Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

June 1, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Zion Station Units 1 and 2
LaSalle County Station Unit 1 and 2
Byron Station Units 1 and 2
Braidwood Station Units 1 and 2
Supplemental Response to Generic Letter No. 83-28
"Required Actions Based On Generic Implications of Salem ATWS Events"
NRC Docket Nos. 50-237/249, 50-254/265, 50-295/304, 50-373/374, 50-454/455 and 50-456/457

Reference (a): Generic Letter No. 83-28 D. G. Eisenhut letter to All OLs and CPs dated July 8, 1983 (NL-83-0003)

(b): P. L. Barnes to H. R. Denton letter dated November 5, 1983 (NL-83-0520)

(c): P. L. Barnes to H. R. Denton letter dated February 29, 1984 (NL-84-0254)

Dear Mr. Keppler:

Reference (a) requested that the Commonwealth Edison Company provide a written report of the status of current conformance with the positions contained in the subject letter. References (b) and (c) provided that status.

The attachments to this letter supplements the positions as reported in References (b) and (c).

To the best of my knowledge and belief, the statements contained in the Attachment are true and correct. In some respects, these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison employees, consultants and contractors. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

A055
1/1

Please address any questions that you or your staff may have concerning our response to Generic Letter No. 83-28 to this office.

Respectfully,

P. L. Barnes

P. L. Barnes
Nuclear Licensing Administrator

Attachment

cc: U.S. NRC, Document Control Desk
Washington, DC 20555

J. G. Keppler - RIII
RIII Inspectors: D, QC, Z, LSC, BY, BW

SUBSCRIBED and SWORN to
before me this 1st day
of June, 1984
Rosalie A. Pienta
Notary Public

8461N

ATTACHMENT

COMMONWEALTH EDISON COMPANY

Supplemental Response To Generic Letter No. 83-28
"Required Actions Based on Generic Implications
of Salem ATWS Events"

Part I	Dresden Station Units 2 and 3
Part II	Quad Cities Station Units 1 and 2
Part III	Zion Station Units 1 and 2
Part IV	LaSalle County Station Units 1 and 2
Part V	Byron Station Units 1 and 2
Part VI	Braidwood Station Units 1 and 2

06-01-84

50-237

SUPPLEMENTAL RESPONSE TO GENERIC LTR 83-28 " REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS

— NOTICE —

THE ATTACHED FILES ARE OFFICIAL RECORDS OF THE DIVISION OF DOCUMENT CONTROL. THEY HAVE BEEN CHARGED TO YOU FOR A LIMITED TIME PERIOD AND MUST BE RETURNED TO THE RECORDS FACILITY BRANCH 016. PLEASE DO NOT SEND DOCUMENTS CHARGED OUT THROUGH THE MAIL. REMOVAL OF ANY PAGE(S) FROM DOCUMENT FOR REPRODUCTION MUST BE REFERRED TO FILE PERSONNEL. 50-237

6/1/84

DEADLINE RETURN DATE 8406050418

RECORDS FACILITY BRANCH

Part I - Dresden Station Units 2 and 3

Supplemental Response to Generic Letter 83-28

06-01-84

Items 2.1 and 2.2 <u>Equipment Classification and Vendor Interface (Reactor Trip System Components); (Programs For All Safety-Related Components)</u>

Commonwealth Edison actively participated in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The final report presented conclusions which were unanimously approved by the 56 utilities involved. A copy of this report is attached for your information.

The NUTAC early on recognized that individual utilities have the greatest experience with the equipment they have in their plants and are most cognizant of its application, surviellance and maintenance history.

The next logical step was to determine how this information was shared. An examination of the current mechanisms for information exchange led to the conclusion that the existing utility/vendor and utility/regulator communications, together with the SEE-IN and NPRDS programs managed by INPO, provide for an effective information exchange program. Recommendations for proper integration and minor enhancements were included in the report.

Commonwealth Edison endorses the conclusions of the NUTAC final report and will use it to develop plant specific procedures to implement the recommended program. These procedures will be in place by July 1, 1985. It is our belief that this approach satisfies the intent of Generic Letter 83-28 Sections 2.1 and 2.2 on Vendor Interface.

Item 4.5 Reactor Trip System Reliability (System Functional Testing)

Commonwealth Edison participated in the BWR Owner's Group review of Reactor Trip System functional testing. Their final report has been issued and reviewed by Dresden on a plant specific basis.

The Dresden Reactor Protection System (RPS or Reactor Trip System) design complies with all applicable regulatory requirements for the reactor protection system.

Consistent with the Technical Specifications, on-line channel functional testing is performed on the multiple and diverse reactor transient trip sensors. During the required trip sensor channel tests each scram contactor which actuates the scram pilot solenoid valves is tested. The simple operation of the scram contactors minimizes concerns of wear and frequent testing assures that any failures are detected early. The Scram Pilot Solenoid Valves which are actuated by the scram contactors are all tested regularly. Redundant Electrical Protection Assemblies (EPA's) which protect the Scram Pilot Solenoid Valves from low voltage chattering (and the associated potential conse- quence of accelerated wear) will be functionally tested subsequent to Technical Specification update. These surveillance testing requirements related to the Scram Pilot Solenoid Valves assure that the probability of undetected failures of these independently acting solenoid valves is small. In summary, the current Reactor Protection System on-line surveillance testing requirement, in conjunction with multiple and diverse scram sensors, assure that the probability of failure of enough control rods to prevent reactor shutdown is negligible.

Channel functional tests are performed on-line for the following sensor trips:

- Reactor Vessel Dome Pressure-High
- Reactor Vessel Water Level-Low
- Main Steam Line Isolation Valve-Closure
- Main Steam Line Radiation-High
- Drywell Pressure-High
- Turbine Control Valve Fast Closure, Control Oil Pressure-Low
- Turbine Stop Valve-Closure
- Condenser Low Vacuum

Channel functional tests are also performed for Average Power Range Monitors and Intermediate Range Monitors.

In References 1 and 2, it is shown that each of the above plant variables used to initiate a protective function is backed up by a completely different plant variable. In fact, for the most frequent transients, scram is initiated by three diverse sensors in all but one case (regulator

failure-primary pressure increase which is initiated by two diverse sensors). This indicates that adequate redundancy exists in the design to provide protection against multiple independent sensor failures. Also, diversity among sensor types reduces the potential for common cause failures, failures due to human error, and increases in failure rate due to wearout.

Each sensor channel functional test includes full actuation of the associated logic, the two output scram contactors in each channel, and the individual CRD scram air pilot valve solenoids for the associated logic division (solenoids from both logic Division A and B are required for scram initiation).

The most credible failures within the RPS logic will de-energize a set of scram solenoids which causes a half scram, i.e., one of the two scram solenoids required for scram initiation is de-energized at all hydraulic control units. These failures would be "SAFE" failures that would increase the probability of plant shutdown.

The less credible logic failures which prevent a channel from de-energizing will be detected during channel functional test in compli- ance with Technical Specification requirements. The tests described above ensure that an increase in failure rate due to a wearout condition or a common cause failure potential could be detected early and corrective action taken before the failure condition becomes systematic.

Other channel functional tests include quarterly testing of the Scram Discharge Volume (SDV) Water Level-High trip and manual scram trip, and test of the reactor mode switch in the shutdown position every refueling. The first two trips involve on-line testing and the latter mode switch test can only be conducted during reator shutdown. The manual scram trip can be tested on-line without creating a scram.

The testing of the SDV Water Level-High trip will be considered adequate when designed redundancy and diversity are incorporated into the system. There will be two diverse and redundant sets of level sensors which scram the reactor in the unlikely event of high water level in the SDV during power operation. These trips are designed to allow sufficient scram water discharge volume given the scram trip point is reached.

The probability of independent failure of enough rods to prevent shutdown is negligible. The most unlikely type of failure would be some common cause mechanism that if undetected over a long period of time could cause unsafe shutdown. The Technical Specification surveillance requirements adequately ensures that a failure mechanism affecting several individual drives (considered to be very remote) would not go undetected. One of the major features that ensures that several drives do not fail at one time due to wearout or a common cause is the staggered maintenance and overhaul of selected degraded CRDs or Hydraulic

Control Units (HCUs) at refueling outages. This ensures a mix of drives by age, component lot, maintenance time and servicing personnel, and testing.

The scram insertion time tests include, in addition to drive timing and insertion capability, a test of operability of the HCU scram insert and discharge valves including associated scram air pilot valves. As stated in the previous paragraph, the required testing given in the Technical Specifications ensures that a systematic failure mechanism in the HCUs would be detected early enough and corrective action taken before the condition becomes a critical failure preventing scram.

The Generic Letter requests on-line functional testing of the diverse trip system at each station. Reviewing the consequences of testing the backup scram valves, it is evident that the operator must scram the unit. Numerous, unnecessary scrams decrease the integrity and reliability of the plant equipment. However, it is believed that the backup scram valves must be able to operate. To ensure operability, the backup scram valves shall be tested every refueling outage in accordance with tests to be incorporated in station procedures.

REFERENCES

1. NEDO-1-189, "An Analysis of Function Common-Mode Failures in GE BWR Protection and Control Instrumentation," L.G. Frederick, et al, July 1970.

2. "BWR Scram System Reliability Analysis," W.P. Sullivan, et al, September 30, 1976 (Transmitted in letter from E.A. Hughes (GE) to D.F. Ross (NRC), "General Electric Company ATWS Reliability Report," September 30, 1976).

Part II - Quad Cities Station Units 1 and 2

Supplemental Response to Generic Letter 83-28

06-01-84

Items 2.1 and 2.2 <u>Equipment Classification and Vendor Interface (Reactor Trip System Components); (Programs For All Safety-Related Components)</u>

Commonwealth Edison actively participated in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The final report presented conclusions which were unanimously approved by the 56 utilities involved. A copy of this report is attached for your information.

The NUTAC early on recognized that individual utilities have the greatest experience with the equipment they have in their plants and are most cognizant of its application, surviellance and maintenance history.

The next logical step was to determine how this information was shared. An examination of the current mechanisms for information exchange led to the conclusion that the existing utility/vendor and utility/regulator communications, together with the SEE-IN and NPRDS programs managed by INPO, provide for an effective information exchange program. Recommendations for proper integration and minor enhancements were included in the report.

Commonwealth Edison endorses the conclusions of the NUTAC final report and will use it to develop plant specific procedures to implement the recommended program. These procedures will be in place by July 1, 1985. It is our belief that this approach satisfies the intent of Generic Letter 83-28 Sections 2.1 and 2.2 on Vendor Interface.

Item 4.5 Reactor Trip System Reliability (System Functional Testing)

Commonwealth Edison participated in the BWR Owner's Group review of Reactor Trip System functional testing. Their final report has been issued and reviewed by Quad Cities on a plant specific basis.

The Quad Cities Reactor Protection System (RPS or Reactor Trip System) design complies with all applicable regulatory requirements for the reactor protection system.

Consistent with the Technical Specifications, on-line channel functional testing is performed on the multiple and diverse reactor transient trip sensors. During the required trip sensor channel tests each scram contactor which actuates the scram pilot solenoid valves is tested. The simple operation of the scram contactors minimizes concerns of wear and frequent testing assures that any failures are detected early. The Scram Pilot Solenoid Valves which are actuated by the scram contactors are all tested regularly. Redundant Electrical Protection Assemblies (EPA's) which protect the Scram Pilot Solenoid Valves from low voltage chattering (and the associated potential conse- quence of accelerated wear) will be functionally tested subsequent to Technical Specification update. These surveillance testing requirements related to the Scram Pilot Solenoid Valves assure that the probability of undetected failures of these independently acting solenoid valves is small. In summary, the current Reactor Protection System on-line surveillance testing requirement, in conjunction with multiple and diverse scram sensors, assure that the probability of failure of enough control rods to prevent reactor shutdown is negligible.

Channel functional tests are performed on-line for the following sensor trips:

- Reactor Vessel Dome Pressure-High
- Reactor Vessel Water Level-Low
- Main Steam Line Isolation Valve-Closure
- Main Steam Line Radiation-High
- Drywell Pressure-High
- Turbine Control Valve Fast Closure, Control Oil Pressure-Low
- Turbine Stop Valve-Closure
- Condenser Low Vacuum

Channel functional tests are also performed for Average Power Range Monitors and Intermediate Range Monitors.

In References 1 and 2, it is shown that each of the above plant variables used to initiate a protective function is backed up by a completely different plant variable. In fact, for the most frequent transients, scram is initiated by three diverse sensors in all but one case (regulator failure-primary pressure increase which is initiated by two

diverse sensors). This indicates that adequate redundancy exists in the design to provide protection against multiple independent sensor failures. Also, diversity among sensor types reduces the potential for common cause failures, failures due to human error, and increases in failure rate due to wearout.

Each sensor channel functional test includes full actuation of the associated logic, the two output scram contactors in each channel, and the individual CRD scram air pilot valve solenoids for the associated logic division (solenoids from both logic Division A and B are required for scram initiation).

The most credible failures within the RPS logic will de-energize a set of scram solenoids which causes a half scram, i.e., one of the two scram solenoids required for scram initiation is de-energized at all hydraulic control units. These failures would be "SAFE" failures that would increase the probability of plant shutdown.

The less credible logic failures which prevent a channel from de-energizing will be detected during channel functional test in compli- ance with Technical Specification requirements. The tests described above ensure that an increase in failure rate due to a wearout condition or a common cause failure potential could be detected early and corrective action taken before the failure condition becomes systematic.

Other channel functional tests include quarterly testing of the Scram Discharge Volume (SDV) Water Level-High trip and manual scram trip, and test of the reactor mode switch in the shutdown position every refueling. The first two trips involve on-line testing and the latter mode switch test can only be conducted during reator shutdown. The manual scram trip can be tested on-line without creating a scram.

The testing of the SDV Water Level-High trip will be considered adequate when designed redundancy and diversity are incorporated into the system. There will be two diverse and redundant sets of level sensors which scram the reactor in the unlikely event of high water level in the SDV during power operation. These trips are designed to allow sufficient scram water discharge volume given the scram trip point is reached.

The probability of independent failure of enough rods to prevent shutdown is negligible. The most unlikely type of failure would be some common cause mechanism that if undetected over a long period of time could cause unsafe shutdown. The Technical

Specifications surveillance requirements adequately ensures that a failure mechanism affecting several individual drives (considered to be very remote) would not go undetected. One of the major features that ensures that several drives do not fail at one time due to wearout or a common cause is the staggered maintenance and overhaul of selected degraded CRDs or Hydraulic Control Units (HCUs) at refueling outages. This ensures a mix of drives by age, component lot, maintenance time and servicing personnel, and testing.

The scram insertion time tests include, in addition to drive timing and insertion capability, a test of operability of the HCU scram insert and discharge valves including associated scram air pilot valves. As stated in the previous paragraph, the required testing given in the Technical Specifications ensures that a systematic failure mechanism in the HCUs would be detected early enough and corrective action taken before the condition becomes a critical failure preventing scram.

The Generic Letter requests on-line functional testing of the diverse trip system at each station. Reviewing the consequences of testing the backup scram valves, it is evident that the operator must scram the unit. Numerous, unnecessary scrams decrease the integrity and reliability of the plant equipment. However, it is believed that the backup scram valves must be able to operate. To ensure operability, the backup scram valves shall be tested every refueling outage in accordance with tests to be incorporated in station procedures.

REFERENCES

1. NEDO-1-189, "An Analysis of Function Common-Mode Failures in GE BWR Protection and Control Instrumentation," L.G. Frederick, et al, July 1970.

2. "BWR Scram System Reliability Analysis," W.P. Sullivan, et al, September 30, 1976 (Transmitted in letter from E.A. Hughes (GE) to D.F. Ross (NRC), "General Electric Company ATWS Reliability Report," September 30, 1976).

Part III - Zion Station Units 1 and 2

Supplemental Response to Generic Letter 83-28

06-01-84

Items 2.1 and 2.2 <u>Equipment Classification and Vendor Interface (Reactor Trip System Components); (Programs For All Safety-Related Components)</u>

Commonwealth Edison actively participated in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The final report presented conclusions which were unanimously approved by the 56 utilities involved. A copy of this report is attached for your information.

The NUTAC early on recognized that individual utilities have the greatest experience with the equipment they have in their plants and are most cognizant of its application, surviellance and maintenance history.

The next logical step was to determine how this information was shared. An examination of the current mechanisms for information exchange led to the conclusion that the existing utility/vendor and utility/regulator communications, together with the SEE-IN and NPRDS programs managed by INPO, provide for an effective information exchange program. Recommendations for proper integration and minor enhancements were included in the report.

Commonwealth Edison endorses the conclusions of the NUTAC final report and will use it to develop plant specific procedures to implement the recommended program. These procedures will be in place by July 1, 1985. It is our belief that this approach satisfies the intent of Generic Letter 83-28 Sections 2.1 and 2.2 on Vendor Interface.

Item 3.2 Post Maintenance Testing (All Other Safety Related Components)

The Station continues to review the Westinghouse Technical Bulletins and Data Letters and expects to complete the project by June 30, 1984 with any required procedure changes being in place at approximately the same time.

Part IV

LaSalle County Station Units 1 and 2

Supplemental Response to Generic Letter 83-28

06-01-84

Items 2.1 and 2.2 <u>Equipment Classification and Vendor Interface (Reactor Trip System Components); (Programs For All Safety-Related Components)</u>

Commonwealth Edison actively participated in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The final report presented conclusions which were unanimously approved by the 56 utilities involved. A copy of this report is attached for your information.

The NUTAC early on recognized that individual utilities have the greatest experience with the equipment they have in their plants and are most cognizant of its application, surviellance and maintenance history.

The next logical step was to determine how this information was shared. An examination of the current mechanisms for information exchange led to the conclusion that the existing utility/vendor and utility/regulator communications, together with the SEE-IN and NPRDS programs managed by INPO, provide for an effective information exchange program. Recommendations for proper integration and minor enhancements were included in the report.

Commonwealth Edison endorses the conclusions of the NUTAC final report and will use it to develop plant specific procedures to implement the recommended program. These procedures will be in place by July 1, 1985. It is our belief that this approach satisfies the intent of Generic Letter 83-28 Sections 2.1 and 2.2 on Vendor Interface.

Item 4.5 Reactor Trip System Reliability (System Functional Testing)

Commonwealth Edison participated in the BWR Owner's Group review of Reactor Trip System functional testing. Their final report has been issued and reviewed by LaSalle County on a plant specific basis.

The LaSalle County Reactor Protection System (RPS or Reactor Trip System) design complies with all applicable regulatory requirements for the reactor protection system.

Consistent with the Technical Specifications, on-line channel functional testing is performed on the multiple and diverse reactor transient trip sensors, Average Power Range Monitor and Intermediate Range Monitor Reactor Trip signal channels, and the multiple and diverse scram Discharge Volume High Water level trips. During the required trip sensor channel tests discussed above, each scram contactor which actuates the scram pilot solenoid valves is tested. The simple operation of the scram contactors minimizes concerns of wear and frequent testing assures that any failures are detected early. The Scram Pilot Solenoid Valves which are actuated by the scram contactors are all tested regularly. Redundant Electrical Protection Assemblies (EPA's) which protect the Scram Pilot Solenoid Valves from low voltage chattering (and the associated potential consequence of accelerated wear) are functionally tested. These surveillance testing requirements related to the Scram Pilot Solenoid Valves assure that the probability of undetected failures of these independently acting solenoid valves is small. In summary, the current Reactor Protection System on-line surveillance testing requirement, in conjunction with multiple and diverse scram sensors, assure that the probability of failure of enough control rods to prevent reactor shutdown is negligible.

Channel functional tests are performed on-line for the following sensor trips:

- Reactor Vessel Dome Pressure-High
- Reactor Vessel Water Level-Low
- Main Steam Line Isolation Valve-Closure
- Main Steam Line Radiation-High
- Drywell Pressure-High
- Turbine Control Valve Fast Closure, Control Oil Pressure-Low
- Turbine Stop Valve-Closure

Channel functional tests are also performed for Average Power Range Monitors and Intermediate Range Monitors.

In References 1 and 2, it is shown that each of the above plant variables used to initiate a protective function is backed up by a completely different plant variable. In fact, for the most frequent transients, scram is initiated

by three diverse sensors in all but one case (regulator failure-primary pressure increase which is initiated by two diverse sensors). This indicates that adequate redundancy exists in the design to provide protection against multiple independent sensor failures. Also, diversity among sensor types reduces the potential for common cause failures, failures due to human error, and increases in failure rate due to wearout.

Each sensor channel functional test includes full actuation of the associated logic, the two output scram contactors in each channel, and the individual CRD scram air pilot valve solenoids for the associated logic division (solenoids from both logic Division A and B are required for scram initiation).

The most credible failures within the RPS logic will de-energize a set of scram solenoids which causes a half scram, i.e., one of the two scram solenoids required for scram initiation is de-energized at some or all hydraulic control units. These failures would be "SAFE" failures that would increase the probability of plant shutdown.

The less credible logic failures which prevent a channel from de-energizing will be detected during channel functional test in compli- ance with Technical Specification requirements. The tests described above ensure that an increase in failure rate due to a wearout condition or a common cause failure potential could be detected early and corrective action taken before the failure condition becomes systematic.

Other channel functional tests include testing of the Scram Discharge Volume (SDV) Water Level-High trip and manual scram trip, and test of the reactor mode switch in the shutdown position every refueling. The first two trips involve on-line testing and the latter mode switch test can only be conducted during reator shutdown. The manual scram trip can be tested on-line without creating a scram.

The testing of the SDV Water Level-High trip is considered adequate based on the current designed redundancy and diversity are incorporated into the system. There are two diverse and redundant sets of level sensors which scram the reactor in the unlikely event of high water level in the SDV during power operation. These trips are designed to allow sufficient scram water discharge volume given the scram trip point is reached.

The probability of independent failure of enough rods to prevent shutdown is negligible. The most unlikely type of failure would be some common cause mechanism that if undetected over a long period of time could cause unsafe shutdown. The Technical Specification surveillance requirements adequately ensures that a failure mechanism affecting several individual drives (considered to be very remote) would not go undetected. One of the major features that ensures that several drives do not fail at one time due to wearout or a common cause is the staggered maintenance and overhaul of selected degraded CRDs or Hydraulic Control Units (HCUs) at refueling outages. This ensures a mix of drives by age, component lot, maintenance time and servicing personnel, and testing.

The scram insertion time tests include, in addition to drive timing and insertion capability, a test of operability of the HCU scram insert and discharge valves including associated scram air pilot valves. As stated in the previous paragraph, the required testing given in the Technical Specifications ensures that a systematic failure mechanism in the HCUs would be detected early enough and corrective action taken before the condition becomes a critical failure preventing scram.

The Generic Letter requests on-line functional testing of the diverse trip system at each station. Reviewing the consequences of testing the backup scram valves, it is evident that the operator must scram the unit. Numerous, unnecessary scrams decrease the integrity and reliability of the plant equipment. However, it is believed that the backup scram valves must be able to operate. To ensure operability, the backup scram valves are tested every refueling outage in accordance with station procedures.

REFERENCES

1. NEDO-1-189, "An Analysis of Function Common-Mode Failures in GE BWR Protection and Control Instrumentation," L.G. Frederick, et al, July 1970.

2. "BWR Scram System Reliability Analysis," W.P. Sullivan, et al, September 30, 1976 (Transmitted in letter from E.A. Hughes (GE) to D.F. Ross (NRC), "General Electric Company ATWS Reliability Report," September 30, 1976).

Part V

Byron Station Units 1 and 2

Supplemental Response to Generic Letter 83-28

06-01-84

Items 2.1 and 2.2 <u>Equipment Classification and Vendor Interface (Reactor Trip System Components); (Programs For All Safety-Related Components)</u>

Commonwealth Edison actively participated in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The final report presented conclusions which were unanimously approved by the 56 utilities involved. A copy of this report is attached for your information.

The NUTAC early on recognized that individual utilities have the greatest experience with the equipment they have in their plants and are most cognizant of its application, surviellance and maintenance history.

The next logical step was to determine how this information was shared. An examination of the current mechanisms for information exchange led to the conclusion that the existing utility/vendor and utility/regulator communications, together with the SEE-IN and NPRDS programs managed by INPO, provide for an effective information exchange program. Recommendations for proper integration and minor enhancements were included in the report.

Commonwealth Edison endorses the conclusions of the NUTAC final report and will use it to develop plant specific procedures to implement the recommended program. These procedures will be in place by July 1, 1985. It is our belief that this approach satisfies the intent of Generic Letter 83-28 Sections 2.1 and 2.2 on Vendor Interface.

Item 4.2 <u>Reactor Trip System Reliability (Preventative Maintenance and Surveillance Program For Reactor Trip Breakers)</u>

Life cycle testing of the shunt trip attachment and the undervoltage trip attachment of the reactor trip switchgear is being conducted by Westinghouse for the Westinghouse Owners Group. This program is aimed toward establishing the service life of these devices, and substantiating periodic test requirements with proper maintenance. The results of this program will be factored into maintenance, replacement and qualification programs. The test program is scheduled for completion in the fourth quarter of 1984.

Byron Station will review the results and recommendations of this program for incorporation into Station procedures.

Item 4.3 <u>Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)</u>

The Westinghouse Owners Group recommended modification will be implemented on both units and should be installed by the first refueling outage.

Item 4.5 <u>Reactor Trip System Reliability (System Function Testing)</u>

The modification referred to in Item 4.3 will incorporate the necessary test features.

Part VI

Braidwood Station Units 1 and 2

Supplemental Response to Generic Letter 83-28

06-01-84

Items 2.1 and 2.2 <u>Equipment Classification and Vendor Interface (Reactor Trip System Components); (Programs For All Safety-Related Components)</u>

Commonwealth Edison actively participated in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The final report presented conclusions which were unanimously approved by the 56 utilities involved. A copy of this report is attached for your information.

The NUTAC early on recognized that individual utilities have the greatest experience with the equipment they have in their plants and are most cognizant of its application, surviellance and maintenance history.

The next logical step was to determine how this information was shared. An examination of the current mechanisms for information exchange led to the conclusion that the existing utility/vendor and utility/regulator communications, together with the SEE-IN and NPRDS programs managed by INPO, provide for an effective information exchange program. Recommendations for proper integration and minor enhancements were included in the report.

Commonwealth Edison endorses the conclusions of the NUTAC final report and will use it to develop plant specific procedures to implement the recommended program. These procedures will be in place by July 1, 1985. It is our belief that this approach satisfies the intent of Generic Letter 83-28 Sections 2.1 and 2.2 on Vendor Interface.

Item 4.2 <u>Reactor Trip System Reliability (Preventative Maintenance and Surveillance Program For Reactor Trip Breakers)</u>

Life cycle testing of the shunt trip attachment and the undervoltage trip attachment of the reactor trip switchgear is being conducted by Westinghouse for the Westinghouse Owners Group. This program is aimed toward establishing the service life of these devices, and substantiating periodic test requirements with proper maintenance. The results of this program will be factored into maintenance, replacement and qualification programs. The test program is scheduled for completion in the fourth quarter of 1984.

Braidwood Station will review the results and recommendations of this program for incorporation into Station procedures.

Item 4.3 <u>Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)</u>

The Westinghouse Owners Group recommended modification will be implemented on both units and should be installed prior to fuel load.

Item 4.5 <u>Reactor Trip System Reliability (System Function Testing)</u>

The modification referred to in Item 4.3 will incorporate the necessary test features.

Nuclear
Utility
Task
Action
Committee

nutac

ON GENERIC LETTER 83-28, SECTION 2.2.2

Vendor Equipment Technical Information Program

March 1984

INPO 84-010 (NUTAC)

8405090007 XA

Vendor Equipment

Technical

Information Program

Developed By

Nuclear Utility Task Action Committee

for

Generic Letter 83-28, Section 2.2.2

INPO 84-010

(NUTAC)

March 1984

Publications produced by a nuclear utility task action committee (NUTAC) represent a consensus of the utilities participating in the NUTAC. These publications are not intended to be interpreted as industry standards. Instead, the publications are offered as suggested guidance with the understanding that individual utilities are not obligated to use the suggested guidance.

This publication has been produced by the NUTAC on Generic Letter 83-28, Section 2.2.2., with the support of the Institute of Nuclear Power Operations (INPO). The officers of this NUTAC were Chairman Edward P. Griffing and Vice Chairman Walter E. Andrews.

Utilities that participated in this NUTAC include the following:

Alabama Power Company
American Electric Power Service Corporation
Arizona Public Service Company
Arkansas Power & Light Company
Baltimore Gas and Electric Company
Boston Edison Company
Carolina Power & Light Company
Cincinnati Gas & Electric Company
The Cleveland Electric Illuminating Company
Commonwealth Edison Company
Consolidated Edison Company of New York, Inc.
Consumers Power Company
The Detroit Edison Company
Duke Power Company
Duquesne Light Company
Florida Power Corporation
Florida Power & Light Company
GPU Nuclear Corporation
Georgia Power Company
Gulf States Utilities Company
Houston Lighting & Power Company
Illinois Power Company
Iowa Electric Light and Power Company
Kansas Gas and Electric Company
Long Island Lighting Company
Louisiana Power & Light Company
Maine Yankee Atomic Power Company
Mississippi Power & Light Company
Nebraska Public Power District
New York Power Authority
Niagara Mohawk Power Corporation
Northeast Utilities
Northern States Power Company
Omaha Public Power District
Pacific Gas and Electric Company
Pennsylvania Power & Light Company
Philadelphia Electric Company
Portland General Electric Company
Public Service Company of Colorado
Public Service Company of Indiana, Inc.
Public Service Company of New Hampshire
Public Service Electric and Gas Company
Rochester Gas and Electric Corporation
Sacramento Municipal Utility District
South Carolina Electric & Gas Company
Southern California Edison Company
Tennessee Valley Authority
Texas Utilities Generating Company
The Toledo Edison Company
Union Electric Company
Vermont Yankee Nuclear Power Corporation
Virginia Electric and Power Company
Washington Public Power Supply System
Wisconsin Electric Power Company
Wisconsin Public Service Corporation
Yankee Atomic Electric Company

EXECUTIVE SUMMARY

This report was prepared by the Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28 "Required Actions Based on Generic Implications of Salem ATWS Events," Section 2.2.2. It describes the Vendor Equipment Technical Information Program (VETIP) developed by the NUTAC in response to the concerns on vendor information and interface addressed in Section 2.2.2 of the generic letter. VETIP is a program that enhances information exchange and evaluation among utilities constructing or operating nuclear power plants and provides for more effective vendor interface.

The NUTAC was comprised of representatives of 56 utilities that are members of the Institute of Nuclear Power Operations (INPO). Staff support for the NUTAC was provided by INPO. This report unanimously presents the final conclusions of the NUTAC and is provided to assist individual utilities in developing specific programs to meet the intent of the generic letter.

Generic Letter 83-28 was developed following investigations by the NRC on the Salem events. As a result of these investigations, the NRC determined that better control and utilization of information regarding safety-related components might have helped to prevent these events. The NUTAC identified a program to better ensure that plant personnel have timely access to such information.

The NUTAC efforts were guided by the recognition that individual utilities have the greatest experience with and are most cognizant of the application of safety-related equipment. Vendor involvement with such equipment is generally greatest during construction and initial operation of the plant. Vendors are not familiar with the surveillance or maintenance histories, nor with the application of the equipment or its environment. This type of information is most readily available at the plant level within individual utilities.

Based on this recognition, the NUTAC investigated the mechanisms currently available to facilitate information exchange among utilities. The NUTAC identified four activities that currently address information about safety-

related components. These are routine utility/vendor and utility/ regulator interchange, and the SEE-IN and NPRDS programs managed by INPO.

It was the assessment of the NUTAC that these existing activities, if properly integrated and implemented, would provide a framework for an overall program to ensure effective communication of safety-related information among all utilities. Accordingly, the program developed to accomplish this goal (VETIP) uses the existing efforts as elements of a more comprehensive program.

The VETIP combines these existing programs, incorporating enhancements, with a coordinated program within each utility. A key element of the VETIP is the development by each utility of an active internal program to contribute information to the NPRDS and SEE-IN Programs and to use the results of these programs.

The effectiveness of the VETIP will be determined by the level of utility participation in these programs. To implement the VETIP, each utility should assess the type of information currently being provided to NPRDS and SEE-IN and expand the scope of reporting if appropriate. Additionally, each utility should evaluate current administrative controls for reporting information and for disseminating the results of the NPRDS and SEE-IN Programs to the plant level. These administrative controls may require modification to ensure that effective coordination is established. Concurrent with these efforts, enhancements will be made to both NPRDS and SEE-IN by INPO within its present institutional objectives.

The VETIP has been developed to ensure that nuclear utilities have prompt access to and effective handling of safety-related equipment technical information. In addition, VETIP is responsive to the intent of Generic Letter 83-28, Section 2.2.2. Further details are provided in the body of this report.

FOREWORD

On February 22 and 25, 1983, during start-ups of the Salem Unit 1 plant, both reactor trip breakers (Westinghouse model DB-50) failed to open on an automatic trip signal. As a consequence, the Nuclear Regulatory Commission (NRC) formed an investigating task force to determine the factual information pertinent to the management and administrative controls that should have ensured proper operation of the trip breakers. The findings and conclusions of the task force are documented in NUREG-0977, "NRC Fact Finding Task Force Report on the ATWS Events at the Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983." A second task force determined the extent to which these investigative findings were generic in nature. The NRC subsequently issued NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant" and Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events."

On September 1, 1983, a group of utility representatives met at the offices of the Institute of Nuclear Power Operations (INPO) to discuss the establishment of an ad hoc utility group to address issues relative to the NRC Generic Letter 83-28, Section 2.2.2. The representatives decided that such a group could provide direction that would be of generic benefit to the utilities and consequently formed the Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. The specific charter for the NUTAC (Appendix A) was adopted, and the target date for completion of activities was established as February 1984.

TABLE OF CONTENTS

1. INTRODUCTION

The objective of Generic Letter 83-28, Section 2.2.2 (Appendix D), is to improve the safety and reliability of nuclear power generating stations by ensuring that the utilities are provided with significant and timely technical information concerning reliability of safety-related components. In a typical nuclear station, hundreds of vendors supply the thousands of components that perform safety-related functions. The variations in vintage and design of plants ensure that although common applications of specific components may exist, there are an equal or greater number of unique applications. To attain the objective in a cost-effective and efficient manner, this NUTAC has developed the program outlined in this document. This positive program has been found to be the most realistic approach to attain the objective.

The Vendor Equipment Technical Information Program (VETIP) described in this document establishes a more formal interaction among the major organizations involved with commercial nuclear power generation. The goal of the interaction is to improve the quality and availability of equipment technical information for use by the utilities. The major components of the VETIP are an information transfer system and a centralized evaluation of industry experiences.

This document provides the unanimous NUTAC position on the guidelines for an effective technical information program. The determination of each individual utility to support and utilize these guidelines is the key to the effectiveness of this program for the industry as a whole. The program does not require the use of nor prescribe standard administrative procedures, but it allows the use of plant-specific procedures compatible with the utility's internal organization and needs. However, the recommendations in this document provide the basis for a uniform industry response to NRC questions and requirements relative to a technical information program. This program will be beneficial to the utilities and, at the same time, it will be responsive to Section 2.2.2 of the NRC Generic Letter 83-28.

2. ACRONYMS AND DEFINITIONS

2.1 Acronyms

A/E	Architect-Engineer
AEOD	Office of the Analysis and Evaluation of Operational Data
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
EPRI	Electric Power Research Institute
ETI	Equipment Technical Information
IEB, IEN	Inspection and Enforcement Bulletins and Notices, issued by the NRC
IEEE	Institute of Electrical and Electronics Engineering
INPO	Institute of Nuclear Power Operations
LER	Licensee Event Report, issued by a utility
MOR	Monthly Operating Report
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSSS	Nuclear Steam Supply System
NUTAC	Nuclear Utility Task Action Committee
O&MR	Operations and Maintenance Reminder
PRA	Probabilistic Risk Assessment
QA	Quality Assurance
SEE-IN	Significant Event Evaluation and Information Network
SER	Significant Event Report
SOER	Significant Operating Experience Report
VETIP	Vendor Equipment Technical Information Program

2.2 Definitions

Component - A component is a mechanical or electrical assembly (including instruments) of interconnected parts that constitutes an identifiable device or piece of equipment. Examples of electrical components include a drawout circuit breaker, a circuit card, instruments, or other subassemblies of a larger device that meet this definition. Examples of mechanical components include valves, piping, pumps and pressure vessels, and associated prime movers and/or operators.

Equipment Technical Information (ETI) - For the purposes of this report, this term includes, as a minimum, the following documentation:

- o vendor-supplied engineering and technical information (drawings, manuals, etc.) and changes thereto

- o equipment qualification data (provided by the equipment vendor or qualification lab)

- o industry-developed information, including utility and NRC-originated information (NPRDS, SER, IEB, IEN, etc.)

NUCLEAR NETWORK™* - An information service provided through INPO. (NUCLEAR NETWORK replaced NUCLEAR NOTEPAD.)

NUREG - These are guidance documents that are issued by the NRC.

Safety-Related - Safety-related structures, systems, and components are those relied upon to remain functional during and following design basis events to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability

*Trademark application by INPO for NUCLEAR NETWORK is pending.

to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10 CFR Part 100.

<u>Vendor</u> - For the purposes of this report, this term is used to identify the manufacturer of the component concerned and/or those who provide the related equipment technical information.

3. VENDOR EQUIPMENT TECHNICAL INFORMATION PROGRAM (VETIP) DESCRIPTION

The VETIP includes interactions among the major organizations involved with commercial nuclear power generation. As illustrated in Figure 1, a utility exchanges safety-related equipment information with vendors, NRC, INPO, and other utilities via reports, bulletins, notices, newsletters, and meetings. The purpose of these information exchanges is to share equipment technical information to improve the safety and reliability of nuclear power generating stations. The NUTAC concluded that the lack of information is not a problem, but that the various information systems available are not integrated properly. The purpose of VETIP is to ensure that current information and data will be available to those personnel responsible for developing and maintaining plant instructions and procedures. These information systems and programs currently exist and are capable of identifying to the industry precursors that could lead to a Salem-type event. VETIP is an industry-controlled and mainly hardware-oriented program that does not rely on vendor action, other than the NSSS supplier, to provide information to utilities. Instead, VETIP provides information developed by industry experience through SERs and SOERs to the vendor for comment before it is circulated to the utilities concerned.

The majority of information provided by vendors is commercial in nature. This usually is provided voluntarily by the vendor, but does little to improve the safety or reliability of existing equipment.

A vendor-oriented program to provide information that would improve the safety and reliability of existing equipment relies on the vendor having an internal program to develop the information. Such programs typically are not in existence. Following design and qualification testing, vendors normally do not continue extensive testing or engineering programs in anticipation of equipment problems. Subsequent failures discovered during operations require several steps to complete the information feedback loop. For example, when a problem occurs and a local vendor representative provides a solution, he would have to provide that information to the vendor headquarters. Then, the headquarters would need a tracking program to identify a trend and subsequently a program to provide the information to the industry. In addition, the vendor often is not in the

best position to analyze the failure. The vendor is not always aware of the component's application and environment nor its maintenance and surveillance history.

The VETIP recognizes that the utility user is in a unique position. The utility user alone has immediate access to the maintenance and surveillance history of the equipment. The utility, not the manufacturer, knows the component's actual application and environment. The utility is the primary source of information on the failure, and the utility has the greatest need for the solution. As such, the utility is the central organizer in any approach to the solution, whether or not the manufacturer gets involved. The utility is in the position to know of the failure analysis and its solution at the earliest possible time. The utility can then disseminate the information to other utilities, with an indication of its significance and urgency.

By sharing the operating history, problems, and solutions within the nuclear industry, independent of any normal vendor contacts, the other users will be informed in a much more timely and uniform way. In this way, the distribution of information is controlled entirely by the nuclear utility industry. The programs that comprise the VETIP currently are in existence. The recommended enhancements contained within this report are suggested ways to improve the current use and application of these existing programs.

3.1 Existing Programs

The existing systems and programs included in the VETIP are the Nuclear Plant Reliability Data System (NPRDS) and the Significant Event Evaluation and Information Network (SEE-IN), both managed by INPO. Also, the VETIP includes existing programs that the utilities now conduct with vendors and other sources of ETI, particularly the NSSS vendor interaction programs and the NRC reporting programs that disseminate significant failure information. Utility-vendor interaction is further enhanced by the INPO supplier participant practices. Through participation in this program, NSSS vendors and A/E firms are working toward greater participation in the NPRDS and SEE-IN Programs.

3.1.1 Nuclear Plant Reliability Data System (NPRDS)

NPRDS is an industrywide system managed by INPO for monitoring the performance of selected systems and components at nuclear power plants. INPO member utilities have agreed to participate in the program. United States plants in commercial operation (except for six atypical, early vintage units) supply basic engineering information and subsequent failure data on the selected systems and components (typically six to seven thousand components from some 30 systems per unit). The value of NPRDS lies in the ready availability of this data base to operation and engineering groups for a broad range of applications. The criteria used to determine the scope of NPRDS reports are as follows:

- systems and components that provide functions necessary for accident mitigation
- systems and components for which loss of function can initiate a significant plant transient

Uniform scoping and reporting criteria are set forth in the Nuclear Plant Reliability Data System (NPRDS) Reportable System and Component Scope Manual (INPO 83-020) and in the Reporting Procedures Manual for the Nuclear Plant Reliability Data System (INPO 84-011).

To support the benefits that can be obtained from NPRDS usage, utilities submit three kinds of information to the NPRDS data base: engineering/test information, failure reports, and operating history. The engineering/test record on a component contains information necessary to identify the component and its application, such as manufacturer, model number, operating environment, size, horsepower, and test frequencies. The information is submitted when the component is placed in service and is stored in the data base. If that component fails to perform as intended, a report is submitted containing a description of the failure mode and cause, the failure's effect on plant operations, corrective actions taken, and other

information necessary to assess the failure. On a quarterly basis, utilities submit information on the number of hours the plant is in different modes of operation. This information is used in conjunction with the engineering and failure reports to generate failure statistics for systems and components.

The data is retrievable from a computer, and the engineering and failure information can be combined in various ways. A search of the failure records can identify problems experienced with components in other plants and the corrective actions taken. There are several hundred searches of the data base in a typical month. Following are some example uses of the data base:

Utility and Plant Staffs

- o accessing comprehensive equipment history files to support maintenance planning and repair
- o avoidance of forced or prolonged outages by identifying other plants with similar or identical equipment that may have spares for a possible loan
- o determination of spare parts stocking, based on industry mean time between failures
- o comparison of component failure rates at a given plant with the industry average failure rates

Design Groups

- o identification of common failure modes and causes
- o selection of vendors based on component application and performance
- o identification of component wearout and aging patterns
- o studies of component performance as a function of operating characteristics, such as test frequency and operating environment
- o input to plant availability improvement programs

Operating Experience Reviewers

- o identification of significant failure modes affecting safety or availability
- o trending of component failure rates
- o development of failure probability estimates for use in fault-tree analyses (reliability or PRA studies)

NPRDS data is available to users through various periodic reports and through on-line access of the data from a computer terminal.

3.1.2 Significant Event Evaluation and Information Network (SEE-IN)

Since the early days of nuclear power plant operations, utilities and manufacturers have attempted to share what has been learned from plant operating experience. As nuclear technology becomes more complex and more demanding, the need for sharing operating experience continues to grow and becomes more important. The safety benefits of avoiding problems already encountered and resolved more than justifies the costs and extra effort required for utilities to keep each other informed. The Nuclear Safety Analysis Center (NSAC), with the support of its utility advisory group, began developing a program to share information learned from analyzing nuclear plant experiences. Shortly after its formation in late 1979, the Institute of Nuclear Power Operations (INPO) joined NSAC in the development and implementation of the program. The program has been named "Significant Event Evaluation and Information Network" (SEE-IN). In 1981, the management of the SEE-IN Program became the sole responsibility of INPO.

Objective

The objective of SEE-IN is to ensure that the cumulative learning process from operating and maintenance experience is effective and that the lessons learned are reported and corrective action taken in a timely manner to improve plant safety, reliability, and availability. This objective is met

by screening available nuclear plant event information systematically, identifying and evaluating the important or significant events, and communicating the results to the utilities and appropriate designers and manufacturers.

Scope

The functional approach to SEE-IN is an eight-step process outlined in Appendix C. While INPO has the program management function, no single organization is responsible for performing all of these functions; rather, the responsibility is spread among key participants in the network. The principle organizations involved in the initial screening of plant event data are the utilities and INPO. Each nuclear utility has an in-house program to screen events that occur in its nuclear plant(s). INPO has a broader charter to screen all nuclear plant events. The sources of input to the screening process include NPRDS, NUCLEAR NETWORK, NRC-mandated reports, IEBs, IENs, etc. The provision to control the data normally is governed by agreements between INPO and the supplying organization (e.g., utilities, NRC, NSSS vendors, international participants, etc.). When a significant event or trend has been identified from the screening process, a Significant Event Report (SER) is prepared by INPO and transmitted to the utilities and other participants on NUCLEAR NETWORK. This event then undergoes an action analysis by INPO. The purpose of the action analysis is to investigate the event or trend in more detail and to develop and evaluate practical remedies. For events requiring utility action, the results of the action analysis are communicated to the utilities, normally in the form of a Significant Operating Experience Report (SOER). In these instances, recommendations are made to resolve the underlying problems. The implementation of applicable recommended remedial actions is the responsibility of the individual utility. Implementation may include changes in plant procedures, equipment design, and/or operator training programs. The two final steps in the SEE-IN process are (1) feedback and INPO

assessment during plant evaluation of actions taken by the utilities as a result of information provided through SEE-IN and (2) periodic assessment of the process effectiveness by INPO.

For events which, through the screening process, are determined not significant but have valuable operations and maintenance information, an Operations & Maintenance Reminder (O&MR) is prepared and processed in the same way as SERs.

The SEE-IN Program provides copies of draft SERs, O&MRs, and SOERs to the affected vendors for review. Vendor comments are considered in preparation of final SEE-IN reports. Once finalized, the reports are sent to the utilities.

The SEE-IN Program includes a cross-reference capability to identify SERs, O&MRs, SOERs, LERs, etc., which report component problems that could cause a significant event. This cross-reference facilitates utility review of the component's prior history before using that component in a safety-related application.

Program Operation

Plant operating experience data is reviewed from several perspectives including design, component and system performance, plant procedures, human factors, personnel training, maintenance and testing practices, and management systems to identify significant events and trends.

Formal Review Sources

A formal review is conducted on NRC information notices, bulletins, AEOD reports, event-related generic letters, etc. A formal review also is conducted on industry-prepared information (including those required by NRC) such as LERs, monthly operating reports, NRC event-related reports, NSSS technical bulletins, NPRDS data, NUCLEAR NETWORK operating experience

entries, international operating experience reports, construction deficiency reports, safety defect reports, and trends identified as significant in the INPO NPRDS and LER data bases. The formal review includes a dual, independent screening process. The review status is documented and tracked by computer.

Other sources of operating experience information are used by the SEE-IN Program on an ad hoc basis as reference or supplemental material but do not receive a formal review. The sources include such items as NRC NUREG documents, EPRI and NSAC reports, and other industry reports or data concerned with plant operating experience. The INPO process for screening is shown in Figure 2.

Utility Contact (SEE-IN)

In addition to the formal and reference information sources, another vital information source is direct contact with power plant technical personnel on an ad hoc basis. Each utility designates a SEE-IN contact to respond to questions from INPO on plant events. The majority of such communications was handled over the telephone or via NUCLEAR NETWORK. Files are maintained by INPO on nuclear utilities and contain names and telephone numbers of designated contacts, telecopier numbers, status of nuclear units (i.e., operating, under construction or planned), and NSSS vendor(s).

3.1.3 Interaction With Vendors

In the interest of operating the plant safely and efficiently, the utility-vendor contact is essential. To accomplish this goal, utilities already interact with various vendors.

The contractual obligations for furnishing equipment and software (manuals, drawings, etc.) are fulfilled upon acceptance at the plant site. Interaction between utilities and vendors, due to deficiencies, may be brought about by the

reporting requirements of 10 CFR 21 and 10 CFR 50.55(e). The continuing contract with vendors for warranty obligations or maintenance work are two examples of active interaction after an initial purchase. In addition, much of the interaction with the vendors during plant life is initiated in response to significant failures, to failure trends experienced at the plant, to spare parts procurement, or to subsequent purchase orders of new equipment.

The interaction with the NSSS vendor, who typically supplies a large portion of the safety-related plant equipment, generally is more active than with the other vendors. There are existing channels through which the NSSS suppliers disseminate information of interest to their client utilities. These include the following:

- In regular meetings, NSSS representatives outline recent developments and maintenance/design recommendations. Any special concerns of the utility can be addressed in follow-up correspondence with the NSSS supplier's service department.

- Bulletins or advisories from the NSSS supplier's service department alert client utilities to special problems experienced by similar plants. Typically included in this correspondence are a description of the problem and the corrective actions taken to resolve it. Recommendations for preventive actions or for particular cautions to be considered by the utility usually are included.

- Owners groups provide an additional forum for the exchange of information that may be of generic interest to member utilities. For example, problems in the design or operation of a system or component may be shared with the group and potential resolutions identified. The owners groups' efforts often are directed at seeking improvements or anticipating problems rather than being only reactive in nature.

Improvements in availability or testing and maintenance procedures are examples of positive results that have come about through owners groups activities. The NSSS supplier makes his broadly-based knowledge available to the group for the specialized evaluations that may be required.

3.1.4 Regulatory Reporting Requirements

Other existing sources of information are the documents that result from the NRC's reporting requirements. These documents include 10 CFR 21 reports, 10 CFR 50.55(e) reports, Licensee Event Reports, and NRC Inspection & Enforcement (IE) Bulletins and Information Notices. 10 CFR 21 specifies reporting requirements relating to component or system deficiencies that may create a substantial safety hazard. This reporting provides the nuclear utility industry notification of significant noncompliances and defects identified by other utilities, architect-engineers, constructors, vendors, and manufacturers associated with nuclear facilities.

10 CFR 50.55(e) requires that the holder of a construction permit notify the NRC of each deficiency found in design and construction, which, if uncorrected, could affect the safe operation of the nuclear power plant adversely.

10 CFR 50.73 requires the holder of an operating license for a nuclear power plant to submit a Licensee Event Report (LER) for events described in 50.73(a)(2). These LERs are incorporated into the INPO LER data base, which provides information to identify and isolate precursor events and identify emerging trends or patterns of potential safety significance.

The NRC Office of Inspection and Enforcement (IE) issues various documents, including bulletins and information notices, to inform licensees and construction permit holders of significant concerns that may result from the NRC evaluation of reports, as required by 10 CFR 21.21, 50.55(e), and

50.73. These documents provide the nuclear utilities with information on events and concerns that are considered significant by the NRC.

3.2 Recommended Enhancements to Existing Programs

The following are recommended enhancements to the existing programs. INPO and the NPRDS User's Group should investigate the feasibility of these recommendations. If found feasible, an implementation program should be developed.

3.2.1 Enhancements to NPRDS

- The present definition of component in NPRDS (extracted from IEEE 603-1980) is more applicable to electrical components. The definition should be improved to describe mechanical components better.

- The present failure reporting guidance needs improvement in the following areas:
  - Guidance is needed to provide better information for analyzing the role of piece parts as a factor in causing component failures.

  - The guidance should be revised to indicate that utilities should supply information when inadequate vendor information is identified as a causal or contributing factor in a failure. The guidance should provide users of the data base the ability to retrieve readily those failures involving inadequate vendor information (example, key word sorting, coding).

  - Present failure reports are often sketchy in providing details of the failure analysis conducted by utilities. The guidance should emphasize the importance of providing more complete results of failure analysis when one is conducted. Although detailed failure analyses are not

always conducted for every failure, when they are conducted they should be provided in NPRDS failure reports. In this way, the SEE-IN Program and other utilities can derive more benefit from the work of each utility.

- Utilities should develop internal methods to ensure that their NPRDS reports are clear and complete and that the program guidance is followed appropriately.

- For some failures it may not be possible for utilities to provide a complete failure description within the time frames for reporting to NPRDS. Utilities should still submit preliminary failure reports within the established time frame. Utilities should revise these reports when the necessary information is available. However, the present system does not provide methods for utilities to indicate that reports will be revised later. NPRDS should be modified to permit each utility to readily identify which of their reports still requires follow-up information. Utilities should report a failure event promptly and include an initial analysis. Detailed and complete information should be provided in a timely manner once final analysis has been completed.

- The present scope of NPRDS reporting may not meet all the needs of individual utilities for monitoring the reliability of their own safety-related components. Each utility that decides that additional systems and components should be added to their basic scope of NPRDS systems and components should request that INPO accept these systems. INPO will consider these requests, identify the additional resource requirements needed to handle these requests, and notify utilities when it is able to accept additional information.

3.2.2 Enhancements to SEE-IN

- Reports should be generated for potential failures caused by faulty or missing vendor-supplied information or other ETI. The VETIP recognizes that the utility will uncover errors in ETI (e.g., during review of the information, writing of instructions, testing, etc.) before anyone else. It is recommended that ETI faults be reported over NUCLEAR NETWORK for review by INPO under the SEE-IN Program.

- The SEE-IN Program should be broadened by INPO to improve the ability to trend NPRDS data. Present methods of trending are largely qualitative and subjective in nature. They depend largely on the ability of analysts to recognize the need to look for degrading or unacceptable system and component reliability. INPO should develop methods to use NPRDS in a more quantitative fashion to detect trend problems. This enhancement is presently under development by INPO.

3.3 Summary Example

One problem that led to the Salem event was that the information contained in the NSSS vendor technical bulletin (issued in 1974) was not processed appropriately and therefore not incorporated into plant procedures. If the systems that comprise the VETIP were functional in the early 1970s, this oversight probably would not have occurred or would have been rectified. Westinghouse had prepared the technical bulletin based on a precursor event that occurred at another nuclear unit. This type of precursor event would have required that an LER be written and submitted to the NRC. INPO also would have reviewed the Westinghouse technical bulletin and the LER. The current criteria for significance screening used by INPO personnel identify this type event as a significant single failure. It is highly likely that an SER would have been generated by INPO and disseminated to utilities via NUCLEAR NETWORK. Utilities would have reviewed the SER through their operating experience report review programs.

In addition, utilities would have had an ongoing program with their NSSS vendors to obtain ETI. Utilities would have had systems in place to track and process this information. Therefore, there are two pathways that would have ensured this type of information was received and evaluated by the utility:

- o NPRDS/SEE-IN (SERs, SOERs)
- o NSSS vendor technical bulletins

The utility's VETIP procedures would have assessed this information and effected positive action to correct the failed component.

4. IMPLEMENTATION OF VETIP

4.1 Responsibilities for Implementation

4.1.1 Utility Implementation Responsibilities

4.1.1.1 Existing Programs

NSSS Vendor Contact

Each utility should have a program in place with its NSSS supplier to obtain technical information. This program consists of a technical bulletin system and necessary direct contact with the NSSS supplier.

NPRDS/SEE-IN

Each utility should indicate or reaffirm its active participation in the NPRDS and SEE-IN Programs. The utility should supply the necessary basic information and should report failures and problems on a timely basis. Adequate internal controls should be in place to ensure that this activity is timely, consistent, and controlled and should include incorporation of future revisions to these programs.

Other Vendors

Each utility should continue to seek assistance and ETI from other safety-related equipment vendors when the utility's evaluation of an equipment or ETI problem concludes that such direct interaction is necessary or would be beneficial. These problems and those of lesser significance will continue to be reported by means of the NPRDS and/or the SEE-IN Programs.

Internal Handling of Equipment Technical Information

The utility should process incoming ETI so the objectives noted below are achieved.

- o Administrative procedures should provide control of incoming ETI whether it arrives directly from the vendor or from other industry or regulatory sources (i.e., NUCLEAR NETWORK, NPRDS, SEE-IN, NRC bulletins, etc.), so it receives the appropriate engineering/technical review, evaluation, and distribution for the following:
  - -- prompt warnings to key personnel
  - -- timely incorporation into maintenance or operating procedures, equipment data/purchasing records, and training programs
  - -- future procedure review and revision cycles
  - -- notification on NUCLEAR NETWORK of significant ETI

  The incorporation of such safety-related information (or changes) remains within the scope of the utility's review and approval requirements.

- o The administrative program should require that maintenance or operating procedures cite appropriate ETI in the reference section of the procedure.

- o Within the performance section of the procedure, appropriate ETI should be incorporated and approved in the engineering, technical, and quality review of the safety-related procedure.

- Internal Handling of Vendor Services

  The vendor, contractor, or technical representative who will perform safety-related services should be a QA approved/qualified supplier of such nuclear safety-related services. Furthermore, the services should be specified in the procurement documentation so that a combination of procedural and QA/QC controls are established.

  A vendor service may be performed using utility procedures. If so, the procurement documentation should specify that the service is performed using utility procedures that have been approved after a technical and quality review cycle typical for other utility service, maintenance, repair, or operating procedures. As an alternative, the service may be performed using vendor or contractor procedures. In this case, the documentation should specify that the service is performed using vendor, contractor, or technical representative procedures that have been reviewed and approved in accordance with the utility procurement program, QA program, and administrative review program. This is to ensure that their documents are processed and approved in a manner equivalent to the utility procedures concerning similar activities.

  In addition to specifying the procedures that will be used, the QA/QC program to be used should also be specified. The utility QA/QC program may be used. In this case, the procurement documentation should specify that the activity will be performed under the cognizance of the utility QA/QC program. Alternately, the vendor or contractor QA/QC program may be used. In this case, the

documentation should specify that the activity will be performed under the cognizance of the vendor, contractor, or technical representative QA/QC program that has been reviewed separately and approved in accordance with the utility QA program. In addition, during the performance of the service, the utility QA program will monitor the effectiveness of their performance and compliance with its approved program by suitable surveillance, inspection, and audit.

4.1.1.2 Enhanced Programs

- NPRDS

  Each utility should incorporate the enhancements to the NPRDS recommended in Section 3.2. This could involve revisions to existing administrative programs or procedures. It also could require revised training or other actions needed to ensure a meaningful and effective implementation of the NPRDS program enhancements.

- SEE-IN

  Each utility should incorporate the enhancements to the SEE-IN program recommended in Section 3.2. As in the NPRDS program, this could involve revisions to existing administrative programs or procedures or to training or other activities so the data reported to the SEE-IN Program is complete and detailed enough to support the system enhancements being undertaken by INPO.

4.1.2 INPO Implementation Responsibilities

- Existing Programs

  The NUTAC determined that present NPRDS/SEE-IN Programs, properly used, currently provide an adequate framework for the effective exchange of information.

- Enhanced Programs

  INPO should implement the enhancements of the NPRDS and SEE-IN Programs (noted in Section 3.2) to augment this VETIP.

4.2 Schedule for Implementation

4.2.1 Existing Programs

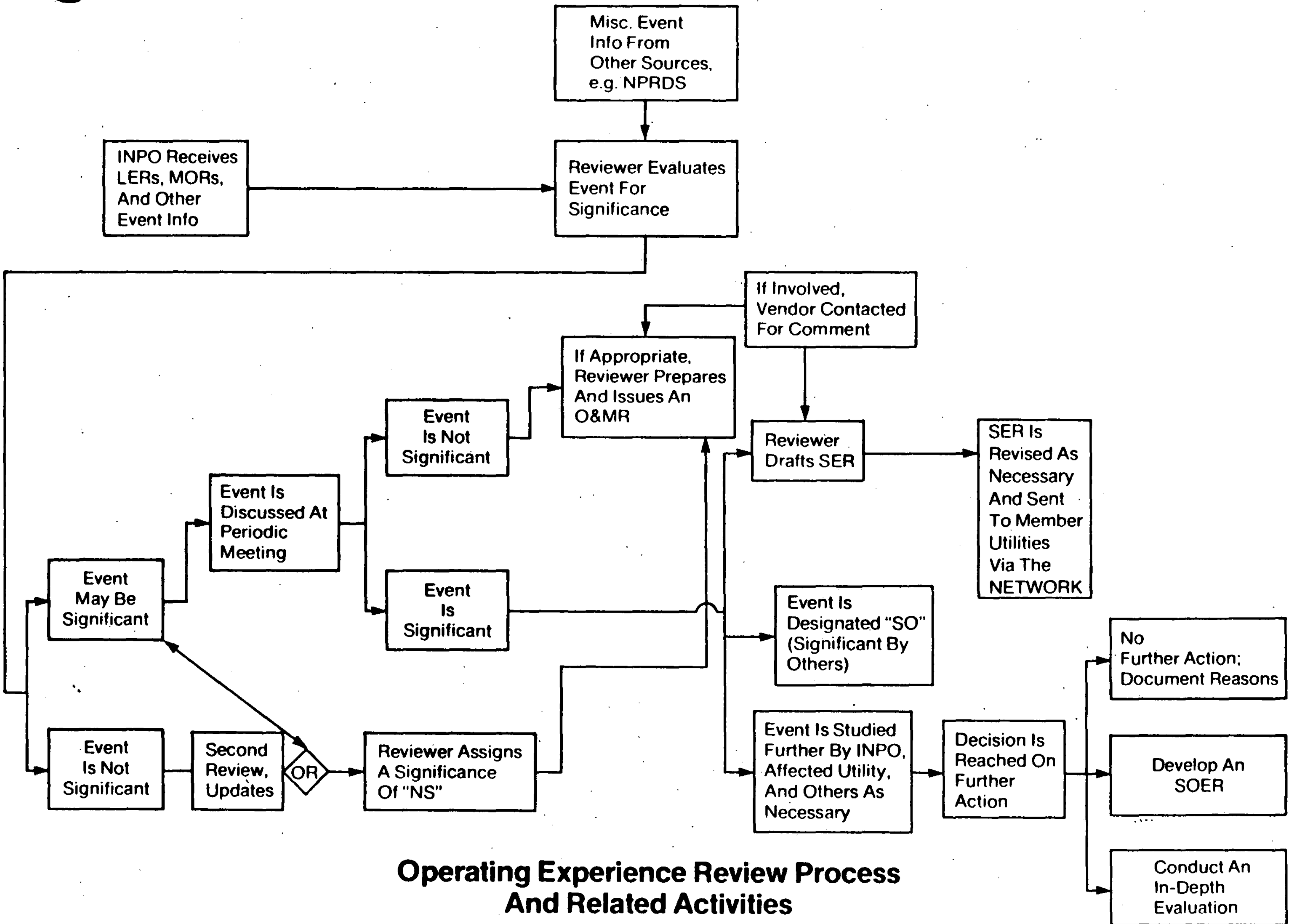
Utilities that find that their existing internal program and procedures do not support those outlined in Sections 3.1 and 4.1.1.A above should make the necessary timely revisions as part of the established review and updating cycle for such documentation. A specific schedule should be established by the individual utility with a target date for full implementation by January 1, 1985.

4.2.2 Enhancements to Existing Programs

4.2.2.1 INPO should work with the NPRDS user's group with the goal of establishing schedules by July 1, 1984, for implementation of the enhancements of the NPRDS program.

4.2.2.2 Utilities should incorporate the enhancements to the NPRDS and SEE-IN programs, recommended in Section 3.2 and 4.1.1.B above into their internal program and procedures on a timely basis.

4.2.2.3 Schedules should be established that are consistent with an overall goal to implement the recommended enhancements to both programs by January 1, 1986.



Operating Experience Review Process And Related Activities

Figure 2

APPENDIX A

SPECIFIC CHARTER FOR
NUCLEAR UTILITY TASK ACTION COMMITTEE
ON GENERIC LETTER 83-28,
SECTION 2.2.2

APPENDIX A

SPECIFIC CHARTER FOR NUCLEAR UTILITY TASK ACTION COMMITTEE ON GENERIC LETTER 83-28, SECTION 2.2.2

This Nuclear Utility Task Action Committee (NUTAC) has been established by a group of utility representatives who have recognized a need for nuclear industry guidance on Generic Letter 83-28, Section 2.2.2. The establishment of this NUTAC has been in accordance with the general charter governing the organization and operation of a NUTAC, as approved by the Institute of Nuclear Power Operations (INPO) Board of Directors. This NUTAC is committed to compliance with this specific charter, its bylaws, and the general charter. This charter has been reviewed and approved by the chairman of the Analysis and Engineering Division Industry Review Group and the president of INPO, and the president of INPO authorizes staff support for this NUTAC.

This committee has adopted the following objective to ensure fulfillment of the goal of achieving industry consensus and guidance on Generic Letter 83-28, Section 2.2.2:

- o development of guidance for use by utilities in response to Generic Letter 83-28, Section 2.2.2

To ensure that this objective results in products that are of generic benefit to the utilities, voting membership on this committee is limited to permanent employees of U.S. nuclear utilities. The chairman and vice chairman of this committee will be elected by the NUTAC from a list of candidates approved by the chairman of the sponsoring IRG. To further ensure that this NUTAC provides products that are of generic benefit to utilities, the NUTAC chairman will maintain close liaison with the sponsoring INPO Industry Review Group.

Additionally, this NUTAC should establish liaison with other recognized industry groups, such as AIF, ANS, EEI, EPRI, and NSSS owners groups and will maintain communication on this industry initiative with the NRC, as appropriate.

Approved: Edward P. Griffing	9/1/83	R B McDonald	9/21/83
Chairman, NUTAC	Date	Chairman, IRG	Date
Walter E. Andrews	9/1/83	E P Wilkinson	9/21/83
Vice Chairman, NUTAC	Date	President, INPO	Date

APPENDIX B

LIST OF REFERENCES

APPENDIX C

SEE-IN FUNCTIONS

SEE-IN Functions

1. Provide basic report of plant event (utilities).

2. Screen events for significance and transmit Significant Event Reports (SERs) via NUCLEAR NETWORK (utilities and INPO with vendor input solicited when specific product is identified).

3. Provide backup data on contributing factors and probable causes and consequences (utilities and vendors).

4. Perform action analysis on significant events to evaluate possible options for short-term remedies and feasible long-term solutions that might be implemented (utilities, INPO, and vendors).

5. Disseminate information, along with an alert of potential implication, to the utilities (INPO).

6. Evaluate the information and implement remedies as appropriate (utilities).

7. Provide feedback on implementation actions (utilities and INPO).

8. Evaluate periodically the effectiveness of the process, including steps 1-7 above (INPO).

APPENDIX D

GENERIC LETTER 83-28
SECTION 2.2.2

(Generic Letter 83-28, Section 2.2.2
is enclosed verbatim)

The enclosure to this letter breaks down these actions into several components. You will find that all actions, except four (Action 1.2, 4.1, 4.3, and 4.5), require software (procedures, training, etc.) changes and/or modifications and do not affect equipment changes or require reactor shutdown to complete. Action 1.2 may result in some changes to the sequence of events recorder or existing plant computers, but will not result in a plant shutdown to implement. Actions 4.1, 4.3 and 4.5.2, if applicable, would require the plant to be shutdown in order to implement.

The reactor trip system is fundamental to reactor safety for all nuclear power plant designs. All transient and accident analyses are predicated on its successful operation to assure acceptable consequences. Therefore, the actions listed below, which relate directly to the reactor trip system, are of the highest priority and should be integrated into existing plant schedules first.

1.1 Post-Trip Review (Program Description and Procedure)

2.1 Equipment Classification and Vendor Interface (Reactor Trip System Components)

3.1 Post-Maintenance Testing (Reactor Trip System Components)

4.1 Reactor Trip System Reliability (Vendor-Related Modifications)

4.2.1 and 4.2.2 Reactor Trip System Reliability (Preventive Maintenance and Surveillance Program for Reactor Trip Breakers)

4.3 Reactor Trip System Reliability (Automatic Actuation of Shunt-trip Attachment for Westinghouse and B&W plants)

Most of the remaining intermediate-term actions concern all other safety-related systems. These systems, while not sharing the same relative importance to safety as the reactor trip system, are essential in mitigating the consequences of transients and accidents. Therefore, these actions should be integrated into existing plant schedules over the longer-term on a medium priority basis. Some of the actions discussed in the enclosure will best be served by Owners' Group participation, and this is encouraged to the extent practical.

Accordingly, pursuant to 10 CFR 50.54(f), operating reactor licensees and applicants for an operating license (this letter is for information only for those utilities that have not applied for an operating license) are requested to furnish, under oath and affirmation, no later than 120 days from the date of this letter, the status of current conformance with the positions contained herein, and plans and schedules for any needed improvements for conformance with the positions. The schedule for the implementation of these improvements is to be negotiated with the Project Manager.

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

July 8, 1983

TO ALL LICENSEES OF OPERATING REACTORS, APPLICANTS FOR OPERATING LICENSE, AND HOLDERS OF CONSTRUCTION PERMITS

Gentlemen:

SUBJECT: REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS (Generic Letter 83-28)

The Commission has recently reviewed intermediate-term actions to be taken by licensees and applicants as a result of the Salem anticipated transient without scram (ATWS) events. These actions have been developed by the staff based on information contained in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." These actions address issues related to reactor trip system reliability and general management capability.

The actions covered by this letter fall into the following four areas:

1. Post-Trip Review - This action addresses the program, procedures and data collection capability to assure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

2. Equipment Classification and Vendor Interface - This action addresses the programs for assuring that all components necessary for accomplishing required safety-related functions are properly identified in documents, procedures, and information handling systems that are used to control safety-related plant activities. In addition, this action addresses the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

3. Post-Maintenance Testing - This action addresses post-maintenance operability testing of safety-related components.

4. Reactor Trip System Reliablity Improvements - This action is aimed at assuring that vendor-recommended reactor trip breaker modifications and associated reactor protection system changes are completed in PWRs, that a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in PWRs, that the shunt trip attachment activates automatically in all PWRs that use circuit breakers in their reactor trip system, and to ensure that on-line functional testing of the reactor trip system is performed on all LWRs.

Licensees and applicants may request an extension of time for submittals of the required information. Such a request must set forth a proposed schedule and justification for the delay. Such a request shall be directed to the Director, Division of Licensing, NRR. Any such request must be submitted no later than 60 days from the date of this letter. If a licensee or applicant does not intend to implement any of the enclosed items, the response should so indicate and a safety basis should be provided for each item not intended to be implemented. Value-impact analysis can be used to support such responses or to argue in favor of alternative positions that licensees might propose.

For Operating Reactors, the schedules for implementation of these actions shall be developed consistent with the staff's goal of integrating new requirements, considering the unique status of each plant and the relative safety importance of the improvements, combined with all other existing plant programs. Therefore, schedules for implementation of these actions will be negotiated between the NRC Project Manager and licensees.

For plants undergoing operating license review at this time, plant-specific schedules for the implementation of these requirements shall be developed in a manner similar to that being used for operating reactors, taking into consideration the degree of completion of the power plant. For construction permit holders not under OL review and for construction permit applicants, the requirements of this letter shall be implemented prior to the issuance of an operating license.

This request for information was approved by the Office of Management and Budget under clearance number 3150-0011 which expires April 30, 1985. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington, D. C. 20503.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Required Actions Based on Generic
Implications of Salem ATWS Events

ENCLOSURE

REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

Position

Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely. A report describing the program for review and analysis of such unscheduled reactor shutdowns should include, as a minimum:

1. The criteria for determining the acceptability of restart.

2. The responsibilities and authorities of personnel who will perform the review and analysis of these events.

3. The necessary qualifications and training for the responsible personnel.

4. The sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to reconstruct the event accurately and in sufficient detail for proper understanding. (See Action 1.2)

5. The methods and criteria for comparing the event information with known or expected plant behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function).

6. The criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Plant Operations Review Committee, will be consulted prior to authorizing restart) and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

7. Items 1 through 6 above are considered to be the basis for the establishment of a systematic method to assess unscheduled reactor shutdowns. The systematic safety assessment procedures compiled from the above items, which are to be used in conducting the evaluation, should be in the report.

Applicability

This position applies to all licensees and OL applicants.

Type of Review

For licensees, a post-implementation review of the program and procedures will be conducted or the staff will perform a pre-implementation review if desired by the licensee. NRR will perform the review and issue Safety Evaluations.

For OL applicants, the NRR review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants shall submit a report describing their program addressing all the items in the position.

Techncial Specification Changes Required

No changes to Technical Specifications are required.

References

Section 2.2 of NUREG-1000
Regulatory Guide 1.33
ANSI N18.7-1976/ANS-3.2
Item I.C.5 of NUREG-0660
10 CFR 50 - 50.72

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

Position

Licensees and applicants shall have or have planned a capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment.

Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Action 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

A report shall be prepared which describes and justifies the adequacy of equipment for diagnosing an unscheduled reactor shutdown. The report shall describe as a minimum:

1. Capability for assessing sequence of events (on-off indications)

    1. Brief description of equipment (e.g., plant computer, dedicated computer, strip chart)

    2. Parameters monitored

    3. Time discrimination between events

    4. Format for displaying data and information

    5. Capability for retention of data and information

    6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable)

2. Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns, and the functioning of safety-related equipment.

    1. Brief description of equipment (e.g., plant computer, dedicated computer, strip charts)

    2. Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate

    3. Duration of time history (minutes before trip and minutes after trip)

4. Format for displaying data including scale (readability) of time histories

5. Capability for retention of data, information, and physical evidence (both hardware and software)

6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable)

3. Other data and information provided to assess the cause of unscheduled reactor shutdowns.

4. Schedule for any planned changes to existing data and information capability.

Applicability

This position applies to all licensees and OL applicants.

Type of Review

Data and information capability will be reviewed by NRR to determine whether adequate data and information will be available to support the systematic safety assessment of unscheduled reactor shutdowns. NRR will perform the reviews and issue a Safety Evaluation.

For licensees, a post-implementation review of the program and procedures will be conducted by NRR or the staff will perform a pre-implementation review if desired by the licensee.

For OL applicants, the NRR review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants shall submit a report describing their data and information capability for unscheduled reactor shutdowns.

Technical Specification Changes Required

To be determined based on evaluation of required documentation.

References

Section 2.2 of NUREG-1000.

EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)

Position

Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement. In addition, for these components, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of these components should be contacted and an interface established. Where vendors can not be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and nonnuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.

Applicability

This action applies to all licensees and OL applicants.

Type of Review

For licensees, a post-implementation review will be conducted. NRR will perform these licensing reviews and issue a Safety Evaluation.

For OL applicants, the NRR review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants should submit a statement confirming that they have reviewed the Reactor Trip System components and conform to the position regarding equipment classification. In addition, a summary report describing the vendor interface program shall be submitted for staff review. Vendor lists of technical information, and the techncial information itself, shall be available for inspection at each reactor site.

Technical Specification Changes Required

No changes to Technical Specifications are required.

Reference

Section 2.3.1 of NUREG-1000.
Section 2.3.2 of NUREG-1000.

6. Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems, and components important to safety required by GDC-1 (defined in 10 CFR Part 50, Appendix A, "General Design Criteria, Introduction").

2. For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgment for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Applicability

This action applies to all licensees and OL applicants.

Type of Review

For licensees, a post-implementation review will be conducted. NRR will perform the review and issue a Safety Evaluation.

For OL applicants, the NRR review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants should submit a report that describes the equipment classification and vendor interface programs outlined the position above.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)

Position

Licensees and applicants shall submit, for staff review, a description of their programs for safety-related* equipment classification and vendor interface as described below:

1. For equipment classification, licensees and applicants shall describe their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and replacement parts. This description shall include:

    1. The criteria for identifying components as safety-related within systems currently classified as safety-related. This shall not be interpreted to require changes in safety classification at the systems level.

    2. A description of the information handling system used to identify safety-related components (e.g., computerized equipment list) and the methods used for its development and validation.

    3. A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B, apply to safety-related components.

    4. A description of the management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed.

    5. A demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions and provide support for the licensees' receipt of testing documentation to support the limits of life recommended by the supplier.

*Safety-related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure: (1) the integrity of the reactor coolant boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100.

Technical Specification Changes Required

No changes to the Technical Specifications are required.

References

Section 2.3.1 of NUREG-1000.
Section 2.3.2 of NUREG-1000.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

Position

The following actions are applicable to post-maintenance testing:

1. Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval. (Note that action 4.5 discusses on-line system functional testing.)

Applicability

This action applies to all licensees and OL applicants.

Type of Review

For licensees, a post-implementation review will be conducted for actions 3.1.1 and 3.1.2 above. The Regions will perform these licensing reviews and issue Safety Evaluations. Proposed Technical Specification changes resulting from action 3.1.3 above will receive a pre-implementation review by NRR.

For OL applicants, the review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants should submit a statement confirming that actions 3.1.1 and 3.1.2 of the above position have been implemented.

Technical Specification Changes Required

Changes to Technical Specifications, as a result of action 3.1.3, are to be determined by the licensee or applicant and submitted for staff approval, as necessary.

Reference

Section 2.3.4 of NUREG-1000.

3.2 POST-MAINTENANCE TESTING (ALL OTHER SAFETY-RELATED COMPONENTS)

Position

The following actions are applicable to post-maintenance testing:

1. Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Applicability

This action applies to all licensees and OL applicants.

Type of Review

For licensees, a post-implementation review will be conducted for actions 3.2.1 and 3.2.2 above. The Regions will perform these licensing reviews and issue Safety Evaluations. Proposed Technical Specification changes resulting from action 3.2.3 above will receive a pre-implementation review by NRR.

For OL applicants, the review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants should submit a statement confirming that actions 3.2.1 and 3.2.2 of the above position have been implemented.

Technical Specification Changes Required

Changes to Technical Specifications, as a result of action 3.2.3, are to be determined by the licensee or applicant for staff approval, as necessary.

Reference

Section 2.3.4 of NUREG-1000.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

Position

All vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either: (1) each modification has, in fact, been implemented; or (2) a written evaluation of the technical reasons for not implementing a modification exists.

For example, the modifications recommended by Westinghouse in NCD-Elec-18 for the DB-50 breakers and a March 31, 1983, letter for the DS-416 breakers shall be implemented or a justification for not implementing shall be made available. Modifications not previously made shall be incorporated or a written evaluation shall be provided.

Applicability

This action applies to all PWR licensees and OL applicants.

Type of Review

For licensees, a post-implementation review will be conducted. The Regions will perform these licensing reviews and issue Safety Evaluations.

For OL applicants, the NRR review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants should submit a statement confirming that this action has been implemented.

Technical Specifications Required

No changes to Technical Specifications are required.

Reference

Section 3 of NUREG-1000.

4.2 REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)

Position

Licensees and applicants shall describe their preventative maintenance and surveillance program to ensure reliable reactor trip breaker operation. The program shall include the following:

1. A planned program of periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier.

2. Trending of parameters affecting operation and measured during testing to forecast degradation of operability.

3. Life testing of the breakers (including the trip attachments) on an acceptable sample size.

4. Periodic replacement of breakers or components consistent with demonstrated life cycles.

Applicability

This action applies to all PWR licensees and OL applicants.

Type of Review

Actions 4.2.1 and 4.2.2 will receive a post-implementation review by NRR. A pre-implementation review will be performed by NRR for actions 4.2.3 and 4.2.4 (the circuit breaker life testing program and the component testing/replacement requirements based upon the life testing results). A Safety Evaluation will be issued.

For OL applicants, NRR will perform the reviews for actions 4.2.1 and 4.2.2 on a schedule consistent with the licensing schedule. NRR will perform a pre-implementation review for actions 4.2.3 and 4.2.4 (the circuit breaker life testing program and the component testing/replacement requirements based upon the life testing results). Safety Evaluations will be issued.

Documentation Required

Licensees and applicants should submit descriptions of their programs to ensure compliance with this action.

Technical Specification Changes Required

No changes to Technical Specifications are required.

Reference

Section 3 of NUREG-1000.

4.3 REACTOR TRIP SYSTEM RELIABILITY (AUTOMATIC ACTUATION OF SHUNT TRIP ATTACHMENT FOR WESTINGHOUSE AND B&W PLANTS)

Position

Westinghouse and B&W reactors shall be modified by providing automatic reactor trip system actuation of the breaker shunt trip attachments. The shunt trip attachment shall be considered safety related (Class IE).

Applicability

This action applies to all Westinghouse and B&W licensees and OL applicants.

Type of Review

For licensees, a pre-implementation review shall be performed for the design modifications by NRR. A Safety Evaluation will be issued.

For OL applicants, the NRR review will be performed consistent with the licensing schedule.

Technical Specification changes, if required, will be reviewed prior to implementation.

Documentation Required

Licensees and applicants should submit a report describing the modifications.

Technical Specification Changes Required

Licensees are to submit any needed Technical Specification change requests prior to declaring the modified system operable.

Reference

Section 3 of NUREG-1000.

4.4 REACTOR TRIP SYSTEM RELIABILITY (IMPROVEMENTS IN MAINTENANCE AND TEST PROCEDURES FOR B&W PLANTS)

Position

Licensees and applicants with B&W reactors shall apply safety-related maintenance and test procedures to the diverse reactor trip feature provided by interrupting power to control rods through the silicon controlled rectifiers.

This action shall not be interpreted to require hardware changes or additional environmental or seismic qualification of these components.

Applicability

This action applies to B&W licensees and OL applicants only.

Type of Review

For licensees, a post-implementation review will be conducted. The Regions will conduct the licensing review and issue a Safety Evaluation.

For OL applicants, the review will be performed consistent with the licensing schedule.

Documentation Required

Licensees and applicants should submit a statement confirming that this action has been implemented.

Technical Specification Changes Required

Include the silicon controlled rectifers in the appropriate surveillance and test sections of the Technical Specifications.

Reference

Section 3 of NUREG-1000.

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

Position

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants.

1. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W (see Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

2. Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.

3. Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

    1. uncertainties in component failure rates
    2. uncertainty in common mode failure rates
    3. reduced redundancy during testing
    4. operator errors during testing
    5. component "wear-out" caused by the testing

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by licensees currently not performing on-line testing shall be justified by information on the sensitivity of reactor trip system availability to parameters such as the test intervals, component failure rates, and common mode failure rates.

Applicablity

This action applies to all licensees and OL applicants.

Type of Review

For licensees, a post-implementation review will be conducted for action 4.5.1. The Regions will perform these licensing reviews and issue Safety Evaluations. Actions 4.5.2 and 4.5.3 will require a pre-implementation review by NRR. Results will be issued in a Safety Evaluation.

Printed in U.S.A.