UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 16, 1989

Note to: Distribution

Through: S. M. Stern

From: W. F. Burton

Subject: MEETING WITH INPO ON PERFORMANCE INDICATORS

Place: Bethesda Maryland; Maryland Nat. Bank Building; Third Floor.

Time: 1:00 PM; May 17, 1989

Topics: - Differences and Similarities between INPO and NRC Performance Indicators.

- INPO Safety System Performance Indicator.

Background on Differences between INPO and NRC Indicators.

INPO and NRC both track four indicators in common: Automatic Scrams while Critical (Scrams), Safety System Actuations (SSA), Forced Outage Rates (FOR) and Collective Radiation Exposure (CRE). NRC will use INPO's CRE data beginning with the Performance Indicator (PI) Report for the First Quarter of 1989. Part of the differences (amount unknown) between the industry averages developed by INPO and NRC for Scrams can be explained by the fact that INPO does not count plants that have less than 25% capacity factor during a year and those events for new plants and INPO does not consider new plants until Jan 1 of the second full year following full power licensing. Similarly, INPO does not reflect new plant SSAs, FORs and CREs in their industry averages. NRC excludes plants in long term shut down from its industry averages. Attachment 1 is a discussion of the differences between the INPO and NRC PIs.

A major difference between the INPO and NRC indicators is in the treatment of SSAs. While both INPO and NRC base SSAs on similar systems, INPO has a higher threshold than NRC for classifying events as SSAs.

INEL has compared the Safety System Actuation counts for the Third and Fourth Quarters of 1988 as delineated by INPO and NRC. Attachment 2 is a summary comparison of the those plants where differences between the SSA counts were identified. Attachment 3 compliments Attachment 2 and details those SSAs identified by NRC but not INPO and a brief description of the reason for our classifying that event as an SSA.

Attachment 4 is a copy of the 1988 INPO PIs published in March 1989. Attachment 5 are the detailed definitions of the INPO indicators, as issued by INPO in February 1988.

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ATTACHMENT 1

DIFFERENCES BETWEEN THE INPO AND NRC INDICATORS

The NRC is attempting to better understand the performance indicators used by INPO and how they may differ from NRC PIs. The INPO and NRC PIs that are common are Unplanned Automatic Scrams While Critical, Safety System Actuations, Forced Outage Rate, and Collective Radiation Exposure. The items below are highlights of the differences, the actual definitions of the INPO indicators, which contain many clarifying notes, are in Attachment 5.

UNPLANNED AUTOMATIC SCRAMS WHILE CRITICAL:

The INPO indicator is defined as the number of unplanned automatic scrams (RPS logic actuations) that occur while the reactor is critical. This is similar to the NRC PI definition for Automatic Scrams While Critical, with the following differences:

- INPO collects scram data on all plants beginning with January 1 of the first calendar year following full power licensing. The NRC scram counts for the PI report includes all critical scrams.

- INPO does not count manual turbine trips which lead to reactor scrams which were effected to protect important equipment or to minimize the effects of transients . The NRC PIs do count such events.

- INPO considers short-term transient conditions in its determination of whether a unit was critical or not. The NRC determines the actual plant condition at the time of the event.

- INPO industry averages exclude data prior to January 1 of the second full year following commercial operation, and those years where the capacity factor is less than 25 percent, or data element were not provided for the full period. NRC industry averages exclude plants in long term shut down.

UNPLANNED SAFETY SYSTEM ACTUATIONS

This INPO indicator is defined as the sum of a.) the number of unplanned ECCS actuations that result from reaching an ECCS actuation setpoint or from a spurious/inadvertent ECCS signal and b.) the number of unplanned emergency AC power system actuations that result from a loss of power to a safeguards bus. This is the same as the NRC PI definition of Safety System Actuations.

Although the INPO and NRC definitions essentially are the same, the body of data to which the definition is applied is different:

- Although it appears that the same ECCS systems are included in the definitions, what is considered a valid actuation seems to be different. INPO SSA definition requires the actuation of a "major" system, whereas the NRC interprets this actuation as either a valid or spurious signal (whether the equipment starts or not). For both INPO and NRC, an undervoltage signal on a safeguards bus is counted as a diesel start.

- INPO industry averages exclude plant data prior to January 1 of the second full year following commercial operation. NRC industry averages exclude plants in long term shut down.

- Since INPO has a higher threshold than NRC for classifying SSAs, the NRC indicator would have a higher value than the INPO indicator.

FORCED OUTAGE RATE:

The formula for computing the FOR used by INPO and NRC are the same.

In computing industry averages, INPO uses data for units beginning January 1 of the second full calendar year following full power licensing, and has a requirement that data elements be provided for at least 50% of the time period to be included in the industry average. NRC excludes plants in long term shut down from its industry averages.

COLLECTIVE RADIATION EXPOSURE:

NRC uses the INPO supplied data.

ATTACHMENT 2

PLANTS FOR WHICH INPO AND NRC HAD DIFFERENT COUNTS FOR SSAs FOR THE THIRD AND FOURTH QUARTERS OF 1988

PLANT	YR-QTR	INPO	NRC
ARKANSAS 1	88-4	1	0
ARKANSAS 2	88-3	0	1
BROWNS FERRY 1	88-4	1	3
BROWNS FERRY 2	88-4	0	2
COOPER	88-3	1	2
CRYSTAL RIVER 3	88-4	0	2
DAVIS BESSE	88-3	1	0
DAVIS BESSE	88-4	1	0
DIABLO CANYON 1	88-3	0	1
DIABLO CANYON 2	88-3	0	2
DUANE ARNOLD	88-4	1	2
FORT CALHOUN	88-4	0	1
GRAND GULF	88-4	0	1
HOPE CREEK	88-3	0	2
INDIAN POINT 3	88-4	0	1
MCGUIRE 1	88-4	0	1
NORTH ANNA 1	88-3	0	1
NORTH ANNA 2	88-3	0	1
OCONEE 3	88-3	1	0
OYSTER CREEK	88-4	0	1
PEACH BOTTOM 3	88-3	1	2
PERRY	88-4	0	1
QUAD CITIES 2	88-4	0	1
RANCHO SECO	88-4	0	1
SALEM 2	88-3	1	0
SEQUOYAH 1	88-3	2	0
SEQUOYAH 2	88-3	0	1

AUTO SCRAMS WHILE CRITICAL
AGREE ON ALL SCRAMS

Safety System Actuations (ESFs)

Diesel Start Signal	Any valid low volt signal on a safety bus which should cause the diesel generator to start.
ECCS	Any SI signal which causes the following system to actuate. HPCS, HPCI, LPCS, LPCI, LPSI, HPSI (not accumulator injection), no SI if no major equipment operated (i.e. valves did not move or pumps did not start).
Arkansas 1	88-4 Can find no event in the fourth quarter where an ESF occurred.
Arkansas 2 50-368	88-3, 368880011, 08/01/88 HPSI manually started to control PZR level after the scram.
Beaver Valley 1	88-4, RE-14095 Beaver Valley 1 cancelled the event. We now agree with INPO.

Browns Ferry 1 50-259	88-4, 259880044, 11/01/88, 1311 Operator did not hold switch long enough to ensure breaker shut causing a low volt signal on 4160 shutdown board 'B'.
INPO counted one event.	88-4, 259880045, 11/01/88, 1654 Breaker failed to close while shifting power supplies causing undervoltage for 30 seconds and a diesel start.
	88-4, 259880045, 11/01/89, 1700 Breaker failed to close while shifting power supplies causing undervoltage for 30 seconds and a diesel start.
Browns Ferry 2 50-260	88-4, 260880017, 12/18/88, 0435 '2C' core spray pump started. Discharge valve tagged out.
	88-4, 260880016, 12/09/88, 1344 '2D' RHR pump started in normal standby. LPCI lineup manual injection valve shut.
	88-4, 259880049, 12/17/88, 1721 RHR pump start not in LPCI mode as stated in LER.
Cooper 50-298	88-3, 298880021, 08/25/88, 0040 Auto start HPCI and RCIC (no PI) after scram.
	88-3, 298880026, 09/30/88, 1257 Undervoltage on emergency transformer for about 2 seconds caused diesel start.

Crystal River 3	88-4, 302880021, 10/14/88, 0949 ESFAS actuation injected 1000 gallons borated water by LPCI.
50-302	88-4, 3028800024, RE-13838, 10/28/88, 0327 SI initiated for a time to maintain pressurizer level in RE.
Davis Besse	88-3 Can find no LER for this time frame that was an ESF.
	88-4 Can find no LER for this time frame that was an ESF.
Diablo Canyon 1 50-275	88-3, 323880008, 07/17/88, 0746 Loss of startup power to Unit 1 and 2 caused Unit 1 diesel to start.
Diablo Canyon 2 (2 counts) 50-323	88-3, 323880008, 07/17/88, 0746 Loss of startup power to Unit 1 and 2 caused Unit 2 diesel to start and HPCI on high steamline differential pressure.
Duane Arnold	88-4, 331880016, 10/17/88, 2334 Low volts on bus 1A3. 'A' diesel tagged out.
50-331	88-4, 331880014, 10/26/88, 0353 Moisture in reactor level switches caused low level signal and HPCS start.
	88-4, RE-14295 Plant cancelled event.

Farley 1	88-4, 348880024 LER states only RHR pump started, no high head SI pump as stated in RE.
Fort Calhoun 50-285	88-4, 285880024, 10/03/88, 1243 Momentary low volts on 1A4, 4160V bus caused diesel shut.
	88-4, 285880038, 12/31/88, 2156 No SIAS components actuated.
Hope Creek 50-354	88-3, 354880019, 07/28/88, 1029 'C' core spray pump started due to a human factor testing design deficiency.
	88-3, 354880022, 08/26/88, 1825 HPCI and RCIC (no PI) started on low reactor level after scram.
Indian Point 3 50-286	88-4, 286880006, 10/09/88, 1852 Breaker 52/5A opened. Diesel started and loaded bus.
McGuire 1 50-369	88-4, 369880038, 11/29/88, 1050 Switch placed incorrectly caused loss of power to 1/2 of unit. Diesel started and loaded.
North Anna 1 50-338	88-3, 338880020, 08/06/88, 2257 90% undervoltage on bus caused diesel to start and load bus.
North Anna 2 50-339	88-3, 339880002, 07/26/88, 1130 High head SI pump started during testing.
Oconee 3	88-3 Can find no ESF events.

Oyster Creek	50-219	88-4, 219880022, 10/02/88, 1357 Fault on 'B' side electrical distribution. DG did not start due to a problem in cable to DG.
Palisades		88-4, RE-13956 Plant cancelled the event due to the event not being an ESF.
Peach Bottom 3	50-278	88-3, 277880020, 07/29/88, 1858 Lost off-site power due to a transformer shorting. Diesel started and loaded bus.
		88-3, 278880009, 08/31/88, 2145 Startup feeder breaker opened and other leg of off-site power unavailable. Diesel started and loaded bus.
Perry	50-440	88-4, 440880043, 10/30/88, 2159 HPCS start signal when during HPCS breaker maintenance HPCS room cooler started.
Quad Cities 2	50-265	88-4, 265880027, 11/14/88, 1650 Technician shorted leads causing HPCI start. Operator secured pump before injection could occur.
Rancho Seco	50-312	88-4, 312880018, 12/09/88, 1826 'B' HPI pump manually started to maintain RCS inventory.
Salem 2		88-3 No ESF's - two scrams, but no indication of any ESF starts.

Sequoyah 1 50-327	88-3, (2) 327880029, 08/04/88 and 08/05/88 Undervoltage signal in starting circuit caused load shedding and diesel starts. No actual low volts on bus.
Sequoyah 2 50-328	88-3, 328880034. 08/15/88, 1651 Lost '1A' start bus which powers '1BB' shutdown board. All four diesel started.
GRAND GULF 50-416	88-4, 41688019, 10/10/88 HPCS INJECTED DUE TO TWO-WAY RADIO NEAR LOW LEVEL INSTRUMENTS.

ATTACHMENT 4

1988 PERFORMANCE INDICATORS FOR THE U.S. NUCLEAR UTILITY INDUSTRY

INSTITUTE OF NUCLEAR POWER OPERATIONS

March 1989

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1988 PERFORMANCE INDICATORS FOR THE U.S. NUCLEAR UTILITY INDUSTRY

INSTITUTE OF NUCLEAR POWER OPERATIONS

March 1989

P E R F O R M A N C E   I N D I C A T O R S

Nuclear plants with few unplanned scrams, few significant events, low personnel radiation exposure and high equivalent availability are generally recognized as well-managed overall. Such plants are more reliable and can be expected to have higher margins of safety.

In recognition of this and in keeping with its goal to promote excellence and the highest margin of safety, the Institute collects industry data on key performance indicators and shares this data with its members and participants.

Overall plant performance graphs, such as those provided in this report, summarize the industry's performance through the end of the year. Improving trends are evident in all areas.

The performance indicator program, now six years old, was refined in 1985 when three special review groups joined the Institute in developing a set of overall performance indicators designed to promote long-term industry improvement. Senior nuclear utility managers, a senior nuclear executive from each of the U.S. nuclear steam system suppliers and a group of outside experts contributed to this effort.

The groups agreed on 10 overall indicators as an important management tool for goal setting and for monitoring plant performance.

By April 1986, each U.S. utility with an operating unit had set challenging short-term and long-term 1990 goals for most of the overall indicators.

Many of INPO's international participants use the performance indicators. Several have also established long-term performance goals.

This foldout section provides performance data from 1980 to 1988 in selected areas. Graphs are included for seven of the 10 overall performance indicators. Industrywide goals for 1990 are included for these seven indicators. The 1990 goals were determined by averaging the individual unit goals furnished to INPO by each utility.

Data collection for the remaining three indicators began in 1987 and 1988. Sufficient data is not yet available to show meaningful trends for these indicators. These indicators are safety system performance, thermal performance and fuel reliability.

INPO and the industry are continuing to review the performance indicator program. Three relatively new indicators—safety system performance, thermal performance and fuel reliability—are now being tracked each year.

Safety system performance

Safety system performance is defined separately for each of three boiling water reactor and pressurized water reactor safety systems. The indicator is based on the hours that components in the safety system are unavailable to perform their intended functions. A low value indicates a greater margin of safety in preventing reactor core damage, and suggests a reduced chance of extended plant shutdown due to safety system failure during an operational event. Data collection for this indicator began in 1988.

Thermal performance

Thermal performance is defined as the ratio of corrected design gross heat rate to the adjusted actual gross heat rate. The design gross heat rate is corrected to reflect plant modifications and operating deviations from the initial thermal design. The actual gross heat rate is adjusted for circulating water temperature and the effect of feedwater pump efficiency.

Thermal performance reflects emphasis on thermal efficiency and maintenance of balance-of-plant systems, as does gross heat rate. However, this indicator provides a more meaningful basis for unit-to-unit performance comparisons than gross heat rate. Data collection for the thermal performance indicator began in 1987.

Fuel reliability

Fuel reliability is measured by the amount of fission products released into reactor coolant. More reliable fuel releases fewer fission products. High fuel reliability reduces radiological impact on plant operations and maintenance activities.

Fuel reliability is measured differently for boiling water reactors and pressurized water reactors due to design differences. Data collection for the fuel reliability indicator began in 1987.

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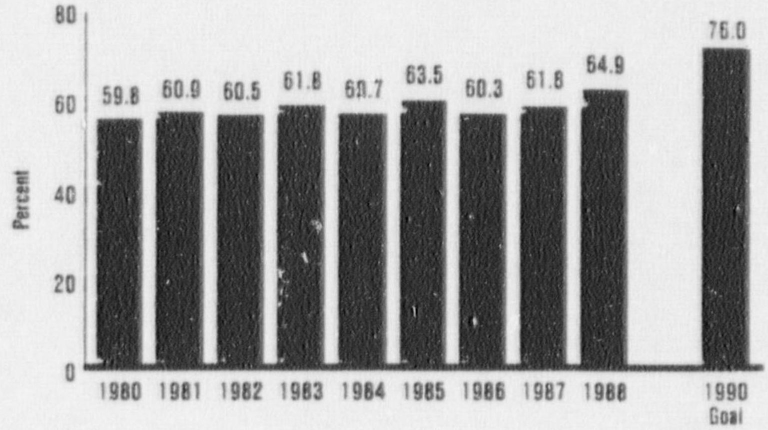
Equivalent availability factor

Equivalent availability factor is the ratio of the total power a unit could have produced, considering equipment and regulatory limits, to its rated capacity, expressed as a percentage. A high equivalent availability factor indicates effective plant programs and practices to maximize electrical generation.

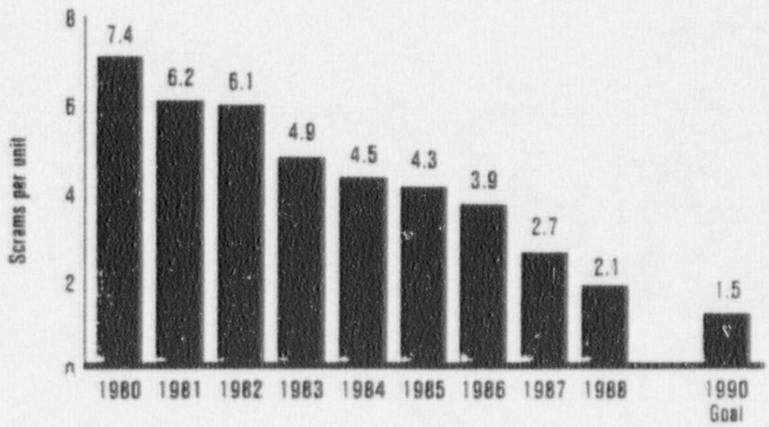
The industrywide average of 64.9 percent is an improvement over 1987, but is impacted by eight units that were shut down for most of 1988. Excluding these plants, industrywide equivalent availability rises to 70.7 percent.



Unplanned automatic scrams

The graph shows the average number of unplanned automatic scrams while the reactor is critical that occurred at nuclear plants operating with a 1988 capacity factor of 25 percent or greater. A low number is desirable, since scrams result from equipment failures or human error. 1980-83 scram numbers were estimated from the number of automatic scrams while the units were synchronized to the power grid. For years after 1983, the data was expanded to include unplanned automatic scrams that occurred anytime the reactor was critical.

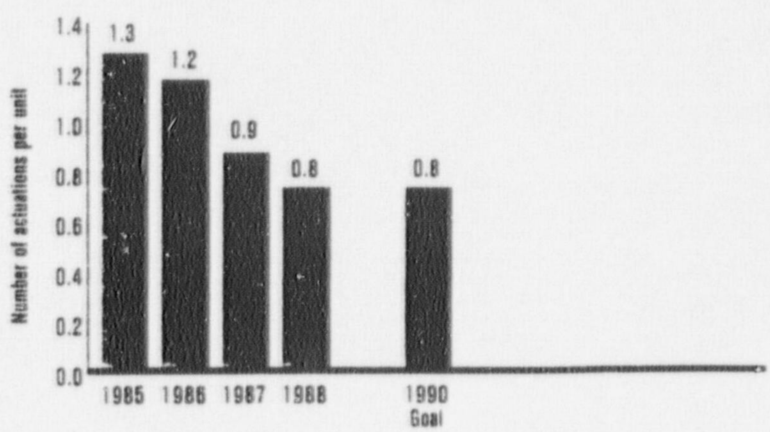
The 1988 industry average represents more than a threefold improvement over the 1980 data.



Unplanned safety system actuations

Unplanned safety system actuations comprise unplanned emergency core cooling system actuations and emergency AC power system actuations due to loss of power to a safeguards bus. Fewer actuations indicate greater care in plant operation, which contributes to a higher margin of safety.

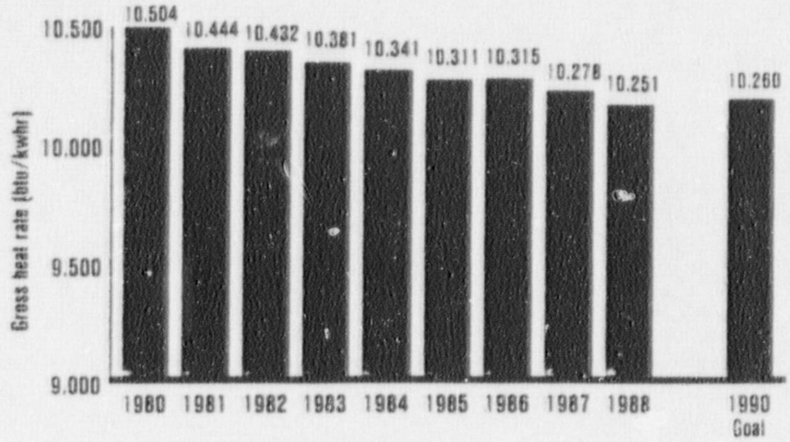
The industry as a whole reached the 1990 goal in this area in 1988.



Gross heat rate

Low gross heat rate, or Btu per kilowatt-hour, reflects emphasis on thermal efficiency and attention to detail in the maintenance of balance-of-plant systems. Efficient, well-tuned plants enable operators to detect abnormal trends and correct them early. The minimum heat rates attainable are a function of plant design.

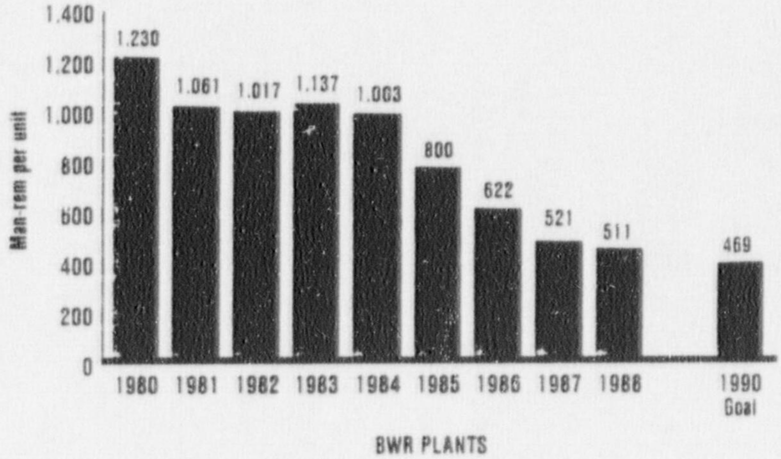
The industry as a whole surpassed the 1990 goal in this area in 1988.

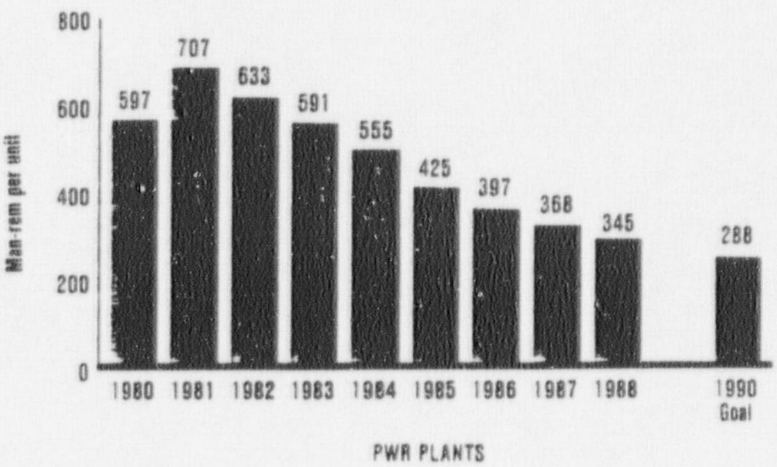


Collective radiation exposure per unit

This indicator examines the average collective radiation exposure in man-rem per unit for both boiling water reactors and pressurized water reactors since 1980. Low exposure indicates good management controls and attention and that radiological protection programs are effective.

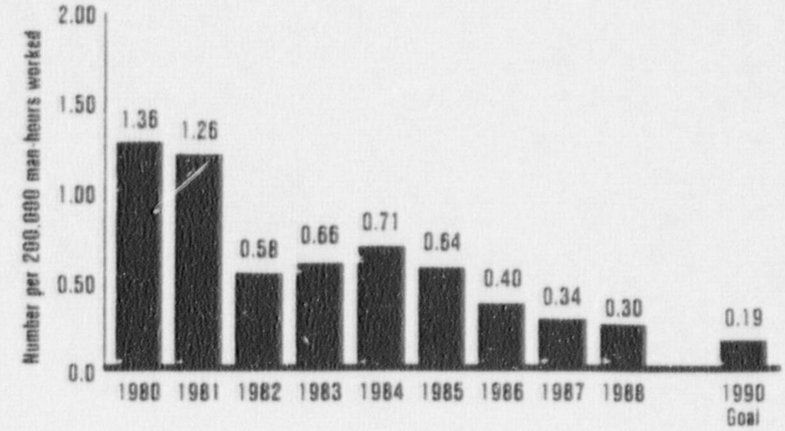
The 1988 industry averages for boiling water and pressurized water reactors represent improvements of 58 percent and 42 percent, respectively, over 1980 averages.





Lost-time accident rate

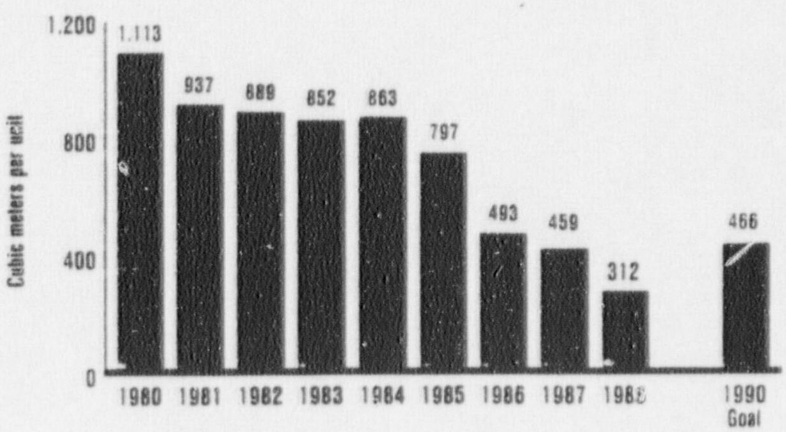
Lost-time accident rate is the number of worker injuries involving days away from work for every 200,000 man-hours (100 man-years) worked. This indicator reflects the industry's progress in improving industrial safety for workers over the past several years.



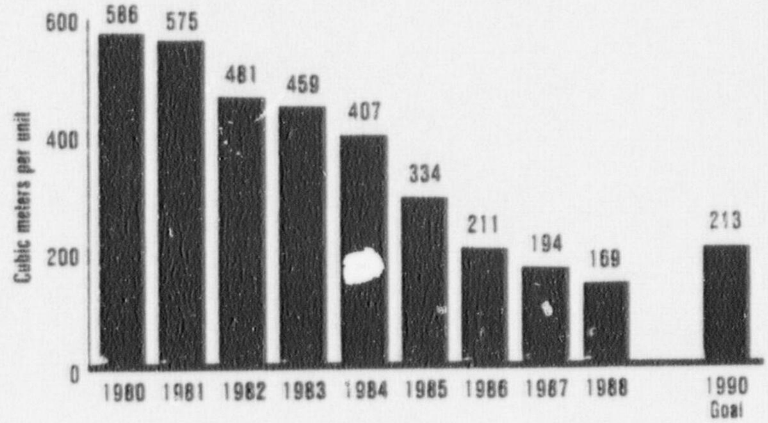
Low-level, solid radioactive waste per unit

The average volume of radioactive waste per unit for both boiling water and pressurized water reactors since 1980 is shown on these charts. Minimizing the production of radioactive wastes reduces storage, burial and transportation needs and thereby reduces the environmental impact of nuclear power. Management attention and good control over many plant activities are required to achieve this.

In 1988 the industry as a whole surpassed the 1990 goals in this category for both types of units—a 72 percent reduction since 1980.



BWR PLANTS



PWR PLANTS

UNPLANNED AUTOMATIC SCRAMS WHILE CRITICAL

PURPOSE

The purpose of the unplanned automatic scrams while critical indicator is to monitor industry performance in reducing the number of unplanned automatic reactor shutdowns. It provides an indication of how well a plant is operated and maintained because scrams result from equipment failures or personnel errors that cause undesirable and unplanned thermal-hydraulic or reactivity transients. Manual scrams and automatic scrams as a result of manual turbine trips to protect equipment or mitigate consequences of a transient are not counted because operator-initiated scrams and actions to protect equipment should not be discouraged.

DEFINITION

The indicator is defined as the number of unplanned automatic scrams (reactor protection system logic actuations) that occur while the reactor is critical. The indicator is further defined as follows:

- Unplanned means that the scram was not part of a planned test or evolution.

- Scram means the automatic shutdown of the reactor by a rapid insertion of all control rods that is caused by actuation of the reactor protection system. The scram signal may have resulted from exceeding a setpoint or may have been spurious.

What about Rod Drops?

- Automatic means that the initial signal that caused actuation of the reactor protection system logic was provided from one of the sensors monitoring plant parameters and conditions, rather than the manual scram switches or, in certain cases described below, manual turbine trip switches (or pushbuttoms) in the main control room.

- Critical means that during the steady-state condition of the reactor prior to the scram, the effective multiplication factor (k_{eff}) was equal to one.

DATA ELEMENTS

The basic data element required to determine each unit's value for this indicator is the number of unplanned automatic scrams while critical. In the U.S., the number of scrams is determined by INPO from an analysis of licensee event reports that are submitted by the utilities to the U.S. Nuclear Regulatory Commission.

In addition to the number of scrams, sufficient data must be available for calculating the cumulative capacity factor for the period (see Calculations and Data Qualification Requirements below). The data elements necessary for calculating cumulative capacity factor are as follows:

- gross electrical generation (defined under the thermal performance indicator)

- o gross maximum capacity (defined under the equivalent availability factor indicator)
- o period hours (defined under the equivalent availability factor (indicator)

CALCULATIONS

The unit and industry values for this indicator are determined as follows:

- o value for a unit = sum of unplanned automatic scrams while critical
- o value for the industry = average (mean) of the unit values

The cumulative capacity factor (CCF) for the period is calculated as follows:

- o $\text{CCF} = \dfrac{\text{(total gross electrical generation)} \times 100\%}{\text{(period hours)} \times \text{(gross maximum capacity)}}$

DATA QUALIFICATION REQUIREMENTS

Data collection begins January 1 of the first full calendar year following full power licensing for U.S. units (following commercial operation for international units). Data for U.S. units are included in the calculation of industry values beginning January 1 of the second full calendar year following full power licensing. However, in order to be included in the industry values, data elements must be provided for the entire time period, and the unit must have a cumulative capacity factor of at least 25 percent for the time period. Requiring this capacity factor minimizes the effects of plants that are shut down for long periods of time and whose limited data may not be statistically valid.

CLARIFYING NOTES

- o Scrams that are planned to occur as part of a test (e.g., a reactor protection system actuation test), or scrams that are part of a normal operation or evolution and are covered by controlled procedures, are not included.
- o Reactor protection system actuations that occur while all control rods are inserted are not counted because no control rod movement occurred as a result of the actuation.
- o Each scram caused by intentional manual tripping of the turbine will be analyzed. Those scrams which clearly involve a conscious decision by the operator to manually trip the turbine to protect important equipment or to minimize the effects of a transient will be counted as manual scrams.

- During a startup, shutdown, or changing power condition, the reactivity transients may cause the reactor to go subcritical or super-critical for a short period of time. However, the plant is considered critical for purposes of this indicator if the reactor was critical prior to the reactivity transient and may be assumed to return to a critical condition after the transient is completed (e.g., a plant is considered to remain critical after initial criticality is declared on a reactor startup, and to be critical until taken permanently subcritical on a reactor shutdown).

UNPLANNED SAFETY SYSTEM ACTUATIONS

PURPOSE

The purpose of the unplanned safety system actuation indicator is to monitor progress in reducing the number of occurrences of significant off-normal plant conditions. Emergency core cooling system (ECCS) actuations indicate events that are severe from a thermal-hydraulic perspective, while emergency AC power system actuations indicate a significant degradation of a vital support system. Limiting the number of unplanned safety system actuations indicates that a larger margin of nuclear safety is being maintained.

DEFINITION

This indicator is defined as the sum of the following safety system actuations:

- o the number of unplanned ECCS actuations that result from reaching an ECCS actuation setpoint or from a spurious/inadvertent ECCS signal
- o the number of unplanned emergency AC power system actuations that result from a loss of power to a safeguards bus

An unplanned safety system actuation occurs when an actuation setpoint for a safety system is reached or when a spurious or inadvertent signal is generated (ECCS only), and major equipment in the system is actuated. Unplanned means that the system actuation was not part of a planned test or evolution.

For PWRs, the ECCS actuations to be counted are actuations of the high pressure injection system, the low pressure injection system, or the safety injection tanks (accumulators, core flood tanks). For BWRs, the ECCS actuations to be counted are actuations of the high pressure coolant injection system, the low pressure coolant injection system, the high pressure core spray system, or the low pressure core spray system. Safety systems that may be used during normal plant operations (e.g., startup, shutdown) have not been included.

DATA ELEMENTS

The following data are required to determine each unit's value for this indicator:

- o the number of unplanned ECCS actuations that result from reaching an ECCS actuation setpoint or from a spurious/inadvertent ECCS signal
- o the number of unplanned actuations of emergency AC power systems that result from a loss of power to a safeguards bus

CALCULATIONS

The unit and industry values for this indicator are determined as follows:

- value for a unit = (number of ECCS actuations) ÷ (number of emergency AC power system actuations)
- value for the industry = average (mean) of the unit values

DATA QUALIFICATION REQUIREMENTS

Data collection begins January 1 of the first full calendar year following full power licensing for U.S. units (following commercial operation for international units). Data for U.S. units are included in the calculation of industry values beginning January 1 of the second full calendar year following full power licensing. In addition, data elements must be provided for the entire time period in order to be included in the industry values.

CLARIFYING NOTES

- Reactor protection system actuations are not included, because unplanned automatic reactor scrams are a separate performance indicator. Actuations of other safety-related systems such as auxiliary feedwater, reactor core isolation cooling, or residual heat removal, are not included since they are often actuated during normal operations (e.g., startup, shutdown) that do not represent significant off-normal plant conditions.
- Only one ECCS actuation is counted for each event that actuates one or more ECCS systems. For example, actuation of both the high pressure injection and the low pressure injection systems during the same event would count as one ECCS actuation. The intent is to count actuation events, not individual system actuations.
- For actuations to be counted, major equipment (e.g., pumps, diesels) must be actuated. For example, the spurious opening of one motor-operated valve in the high pressure injection system would not count as an ECCS actuation.
- Emergency AC power system actuations due to spurious or inadvertent starts of the emergency AC power source are not counted because these actuations represent no degradation in plant safety, and the inadvertent start does not cause a plant transient. ECCS system actuations due to spurious or inadvertent starts are counted because these actuations can result in a plant transient or equipment damage.
- When power is lost to one or more safeguards buses at a unit, only one emergency AC power system actuation is counted.
- Safety system actuations (as defined above) are counted during all plant conditions (e.g. operating, shutdown).

FORCED OUTAGE RATE

PURPOSE

The purpose of the forced outage rate indicator is to monitor industry progress in minimizing unplanned outages that are forced as a result of equipment failure or other conditions. This indicator reflects the effectiveness of plant programs and practices (e.g., preventive maintenance and the correction of design problems) in maintaining systems available for safe electrical generation. Experience has shown that units with high equivalent availability factors and low forced outage rates are often well maintained, follow good operating practices, and can be expected to have a higher margin of safety.

DEFINITION

This indicator is defined as the percentage of time that the unit was unavailable due to forced events compared to the time planned for electrical generation. Forced events are failures or other unplanned conditions that require removing the unit from service before the end of the next weekend. Forced events include startup failures and events initiated while the unit is in reserve shutdown (i.e., the unit is available but not in service).

DATA ELEMENTS

The following data is required to determine each unit's value for this indicator:

- forced outage hours: the time attributable to unit startup failures and unscheduled outages required before the end of the next weekend -- Forced outage hours include the time from opening the output breaker or declaring the unit unavailable for synchronizing to the grid, until the output breaker is closed or the unit is declared available in reserve shutdown.
- service hours: the time during which the unit is synchronized to the system

CALCULATIONS

The unit and industry values for this indicator are determined as follows:

- value for a unit = $\dfrac{\text{(forced outage hours for the time period)} \times 100\%}{\text{(sum of the forced outage hours and service hours for the time period)}}$
- value for the industry = average (mean) of the unit values

DATA QUALIFICATION REQUIREMENTS

Data collection begins January 1 of the first full calendar year following full power licensing for U.S. units (following commercial operation for international units). Data for U.S. units are included in the calculation of industry values beginning January 1 of the second full calendar year following full power licensing. In addition, data elements must be provided for at least 50 percent of the time period in order to be included in the industry values.

CLARIFYING NOTES

- If a unit is in an unplanned outage that was (or could be) deferred past the next weekend after the problem was identified, but could not have been deferred until the next planned outage, then the unit is in a "maintenance outage" rather than a forced outage. Also, a unit is in a "planned outage" rather than a forced outage if it is unavailable due to inspection, maintenance, testing, overhaul, or refueling which has been scheduled "well in advance." This usually means at the start of the current fuel cycle.

- In some cases, the opportunity exists during forced outages to perform some maintenance that would have been performed during the next planned outage. If the additional work extends the outage beyond that required for the forced outage, the remaining outage time is considered a planned or maintenance outage.

- If the duration of a "planned outage (basic)," i.e., the initially scheduled outage period, is extended to complete planned and scheduled work that was originally defined as a part of the planned outage, but could not be completed as scheduled, then the period of extension is called a "planned outage (extended)." Any condition identified during the planned outage (basic) that was not initially scheduled, requires corrective action to make the unit available, cannot be completed during the planned outage (extended) period, and cannot be deferred, should be considered a forced outage. The forced outage hours are counted from the time a planned outage (basic) was terminated until the unit is made available.

- The forced outage rate definition and calculation are consistent with that used by the Generating Availability Data System (GADS) of the North American Electric Reliability Council (NERC).

COLLECTIVE RADIATION EXPOSURE

PURPOSE

The purpose of the collective radiation exposure indicator is to monitor efforts to minimize total radiation exposure at each facility and in the industry as a whole. Radiation exposure has been demonstrated to be related to health risks. This parameter is a measure of the effectiveness of radiological protection programs in minimizing this health risk to plant workers.

DEFINITION

Collective radiation exposure is the total external whole-body dose received by all on-site personnel (including contractors and visitors) during a time period, as measured by the primary dosimeter, thermoluminescent dosimeter (TLD) or film badge. Exposure measured by direct reading dosimeters should be included only for those periods when more accurate data are not available. Collective radiation exposure is reported in units of man-rem.

DATA ELEMENTS

The total collective radiation exposure for the station is the only data required to determine each unit's value for this indicator. This indicator value is normally based on data obtained quarterly. However, since TLD or film badge information may not be available for the current quarter, the following data is reported:

- o total man-rem for the current quarter (TLD, film badge, or direct reading dosimeter)
- o total man-rem for the previous quarter (TLD or film badge only) -- This replaces the data for the previous quarter which may have used direct reading dosimeter values.

The conversion from Sieverts to rem is 1 Sievert = 100 rem.

CALCULATIONS

The unit and industry values for this indicator are determined as follows:

- o value for a unit (for periods of one year or less) = total unit collective radiation exposure during the period

  To allow more meaningful comparison of unit performance, collective radiation exposure is presented for a three-year period to minimize the impact of annual variations due to refueling and planned maintenance outages. The unit values for the three-year period are determined as follows:

three-year unit value = average (mean) of the annual unit values

- value for the industry = average (mean) of the unit values

DATA QUALIFICATION REQUIREMENTS

Data collection begins January 1 of the first full calendar year following full power licensing for U.S. units (following commercial operation for international units). Data for U.S. units are included in the calculation of industry values beginning January 1 of the second full calendar year following full power licensing. In addition, data elements must be provided for at least 50 percent of the time period in order to be included in the industry values.

CLARIFYING NOTES

- For multi-unit stations that do not track collective radiation exposure separately for each unit, unit values are estimated by dividing the station data by the number of operating units at the station. This allows for more meaningful comparisons among single and multi-unit stations.

- Due to design differences, this indicator is presented by reactor type (e.g., BWR or PWR).

- This indicator measures the total exposure received on-site by all personnel and therefore includes contractors and visitors.