DOCKETED
USNRC

2001 DEC 21 PM 1:19

OFFICE OF THE SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

DOCKET NUMBER
PROPOSED RULE PR 50
(66FR59546)

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NEI
NUCLEAR ENERGY INSTITUTE

**Anthony R. Pietrangelo**
DIRECTOR, RISK AND
PERFORMANCE BASED REGULATION
NUCLEAR GENERATION

December 13, 2001

Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attention: Rulemakings and Adjudications Staff
Mail Stop O-16C1

Subject: NEI[1] comments on the draft §50.69 *(ref. 66 Fed Reg. 59546, dated November 29, 2001)*

The NRC recently released draft rule language for §50.69 following a public workshop on draft rule concepts that were previously released to the public. We commend the NRC for these actions to increase stakeholder interaction. These actions will contribute to a more efficient rulemaking process and ultimately to a clear and effective rule.

To this end, this letter provides the enclosed comments on the draft §50.69 language for the NRC's consideration. In addition, we would like to request a meeting with the NRC's Risk-Informed Licensing Panel in early January 2002 to discuss the draft rule and our comments. We are sensitive to the fact that the proposed rule package is due to the Commission in April 2002 and want to ensure that our interactions are timely and fully supportive of the staff's intent to meet its deadline.

While the enclosure focuses on draft rule comments, we believe our future discussions must be broadened to include other elements of the regulatory framework for implementing §50.69, including the NEI guidance that is currently under staff review. A broader discussion would promote a better understanding by all parties of the issues and concerns raised at the November 15 workshop and better support our common objective of a clear and effective rule.

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[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including regulatory aspects of generic operational and technical issues. NEI members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

Template = SECY-067 SECY-02

Enclosure 1 provides our detailed comments on the §50.69 proposals.

Enclosure 2 is a draft white paper on the bases for concluding that licensee nuclear industrial treatment programs provide adequate confidence that the RISC-3 design bases functions will be satisfied. This white paper expands on many of the comments we provided on November 15.

We look forward to future interactions with the staff on the development of 50.69. Please call Adrian Heymer (202-739-8094, aph@nei.org) or me (202-739-8081, arp@nei.org) if there are any questions on the enclosures.

Sincerely,

Anthony R. Pietrangelo

**Enclosure 1**

## Detailed Industry Comments on November 29, 2001, Preliminary NRC 10 CFR 50.69 Proposals

### General Comments

We agree that the intent of Option 2 is to permit licensees to use a risk-informed categorization process to identify the safety-significance of structures, systems and components (SSCs), and to remove SSCs of low safety significance from the scope of certain special treatment requirements. In the development of this proposal, several underlying principles have been discussed:

- The (10 CFR 50.2) design bases are not changed;
- The level of assurance of functionality may be lower for low safety-significant SSCs than for high safety-significant SSCs;
- The process should utilize performance-based approaches to the extent practical; and
- The existing regulatory framework should be utilized for implementation of 10 CFR 50.69 unless there is a compelling reason to do otherwise.

We believe these principles will help guide the development of the proposed rule and regulatory guidance.

### Specific Comments

Preamble to the Draft Rule Language

The introductory paragraph to the draft rule language states, "The proposal would permit power reactor licensees and applicants to implement an alternative regulatory framework with respect to treatment requirements currently imposed beyond practices for commercial grade equipment to add assurance of capability of structures, systems and components (SSCs) to perform their intended functions."

This statement appears to preclude a determination that commercial (nuclear industrial treatment) practices can provide adequate confidence that low safety significant SSCs remain capable of performing their design bases functions. We believe there is ample evidence and bases to support a determination that nuclear industrial treatment practices can provide adequate confidence that design bases functionality will be maintained. Enclosure 2 provides additional basis as to the adequacy of nuclear industrial nuclear treatment controls for RISC-3 SSCs.

Optional and Selective Implementation

The introduction to the rule should clearly state that §50.69 is an optional rule. In addition, the supporting Statements of Consideration should reflect policy decisions that have been made regarding selective implementation.

§50.69(c) Categorization Process Requirements

This section of the rule could be greatly simplified. The opening language states that a licensee must use a categorization process that has been approved by the NRC. It is duplicative to state all of the subsequent requirements given that a NRC approved process must be used. These requirements appear to be very similar to language used in Regulatory Guide 1.174 and should not be duplicated in the rule. There is regulatory precedence for this approach in 10 CFR 50.55a. In addition, less prescriptive language minimizes the need for future rule changes in this section.

§50.69(d), Requirements for Structures, Systems, and Components

(1)(1): The intent of the statement, "Existing regulatory requirements continue to apply," should be clarified in the supporting Statements of Consideration for the proposed rule. We agree that the intent is for regulatory requirements that were in place prior to the adoption of §50.69 would continue to apply.

(1)(i): We do not understand the statement, "The licensee shall ensure that the assumptions in the categorization process and the treatment being applied to these SSCs are consistent." Further discussion is needed on the intent of this requirement.

Need for Summary Description of Low Safety-Significant SSC Treatment program: We do not believe that it is necessary to provide a description of the industrial treatment controls for low safety-significant structures, systems and components in this rule. As presented, the level of detail is equivalent to the level of detail for a number of the 10 CFR 50, Appendix B criteria for high safety-significant SSCs. The level of detail for regulatory requirements and regulatory guidance should reflect a performance-based approach and be consistent with the safety-significance of the applicable equipment. It is acceptable for the level of detail to be lower for low safety-significant SSCs.

The rule should just list the specific criteria, and the Statement of Considerations for §50.69 should state that a program summary would be added as a new subsection in the FSAR chapter or referenced topical report on QA program descriptions, consistent with the existing regulatory requirements.

Oversight Process: We do not understand the intent of subparagraph (G), *Oversight Process.* We believe that performance assessments and audits are the means for determining whether the treatment controls are effective and sufficient. We recommend that the title of this element be changed to Assessment.

Configuration Control: We agree that configuration control is important and essential. The complete set of treatment controls and other licensee and regulatory technical requirements ensure configuration control and management. As a result, this element can be deleted because the industrial treatment program is only one part of configuration management.

(d)(3):

10 CFR 50.44: The proposed changes in paragraph (ii) need to be consistent with the revision to §50.44 that is scheduled to be published for public comment in the near future.

Environmental qualification: 10 CFR 50.49 is a special treatment requirement and should not apply. It should be sufficient to require equipment to function in the design bases service environments.

10 CFR 50.55a: The proposals do not include any relief from the requirements of 10 CFR 50.55a, which is a long and complex regulation. The intent of option 2 is to define the set of equipment that need not be subject to NRC special treatment requirements because the equipment is of low safety-significance; i.e., the failure of such equipment would not endanger the protection of public health and safety. The absolute retention of §50.55a is counter to the intent of Option 2. The continued imposition of §50.55a is not warranted. It adds unnecessary burden through the administrative compliance requirements and additional interactions through another regulatory process. Nuclear industrial treatment will provide adequate confidence that the equipment will satisfy its design bases. In addition, there are other incentives and non-NRC local initiatives that encourage the use of, and adherence to national industry standards and practices.

We recommend that §50.69 add §50.55a to the list of requirements that are not applicable to RISC-3 SSCs. The Statement of Consideration should state an expectation that a licensee adopting §50.69 would use the ASME code cases to define any set of alternative controls for RISC-3 SSCs.

10 CFR 50.65: The existing implementation guidance for 10 CFR 50.65(a)(4) provides an option for licensees to use a risk-informed process to focus the scope of (a)(4) assessments. This option should be retained for licensees adopting §50.69.

Appendix J, Type B and Type C Leakage Testing Requirements: The criteria included are duplicative of the categorization guidance in NEI 00-04 and should be deleted.

Appendix A to Part 100: §50.69 should provide sufficient regulatory flexibility that would allow a licensee to use an alternative approach for providing adequate confidence that the equipment will met its seismic design basis requirements.

§50.69 should provide a consistent degree of regulatory flexibility for all NRC special treatment requirements. In the development of an approach for risk-informing the requirements of §50.55a the use of alternative national consensus standards is permitted. The use of such alternative standards does not change a fundamental Option 2 principle of assuring that the design basis functions will be met. These same principles should be applied to seismic criteria.

The level of functional assurance for RISC-3 SSCs does not have to be the same as for RISC-1 SSCs. Under the existing §50.59 process, a licensee may change the design without prior NRC review and approval. Since Appendix A to Part 100 was introduced there has been significant progress at developing improved seismic engineering design methods and implementing guidance. Recently, national consensus code committees (International Building Code) have published detailed guidance on seismic requirements for equipment and structures for all areas of the country. This guidance includes increased design factors and requirements for buildings and equipment whose failure could pose a threat to public health and safety. In addition, there has been significant progress in understanding the critical requirements for seismic capability of equipment and structures through the development of the seismic experience programs. This resulted in the adoption of experience-based methods for qualifying specific safety-related SSCs for a seismic environment.

§50.69 should provide licensees sufficient regulatory flexibility to be able to use such methods following review and approval by the NRC staff.

10 CFR Part 54: This is clearly a special treatment requirement and should be added to the scope of 50.69 to achieve consistency across the regulatory framework.

§50.69(e) Submittal and Approval Process

License Amendment: A licensee should not have to submit a license amendment to implement §50.69. A licensee does not have to submit a license amendment to implement Risk-Informed Inservice Inspection, or Option B to Appendix J to 10 CFR 50, unless the existing Technical Specifications are being changed as a result of the adoption of those regulations.

§50.69 provides licensees the regulatory flexibility to take exception from NRC special treatment requirements for low safety-significant SSCs. Treatment requirements are being changed. The established process for changing treatment requirements is 10 CFR 50.54(a). A licensee may change RISC-1 SSC treatment requirements under §50.54(a) without a license amendment. A reduction in commitment is the measure for determining when NRC prior review and approval is required.

In its submittal notifying the NRC of its intent to adopt §50.69, a licensee makes a commitment to NEI 00-04, which would implement an industrial treatment program comprising of the following elements: design control, procurement, inspection and testing, maintenance and installation, corrective action, and assessment. A licensee would request NRC review and approval of a change to the industrial treatment program if a criterion is being deleted.

<u>§50.69 (f) Program Description, Documentation and Reporting</u>

We do not understand the need for paragraph (2). This appears to duplicate §50.69(c).

In paragraph (3), we recommend that the phrase, "...could have prevented..." should be amended to "...would have prevented..." The term would is more consistent with a performance-based regulatory process. A direct interpretation of the phrase will result in excessive and unnecessary regulatory discussions and contention on events, which are of extremely low probability, which is contrary to the intent of a risk-informed, performance-based regulatory regime.

<u>Retention of records</u>, paragraph (4): The need to retain records until the license is terminated is unnecessary. It ignores the potential for additional SSC categorization adjustments following the adoption of an Option 3 (Risk-Informing NRC Technical Requirements). It is possible that the design basis could be changed following the adoption of an Option 3 regulation. As such, the SSC might not be subject of any NRC regulatory requirements. We suggest the phase, "...until the license is terminated," be changed to, " until the SSC is no longer subject to NRC requirements."

<u>§50.69(g) Change Control</u>

1. We believe that the existing regulatory change control processes are sufficient. Over the past five years there has been an extensive effort to clarify and improve the existing regulatory change control processes. Based on these improved understandings, there should be no requirement to include a description of the categorization process in the FSAR. In notifying the Commission of its intent to adopt §50.69 a licensee makes a commitment to implement the categorization

process in accordance with an approved NRC method or guideline. A summary description of the categorization process should be described in a licensee-controlled document that is controlled through the industry's commitment management program, as described in the NRC endorsed guideline, NEI 99-04, *Guidelines for Managing NRC Commitment Changes.*

2. Changes to the treatment process have already been discussed in our comments on §50.69(e).

3. We do not understand the intent of paragraph (3). We assume the existing §50.59 process plus other regulatory change control mechanisms are sufficient.

**Enclosure 2**

# Nuclear Industrial Treatment for RISC-3 SSCs Technical Basis

This paper summarizes the rationale for concluding that nuclear industrial level treatment provides adequate confidence in the functionality of low safety-significant structures, systems and components (SSCs). These SSCs have been categorized as low safety-significant using a robust, NRC approved categorization process for risk-informing the scope of NRC special treatment requirements.

The bases for this determination are:

- The functional equipment requirements will be maintained,
- Performance data indicating that industrial treatment programs provide reliability very comparable to current special treatment requirements, and
- Monitoring to assure that the reliability of the SSCs will not be degraded to the extent that it will adversely affect safety or affect the Option 2 categorization conclusions.

## Definition of Nuclear Industrial Treatment

Nuclear industrial treatment is a set of practices that provide adequate confidence that the required functions will be satisfied under designed service conditions. Such practices are identified through applicable national, local and industry codes and standards, vendor recommendations, or operating experience. Implementation measures are applied commensurate with the relative importance and complexity of the activity, and the skill of the craft. These measures are accomplished through plant procedures, guidelines, and work instructions. The scope of treatment includes: design control, procurement, inspection, testing, work processes, maintenance, assessment and corrective action.

## Functionality of RISC-3 SSCs

Equipment functionality is assured through the implementation of a nuclear industrial set of controls, sometimes known as balance-of-plant controls, and through monitoring the functionality of the equipment—a performance-based approach. NEI-00-04, *Option 2 Implementation Guideline,* provides a high-level program summary of a nuclear industrial treatment program.

NEI-00-04 includes a performance-based functional monitoring element for RISC-3 SSCs. This monitoring element provides adequate confidence that the design bases function will be satisfied. These SSCs have been previously subject to the full spectrum of monitoring and corrective action (goal setting) requirements under the maintenance rule, 10 CFR 50.65. Under Option 2, the functional monitoring of

RISC-3 SSCs could be the same as that for implementing §50.65 without the goal setting requirement and need for balancing reliability and availability. This level of monitoring is acceptable because of the low safety significance of the equipment.

Where performance monitoring under the maintenance rule is impractical, conditioning monitoring is applied. The same would be true for the functional monitoring program for RISC-3 SSCs. Just as in the case for performance monitoring, the level of detail and documentation for a RISC-3 condition monitoring program is consistent with the safety-significance of the activity. As such, the level of detail and documentation would be reduced from that applied for the implementation of §50.65. Credit is taken for the design and procurement process, general operator machinery rounds and inspections; system engineering and management walk downs of machinery. Defects and deficiencies are resolved through the normal plant procedures and corrective action program.

*Harsh Environments and Seismic Bases*

It is impractical to perform performance monitoring for seismic and harsh environments to assure that those specific design bases attributes will be satisfied. As a result, NEI 00-04 provides an additional 15 pages of guidance with examples for RISC-3 equipment required to operate in harsh environments that are not part of normal operating conditions, or where seismic is a design basis requirement.

(i) *Harsh Environment*

The methodology for determining the application of controls and specifications for RISC-3 equipment operating in harsh environments is based on the existing methodologies for designing and assessing the operational capability of safety related and nonsafety-related equipment. The guideline offers and describes three options for replacements or alternative design: identical replacements, equivalent replacements, or new designs (design change).

Licensees already have programs and procedures in place for selecting identical and equivalent equipment and for changing design for safety-related and nonsafety-related equipment. Licensee engineering and procurement personnel make a determination on whether alternative equipment will operate in the specified environment.

Construction has been completed, equipment is in place and the engineering specifications are known. If a replacement is required, the same engineering specifications would be used but the supplier and plant controls and processes could be different. The process used for selecting the equipment and supplier is similar to that used for the replacement of existing safety-related equipment, but there would be no need for qualification testing or extensive documentation.

It is unlikely that the new design option would be used when an identical or equivalent component could be procured. This is because of the cost in developing and reviewing a new design and the procurement specification. Any new design would have to be functionally equivalent to the existing engineering procurement specifications. The new design would be subject to the §50.59 review process, and the guidance on changing design bases analytical methodologies.

(ii) *Seismic*

The guidance provided for determining the application of controls and specifications for RISC-3 equipment that is subject to seismic design specifications is similar to that described for harsh environments in (i) above. The emphasis is predominantly on replacement of equipment. There are three replacement options: identical replacements, equivalent replacement, and a new design (design change). The equipment is in place and the engineering specifications are known. If a replacement is required, the same design bases specifications would be used but the supplier, plant controls and processes could be different, and there would be no need for qualification testing or extensive documentation. The process used for selecting the equipment and supplier is similar to that used for the replacement of existing safety-related equipment.

In each of the three replacement options there would be no change in anchorage requirements. The equipment should be anchored in the same manner to the same criteria as for safety-related equipment. For the other seismic design attributes, a licensee has the flexibility to use: the same criteria and methods as for existing safety-related SSCs (Identical replacement); an equivalency determination (equivalent replacement); or an alternative design that satisfies the design bases requirements (new design replacement).

In the new design option, a licensee would have the flexibility of using the existing design criteria and methods, or alternative methods, such as those proposed in a nationally recognized consensus standard, e.g., International Building Code. Such an approach would need to include the appropriate application of the caveats included in any code or approved methodology. In addition, the ground response spectrum defined by the standard must be equivalent to, or envelope the plant specific design bases ground motion spectrum. The new design would be subject to the §50.59 review process that includes provisions on changing design methodologies.

**Comparison of performance data on SSCs subject to special treatment and nuclear industrial (BOP) treatment**

Generic industry equipment performance data and studies demonstrate the robustness of industrial grade (nonsafety-related) equipment in conditions that are comparable to design bases conditions. Such data include information on tests that have been performed on commercial equipment with built in defects to assess the robustness of the equipment to perform its designed function. In addition, a recent South Texas Project Nuclear Operating Company report (Attached) of industry reliability data over the past 30 years indicates that there is no significant reliability difference between safety-related and nonsafety-related equipment under normal operating conditions.

**Monitoring of availability and reliability of equipment**

The nuclear industrial treatment controls that would be applied to RISC-3 SSCs are the same controls and practices that are being applied to nonsafety-related equipment. They are the same controls that have enabled plants to attain capacity factors in excess of 90 percent. The availability and reliability of nonsafety-related SSCs is very similar to that of safety related equipment, even though standard industrial practices, as opposed to safety-related special treatment requirements are being applied. In addition, these same controls provide for and assure an acceptable level of availability and reliability of nonsafety-related, risk-significant equipment for compliance with the maintenance rule implementation requirements.

Even if the reliability and availability of RISC-3 equipment decreases after the adoption of Option 2, the sensitivity assessment of the potential impact on categorization of increasing the failure rate by a factor of five addresses any concern over the potential safety impact from an increased failure rate. The monitoring element would identify degradation of equipment long before the value in the sensitivity assessment was reached.

**Program description of nuclear industrial treatment**

NEI 00-04, provides a ten page description of a process for the treatment of RISC-3 SSCs with examples and a summary of a nuclear industrial set of treatment controls. The summary covers: Corrective action; maintenance, which includes corrective, predictive and preventative; configuration control covering, design control, procedural control, planning and scheduling, procurement and installation; and monitoring and assessment of equipment, encompassing inspections, testing, and audits. Licensees that choose to adopt §50.69 will commit to using an NRC approved guideline for implementation, such as NEI 00-04. Changes in categorization would be controlled through the process for managing NRC commitments as described in the NRC endorsed industry guideline, NEI-99-04, *Guideline for Managing NRC Commitments.* The existing process for managing changes to QA program descriptions, §50.54a, would govern changes to nuclear industrial treatment controls.

STPNOC Summary of Results of a Reliability Comparison between
**Safety-Related and Nonsafety-Related SSCs**

STPNOC asserts that, for components within the scope of the STPEGS Graded QA Program, non-safety-related component failure rates are not appreciably greater than corresponding safety-related component failure rates meant for similar component types. To support this assertion, STPNOC has performed a data analysis of Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX) data. Nuclear industry data reporting to the Nuclear Plant Reliability Data System (NPRDS) spans the time period from 1977 through 1996. The EPIX Maintenance Rule and Reliability Information (MRRI) database includes component failure data since 1996. NPRDS component engineering data includes indication of safety class, thus enabling a distinction between safety-related component and non-safety-related component failure rates. While the MRRI database does not include a safety-class distinction, INPO was able to provide STPNOC an MRRI database file for 1997-1999 data that is "back-linked" to NPRDS, thus providing indication of safety class. The NPRDS data and MRRI data were first analyzed separately and then merged to provide a large-scope analysis to support responses for the STPEGS GQA RAIs. STPNOC has developed a report; entitled "Safety-Related Versus Non-Safety-Related Equipment Failure Frequency Data Analysis for Nuclear Power Plants in the United States" dated April 6, 2000, describing this NPRDS-MRRI data analysis. This report is available upon request.

The scope of this merged NPRDS-MRRI analysis included consideration of over 670,000 component records and over 166,000 component failure records for those components. The historical data analyzed consisted of over 74 billion component-hours of experience. GQA RAI 42 Tables 1 and 2 (attached) provide analysis results information for all 33-component type data categories contained in the merged NPRDS-MRRI database. These tables show that the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS-MRRI data. This analysis shows that, of 33 component type categories investigated, 21 had higher safety-related failure frequency values than corresponding non-safety-related categories. Non-safety-related failure frequency values were significantly higher than corresponding safety-related failure frequencies in only one of the 33 categories (the "containment penetration" component type category). The analysis shows that, for most component types, the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data.

An argument often made in this type of comparison is that there is more safety-related component experience in the database than non-safety-related component experience. This is valid. However, the failure frequency parameters, calculated simply in terms of reported failures per component-hour of experience in this analysis, are being compared on a consistent basis. For example, in the circuit breaker component type category, there are 7,723,785,888 component-hours of safety-related circuit breaker experience. During that experience base, 6,457 failures of safety-related circuit breakers were reported, yielding a failure frequency of 8.36E-07 (=6,457/7,723,785,888) failures per component-hour. Similarly, there are 1,777,678,176 component-hours of non-safety-related circuit breaker experience in the database. During that experience base, 1,345 failures of non-safety-related circuit breakers were reported, yielding a failure frequency of 7.57E-07 (=1,345/1,777,678,176) failures per component-hour. The failure frequency parameters are calculated and compared on the same basis. One can conclude that we have a greater degree of confidence that the historical failure frequency for safety-related circuit breakers represents the "true" failure frequency (calculated for infinite experience), than we do for the non-safety-related circuit breakers. However, in this case, there are large numbers of component-hours of experience for both safety-related and non-safety-related components, indicating that we have relatively high confidence in both results.

Another way of looking at this is that, if we were to "scale" the safety-related experience down to the non-safety-related experience level, we would multiply both the component-hours of experience and the reported failure count by the ratio of non-safety-related to safety-related component-hours of experience (1,777,678,176/6,457/7,723,785,888). If we do this, we get the same results as with the actual experience numbers. Likewise, we would get the same results if we were to scale the non-safety-related experience up to the safety-related experience. That is, if we increase or decrease the component-hours of experience for a component type category of interest in the database by some factor, we would expect to have a higher or lower number of reported failures by the same factor.

**GQA RAI 42 TABLE 1. SUMMARY OF MERGED NPRDS-MRRI COMPONENT TYPE CATEGORY SAFETY-RELATED VERSUS NON-SAFETY-RELATED FAILURE FREQUENCY COMPARISON RESULTS**

| COMPONENT DATA CHARACTERISTIC DESCRIPTION | NUMBER IN CATEGORY |
|---|---|
| TOTAL COMPONENT CATEGORIES ANALYZED: | 33 |
| NUMBER OF CATEGORIES WITH SAFETY-RELATED DEMAND FAILURE RATE GREATER THAN NON-SAFETY-RELATED FAILURE FREQUENCY: | 21 |
| NUMBER OF CATEGORIES WITH NON-SAFETY-RELATED DEMAND FAILURE RATE GREATER THAN SAFETY-RELATED FAILURE FREQUENCY: | 12 |
| CATEGORIES WHERE SAFETY-RELATED DEMAND FAILURE RATE IS MORE THAN A FACTOR OF 2 LESS THAN NON-SAFETY-RELATED FAILURE FREQUENCY: | 3 |
| CATEGORIES WHERE SAFETY-RELATED DEMAND FAILURE RATE IS MORE THAN A FACTOR OF 3 LESS THAN NON-SAFETY-RELATED FAILURE FREQUENCY: | 1 |
| TOTAL COMPONENT-HOURS OF EXPERIENCE DATA: | 74,615,379,120 |
| TOTAL FAILURE EVENT RECORDS ANALYZED: | 116,413 |
| TOTAL FUNCTIONAL FAILURES IN RECORD SET: | 116,413 |
| SAFETY-RELATED COMPONENT-HOURS OF EXPERIENCE: | 60,968,091,504 |
| NON-SAFETY-RELATED COMPONENT-HOURS OF EXPERIENCE: | 13,647,287,616 |
| SAFETY-RELATED FUNCTIONAL FAILURES IN RECORD SET: | 93,697 |
| NON-SAFETY-RELATED FUNCTIONAL FAILURES IN RECORD SET: | 22,716 |

Attachment

# GQA RAI 42 TABLE 2. MERGED NPRDS-MRRI COMPONENT TYPE CATEGORY DATA ANALYSIS RESULTS

| COMPONENT TYPE ID | COMPONENT DESCRIPTION | SAFETY-RELATED COMPONENT-HOURS | SAFETY-RELATED COMPONENT FAILURES | NON-SAFETY-RELATED COMPONENT-HOURS | NON-SAFETY-RELATED COMPONENT FAILURES | SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT-HOUR) | NON-SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT-HOUR) | NON-SAFETY-RELATED > SAFETY-RELATED FREQUENCY | NON-SAFETY-RELATED > 2*SAFETY-RELATED FREQUENCY | NON-SAFETY-RELATED > 3*SAFETY-RELATED FREQUENCY |
|---|---|---|---|---|---|---|---|---|---|---|
| ACCUMU | Accumulators, tanks, air receivers | 320,096,904 | 286 | 51,778,080 | 9 | 8.93E-07 | 1.74E-07 | NO | NO | NO |
| AIRDRY | Air dryers, dehumidifiers | 20,415,504 | 149 | 26,830,248 | 168 | 7.30E-06 | 6.26E-06 | NO | NO | NO |
| ANNUNC | Annunciator modules, alarms | 21,289,632 | 9 | 50,028,864 | 4 | 4.23E-07 | 8.00E-08 | NO | NO | NO |
| BATTRY | Batteries, battery chargers | 188,054,640 | 1,109 | 34,188,936 | 170 | 5.90E-06 | 4.97E-06 | NO | NO | NO |
| BLOWER | Blowers, compressors, fans, vacuum pumps, cooling units | 327,993,024 | 1,601 | 106,903,032 | 808 | 4.88E-06 | 7.56E-06 | YES | NO | NO |
| CKTBRK | Circuit breakers, contactors, controllers | 7,723,785,888 | 6,457 | 1,777,678,176 | 1,345 | 8.36E-07 | 7.57E-07 | NO | NO | NO |
| CRDRVE | Rod drive mechanism, hydraulic control unit | 2,386,497,960 | 3,049 | 84,631,656 | 13 | 1.28E-06 | 1.54E-07 | NO | NO | NO |
| DEMIN | Demineralizers, ion exchangers | 44,136,024 | 72 | 72,290,016 | 255 | 1.63E-06 | 3.53E-06 | YES | YES | NO |
| ELECON | Electrical conductors, bus, cable, wire | 47,311,920 | 229 | 2,645,688 | 9 | 4.84E-06 | 3.40E-06 | NO | NO | NO |
| ENGINE | Engines (gas, diesel) | 42,954,168 | 1,364 | 3,009,408 | 45 | 3.18E-05 | 1.50E-05 | NO | NO | NO |
| FILTER | Filters, strainers, screens | 194,277,624 | 492 | 48,874,176 | 90 | 2.53E-06 | 1.84E-06 | NO | NO | NO |
| GENERA | Generators, inverters, motor generators | 155,717,880 | 1,618 | 41,882,208 | 400 | 1.04E-05 | 9.55E-06 | NO | NO | NO |
| HEATER | Electric heaters | 66,201,648 | 215 | 6,761,136 | 12 | 3.25E-06 | 1.77E-06 | NO | NO | NO |
| HTEXCH | Heat exchanger, | 414,941,280 | 1,468 | 356,166,816 | 1,105 | 3.54E-06 | 3.10E-06 | NO | NO | NO |

| COMPONENT TYPE ID | COMPONENT DESCRIPTION | SAFETY-RELATED COMPONENT-HOURS | SAFETY-RELATED COMPONENT FAILURES | NON-SAFETY-RELATED COMPONENT-HOURS | NON-SAFETY-RELATED COMPONENT FAILURES | SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT-HOUR) | NON-SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT-HOUR) | NON-SAFETY-RELATED > SAFETY-RELATED FREQUENCY | NON-SAFETY-RELATED > 2*SAFETY-RELATED FREQUENCY | NON-SAFETY-RELATED > 3*SAFETY-RELATED FREQUENCY |
|---|---|---|---|---|---|---|---|---|---|---|
| | condenser, steam generator | | | | | | | | | |
| IBISSW | Bistable, switch (mechanical, electronic) | 4,583,711,328 | 7,309 | 1,168,451,712 | 1,367 | 1.59E-06 | 1.17E-06 | NO | NO | NO |
| ICNTRL | Instrument controllers | 898,170,120 | 2,617 | 754,194,216 | 2,054 | 2.91E-06 | 2.72E-06 | NO | NO | NO |
| INDREC | Indicators, recorders, gauges | 1,165,607,472 | 1,572 | 467,257,680 | 452 | 1.35E-06 | 9.67E-07 | NO | NO | NO |
| INTCPM | Integrator/computation module | 5,147,811,144 | 6,485 | 1,254,243,600 | 1,619 | 1.26E-06 | 1.29E-06 | YES | NO | NO |
| IPWSUP | Electronic power supply | 2,421,707,832 | 2,710 | 307,631,568 | 421 | 1.12E-06 | 1.37E-06 | YES | NO | NO |
| ISODEV | Isolation devices | 1,331,855,808 | 774 | 158,385,984 | 96 | 5.81E-07 | 6.06E-07 | YES | NO | NO |
| IXMITR | Transmitters, detectors, elements | 4,019,348,664 | 9,775 | 950,110,272 | 1,298 | 2.43E-06 | 1.37E-06 | NO | NO | NO |
| MECFUN | Governors, couplings, gear boxes | 145,165,920 | 790 | 64,157,760 | 346 | 5.44E-06 | 5.39E-06 | NO | NO | NO |
| MOTOR | Motors (electric, hydraulic, pneumatic) | 894,689,184 | 1,212 | 217,592,112 | 450 | 1.35E-06 | 2.07E-06 | YES | NO | NO |
| PENETR | Containment penetrations, air locks, hatches | 562,056,384 | 922 | 2,977,224 | 121 | 1.64E-06 | 4.06E-05 | YES | YES | YES |
| PIPE | Pipes, fittings, rupture discs | 127,431,000 | 415 | 22,303,536 | 104 | 3.26E-06 | 4.66E-06 | YES | NO | NO |
| PUMP | Pumps, eductors | 745,949,736 | 4,797 | 160,325,160 | 1,136 | 6.43E-06 | 7.09E-06 | YES | NO | NO |
| RELAY | Relays | 8,447,729,424 | 2,922 | 348,630,792 | 275 | 3.46E-07 | 7.89E-07 | YES | YES | NO |
| SUPORT | Supports, hangers, snubbers | 899,955,000 | 908 | 38,081,304 | 44 | 1.01E-06 | 1.16E-06 | YES | NO | NO |
| TRANSF | Transformers, shunt reactors | 259,542,552 | 161 | 194,772,312 | 150 | 6.20E-07 | 7.70E-07 | YES | NO | NO |
| TURBIN | Turbines (steam, gas) | 28,295,040 | 363 | 48,378,888 | 380 | 1.28E-05 | 7.85E-06 | NO | NO | NO |

| COMPONENT TYPE ID | COMPONENT DESCRIPTION | SAFETY-RELATED COMPONENT-HOURS | SAFETY-RELATED COMPONENT FAILURES | NON-SAFETY-RELATED COMPONENT-HOURS | NON-SAFETY-RELATED COMPONENT FAILURES | SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT-HOUR) | NON-SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT-HOUR) | NON-SAFETY-RELATED > SAFETY-RELATED FREQUENCY | NON-SAFETY-RELATED > 2*SAFETY-RELATED FREQUENCY | NON-SAFETY-RELATED > 3*SAFETY-RELATED FREQUENCY |
|---|---|---|---|---|---|---|---|---|---|---|
| VALVE | Valves, dampers | 13,192,044,024 | 20,420 | 3,375,651,384 | 4,061 | 1.55E-06 | 1.20E-06 | NO | NO | NO |
| VALVOP | Valve operators | 4,112,662,464 | 11,279 | 1,450,059,720 | 3,909 | 2.74E-06 | 2.70E-06 | NO | NO | NO |
| VESSEL | Pressure vessel, reactor vessel, pressurizer | 30,684,312 | 148 | 413,952 | 0 | | | | | |
| | | | | | | | | | | |
| | TOTAL: | 60,968,091,504 | 93,697 | 13,647,287,616 | 22,716 | | | 12 | 3 | 1 |