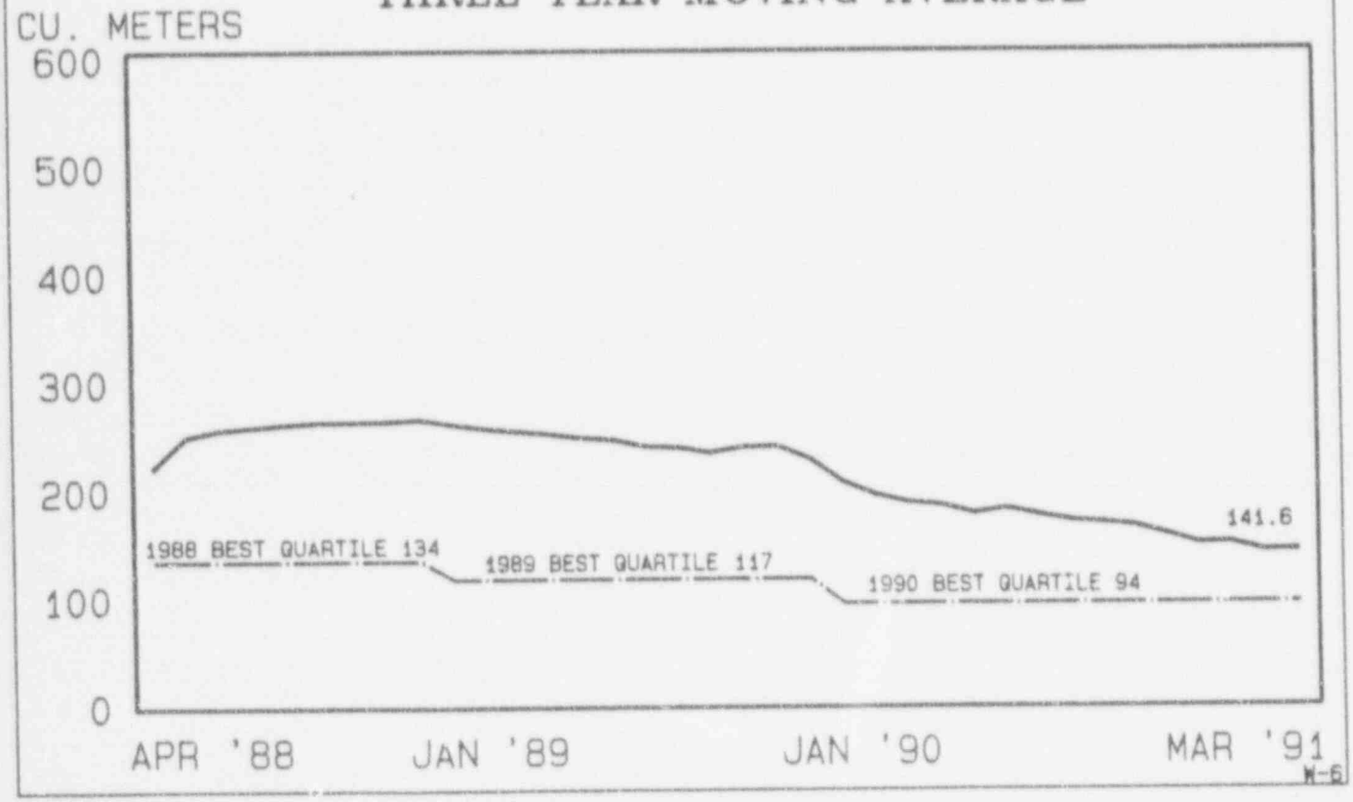

## THERMAL PERFORMANCE INDEX
## YEAR-TO-DATE AVERAGE



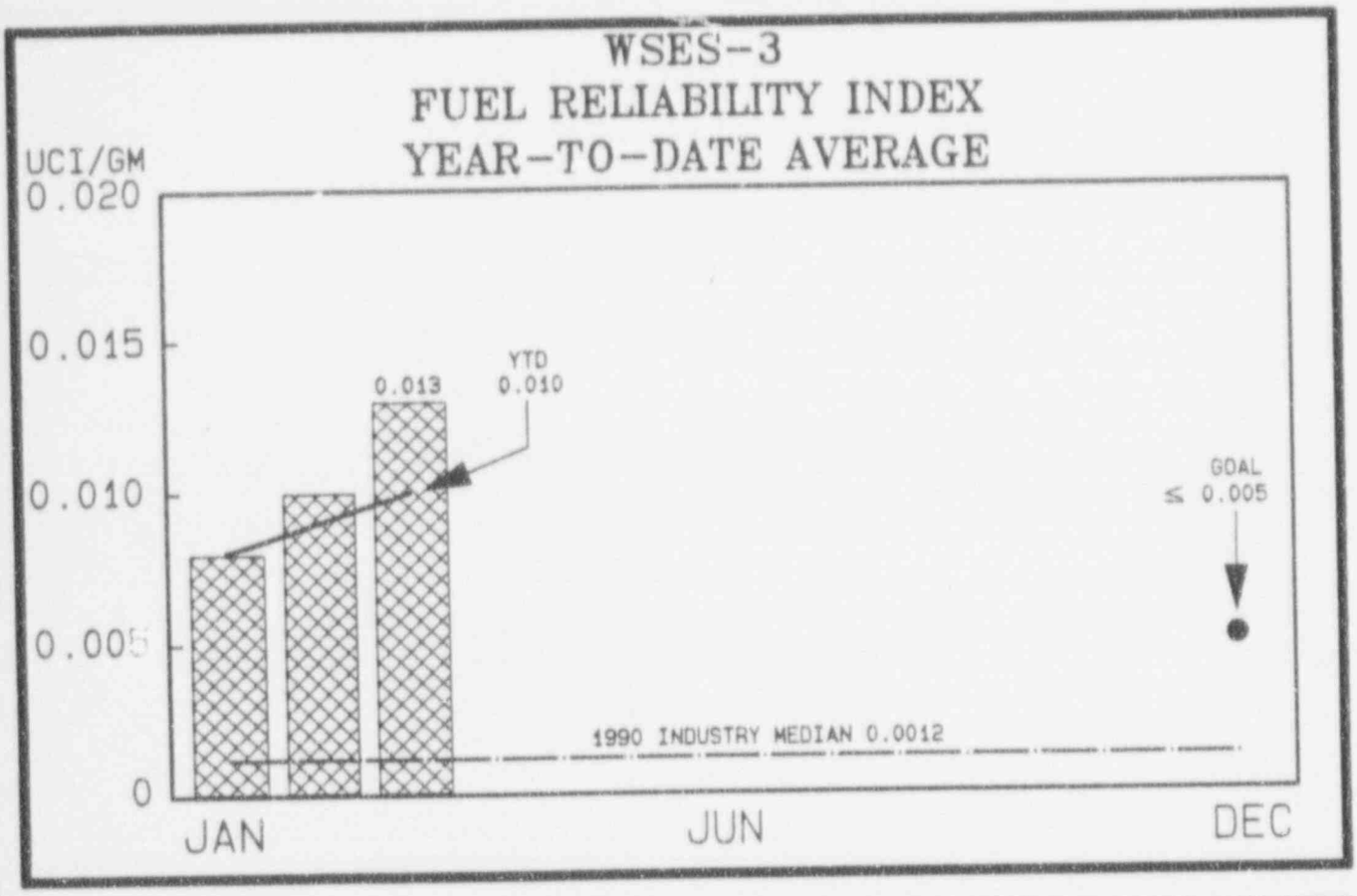
# WSES-3
## VOLUME OF LOW LEVEL WASTE
## YEAR-TO-DATE

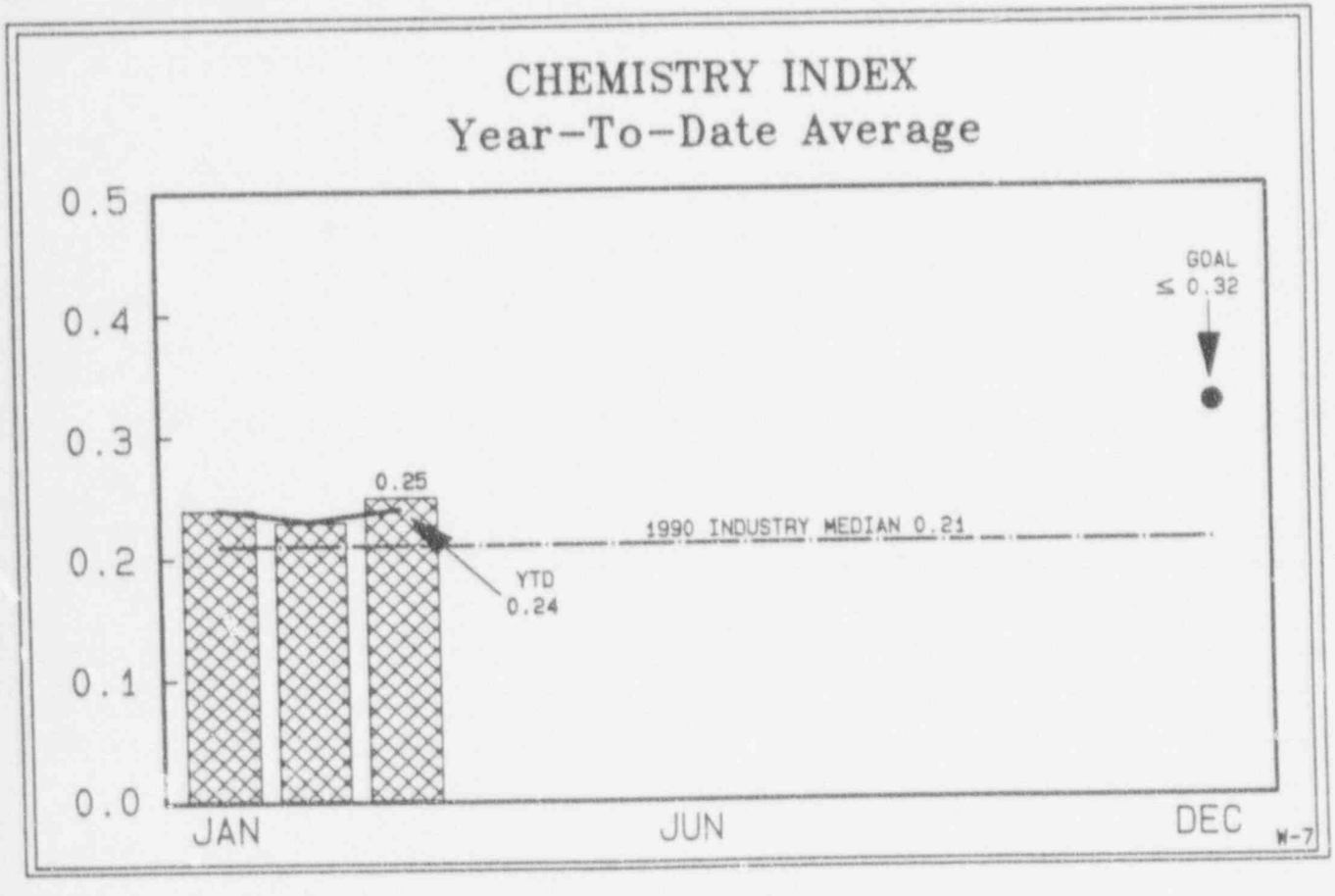



## THREE YEAR MOVING AVERAGE




W-6


WSES-3
FUEL RELIABILITY INDEX
YEAR-TO-DATE AVERAGE
UCI/GM
0.020
0.015
0.010
0.005
0
0.013
YTD
0.010
GOAL
≤ 0.005
1990 INDUSTRY MEDIAN 0.0012
JAN
JUN
DEC


CHEMISTRY INDEX
Year-To-Date Average
0.5
0.4
0.3
0.2
0.1
0.0
GOAL
≤ 0.32
0.25
1990 INDUSTRY MEDIAN 0.21
YTD
0.24
JAN
JUN
DEC
W-7

# WSES-3
# SAFETY SYSTEM PERFORMANCE
# Y-T-D