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August 13, 1993

U.S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Application for Exemption from 10CFR50 Appendix J and Proposed Amendment to the Operating License (PCOL-93/010)

GNRO-93/00100

Gentlemen:

In accordance with 10CFR50.12 and 10CFR50.90, Entergy Operations Inc. (EOI) hereby applies for an exemption from 10CFR50 Appendix J "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors" regarding local leak rate and integrated leak rate testing intervals, and the Technical Specification changes necessary to implement the exemption, for Grand Gulf Nuclear Station (GGNS).

Per Appendix J, three sets of Type A tests are required to be performed at approximately equal intervals during each 10-year service period. Type B and C tests are required during shutdown for refueling with intervals not to exceed 2 years. Additionally, Type B airlock seal tests are required every 3 days for frequently opened containment airlocks. The proposed changes establish Type B and C testing intervals based on component performance history and establish a 10 year interval for Type A testing.

We recognize that both the NRC and the industry have been working on proposed changes to Appendix J for a number of years. While we fully support these efforts, we also recognize that rulemaking has a number of uncertainties associated with it including difficulty in predicting the final scope of the rule changes and the potential for extended delays. Based on our understanding of the NRC's present plans for Appendix J rulemaking, we do not believe that the plant-specific needs of Grand Gulf will be met in achieving a significant reduction in regulatory burden. Awaiting rulemaking is, therefore, not a viable option and would likely result in the same request for plant-specific exemption as we are submitting today.

A017 1/1

We understand the regulatory difficulties in preparing rulemaking which will preserve an adequate level of safety yet not overburden licensees, particularly when most rulemaking must take into account widely varying plant designs and levels of performance. Inevitably, some plants in every rulemaking situation will find themselves assuming a significant cost burden without a commensurate increase in safety. Such is the case for Grand Gulf with respect to the present (and, apparently, the future) formulation of Appendix J.

We are fortunate that NRC management has recognized the predicament that licensees face when confronted with required compliance at a cost out of proportion to the safety benefit. On May 4, at the annual Regulatory Information Conference, Dr. Murley announced a pilot program to give special consideration to licensee requests for changes requiring staff review that involve high cost and relatively low safety benefit. In response to Dr. Murley's initiative, Entergy Operations met with NRR staff on June 8, 1993 to present an initial list of cost beneficial licensing actions (CBLAs) and to resolve process questions associated with the CBLA pilot program.

As we discussed on June 8, this proposed exemption to Appendix J is being submitted under the CBLA program. Although the change does have safety benefit (e.g., occupational dose reduction on the order of 140 rem), its major benefit is economic. Grand Gulf expects cost reductions of at least $20 million over the remaining plant life with the potential for significantly more. This level of burden reduction is unique to Grand Gulf and cannot be achieved under the rulemaking approach which we believe the staff is presently considering.

As you know, Grand Gulf has made, and intends to make, further submittals under the CBLA program. Because of this, the staff requested in our June 8 meeting that CBLA submittals be clearly prioritized. Accordingly, in cases where the Appendix J request is competing with another of our submittals for the same staff review resources, the Appendix J exemption request should receive priority treatment.

The details of the proposed containment leakage testing program as well as supporting justification are provided in Attachment 2. Attachment 3 provides a copy of the marked-up pages from the GGNS Technical Specifications. Attachment 4 is an information copy of the proposed Technical Specifications.

In accordance with the provisions of 10CFR50.4, the signed original of the requested exemption and amendment is enclosed. This exemption and amendment has been reviewed and accepted by the Plant Safety Review Committee and the Safety Review Committee.

This request has been reviewed against the criteria of 10CFR51.22 for categorical exclusion from environmental impact considerations. This request does not involve a significant hazards consideration, does not significantly increase the amounts or change

the types of effluent that may be released offsite, nor would it significantly increase individual or cumulative occupational radiation exposures. Therefore, this request meets the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

Yours truly,

for

CRH/MJM/WBB/ams

attachments: 1. Affirmation per 10CFR50.30
2. GGNS PCOL-93/010
3. Mark-up of Affected Technical Specification Pages
4. Proposed Technical Specification Pages - Information Only

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BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-29

DOCKET NO. 50-416

IN THE MATTER OF

MISSISSIPPI POWER & LIGHT COMPANY
and
SYSTEM ENERGY RESOURCES, INC.
and
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
and
ENTERGY OPERATIONS, INC.

AFFIRMATION

I, M. J. Meisner, being duly sworn, state that I am Director, Nuclear Safety & Regulatory Affairs of Entergy Operations, Inc.; that on behalf of Entergy Operations, Inc., System Energy Resources, Inc., and South Mississippi Electric Power Association I am authorized by Entergy Operations, Inc. to sign and file with the Nuclear Regulatory Commission, this application for amendment of the Operating License of the Grand Gulf Nuclear Station; that I signed this application as Director, Nuclear Safety & Nuclear Affairs of Entergy Operations, Inc.; and that the statements made and the matters set forth therein are true and correct to the best of my knowledge, information and belief.

M. J. Meisner

STATE OF MISSISSIPPI
COUNTY OF CLAIBORNE

SUBSCRIBED AND SWORN TO before me, a Notary Public, in and for the County and State above named, this 13th day of August, 1993.

(SEAL)

Janie F. Calloway
Notary Public

My commission expires:
My Commission Expires November 11, 1996

1.0 Introduction

In response to the NRC's cost beneficial licensing action (CBLA) initiative Grand Gulf proposes to amend its approach to containment leakage testing in accordance with 10CFR50 Appendix J.

We expect our performance-based program to return considerable dividends in safety, plant reliability and cost reduction (approximately $20 million over the life of the facility).

1.1 Recent NRC Initiatives

Over the past several years, the NRC has devoted substantial resources to identification and analysis of the negative effects of regulation, culminating in a number of related initiatives such as the CBLA program, the Regulatory Review Group, and the elimination of requirements marginal to safety program.

A common thread in these efforts is the recognition that, in some cases, regulations are overly prescriptive resulting in the expenditure of unnecessary resources which could better be applied to more safety significant areas. The solution usually proposed for these cases is to convert the prescriptive requirements to performance-based requirements.

The Appendix J containment leakage testing requirements have received extensive regulatory scrutiny in this regard. The NRC has indicated its intent to pursue action on Appendix J (e.g., 57 FR 4166) and has published several possible approaches. For instance, to revise Appendix J to a performance-based regulation, the staff suggested the following (58 FR 6197):

> Limit revised rule to a new regulatory objective: In order to ensure the availability of the containment during postulated accidents, licensees should either: (a) test overall containment leakage no longer than every 10 years and test pressure-containing or leakage-limiting boundary valves on an interval based on the performance history of the equipment; or (b) provide an on-line monitoring capability of containment isolation status.

Grand Gulf believes that the first suggested approach (item (a), above) has considerable merit. In fact, Grand Gulf personnel outlined such a program and presented it to the NRC's workshop on elimination of requirements marginal to safety in April, 1993.

Grand Gulf fully supports the efforts of the NRC, NUMARC, Owners Groups and other industry groups in working towards reform of Appendix J. However, although the NRC and industry have been investigating various alternative approaches to the regulation of containment leakage testing for some time, rulemaking is a long and arduous process, and the ultimate nature and scope of relief is uncertain at best. When Dr. Murley announced the CBLA program at the 1993 Regulatory Information Conference, we recognized the unique opportunity to request plant-specific relief under Appendix J.

1.2 Proposed Grand Gulf Leakage Testing Program

Grand Gulf is requesting NRC approval of the necessary Technical Specification changes and Appendix J exemptions to implement a performance-based containment leakage testing program. As suggested by the staff, the essential elements of the program consist of:

- Type A testing on a frequency not to exceed 10 years
- Type B and C testing on a pre-determined frequency based on component performance history, and
- Containment airlock testing based on performance history.

The proposed changes affect only testing frequencies. Current test methods, acceptance and failure criteria, allowable leakage limits and the like remain unchanged.

The remainder of this document presents the details of the proposed changes and their technical basis. In particular:

- Section 2 provides a discussion of performance-based testing - its basis and applicability to the proposed changes
- Section 3 reviews the proposed Type B and C testing program, the testing history at Grand Gulf, our basis for the program (including an assessment of its risk impact) and a discussion of the cost reduction achievable by the change
- Section 4 provides similar information for the proposed Type A testing program
- Section 5 discusses proposed changes to the containment airlock testing presently contained in the Technical Specifications
- Section 6 summarizes our basis for exemption to the applicable sections of 10CFR50 Appendix J, and
- Section 7 provides our basis for concluding that no significant hazards considerations exist for the proposed Technical Specification changes.

2.0 Performance-Based Testing

2.1 Background

To further enhance safety, reliability and overall performance, the nuclear industry is frequently adopting more of a performance-based approach in planning and controlling plant operations.

The advantages of using performance as a determining factor, as opposed to more rigid prescriptive approaches, include the following.

- Problem Identification - Inherent with a performance-based program are quantifiable criteria which provide a direct indication of performance problems. All personnel have a visible and measurable means to assess performance.

- Integration - Performance-based programs allow explicit consideration of multiple factors affecting overall plant performance. Whereas prescriptive criteria tend to focus on a single aspect of performance, e.g. the safety function of a single component or system, performance-based programs may consider other equally important factors, such as overall plant safety, personnel exposure, etc.

- Optimization - Often, there are tradeoffs to be considered in establishing requirements for systems and equipment. For example, testing is necessary to ensure operability of equipment, but may also contribute to the unavailability of that equipment. Since there are many factors which contribute to variability among plants, it is difficult for prescriptive criteria to achieve an optimum level of performance across all plants. Performance-based criteria allow such optimization to occur.

- Resource Allocation - Use of performance-based criteria encourages the application of limited resources to real performance problem areas. Since such problem areas cannot necessarily be identified *a priori*, nor are such problem areas necessarily "generic" (i.e. applicable to all plants), prescriptive criteria may not adequately address real problems. Conversely, prescriptive criteria may place undue emphasis on areas where problems are not occurring, and thus drain limited resources away from true problems.

- Flexibility - A performance-based testing program provides the flexibility to achieve the desired goal via alternative means. For example, system or component operability may be demonstrable through a number of tests. Rather than performing multiple tests on the same component to demonstrate compliance with multiple regulations, compliance may be demonstrated through alternative tests or programs performed for other reasons. A performance-based testing program inherently has the flexibility to take advantage of such situations.

Performance-based testing programs have four elements:

- Performance goals
- Performance factors
- Performance criteria
- Performance evaluation

Performance goals establish the bases for a performance-based testing program. Performance goals determine the level of functionality required of systems and equipment. The means to achieve this level of functionality can then be defined, and the actual achievement measured and monitored.

Performance factors are those factors which affect either the achievement of the goal, or perhaps the definition of the goal itself. They may include competing elements such that an optimized level of performance may be derived.

The performance criteria provide quantitative bases for the testing program. They include acceptance criteria for the tests, as well as quantitative aspects of the testing program.

Finally, the performance evaluation phase involves periodic feedback on performance to make sure that performance goals are being met.

The performance elements for the proposed GGNS Appendix J testing program are discussed in the following sections.

2.2 Performance Goals

The principal objective of the Appendix J testing program is to ensure that the reactor containment integrity can be maintained isolated with high reliability following accidents. In order to achieve this, the containment penetrations must limit leakage from containment, under pressure, to within specified limits.

The GGNS Individual Plant Examination (IPE) examined the containment isolation function for severe accident scenarios. The study concluded that containment isolation failure sequences were negligible contributors to overall plant risk, with a frequency of occurrence of less than 1E-7 per year. This is due to the overall low frequency of severe accident challenges to containment (less than 1E-4 per year) and the low probability for failure of containment isolation (less than 1E-3). Other potential containment failure modes were found to be more significant than containment isolation failure.

Based on the IPE results, it is reasonable to set a general level of performance for the containment isolation function, such that the IPE results remain valid, i.e. that severe accident sequences involving containment isolation failure remain below 1E-7 per year. This implies that an acceptable level of performance for containment isolation is that leakage from the containment be less than specified limits with high reliability (i.e. the probability of failure to achieve this function is less than 1E-3). This goal can be further defined by specifying the acceptable level of leakage from containment. At present, this is specified in the GGNS Technical Specifications as 0.437 wt % of the containment air per day at a pressure of 11.5 psig. Although this degree of leakage is very low, no change to this allowable leakage rate is being proposed.

2.3 Performance Factors

A detailed engineering assessment of the GGNS containment penetrations has been performed to determine past performance factors affecting performance (see Section 3), and implications of component failure on a penetration-by-penetration basis. The following summarizes the factors that have been identified as important in establishing an alternative testing program.

- Past Component Performance - Approximately 85% of the GGNS containment penetrations and isolation valves have never failed a Type B or Type C Test in over seven years of commercial operation. Of those components which have failed tests in the past, 40% of them have failed more than once. The conclusion from this investigation of past performance is that some components/penetrations are more susceptible to leakage than others, due to design and/or service. The likelihood of failure is very component-specific.

- Service - The operating environment of a component is important in determining its likelihood of failure. For instance, components in a flowing steam environment have been found to leak with a higher probability, due to the effects of valve seat erosion. Valves which open and close frequently during normal operations are more likely to experience leakage.

- Design - Component type and penetration design are found to be significant in predicting the likelihood of a component leaking. For example, one type of penetration is responsible for almost every observed failure of a Type B test at GGNS. Similarly, motor-operated valves are found to leak less frequently than check valves.

- Safety Impact - Some components/penetrations are more significant than others, in terms of the potential impact of their failure in limiting releases from containment under accident conditions. One example is penetration size. The penetration size determines its potential leakage rate. At GGNS, LLRT acceptance criteria are generally lower for smaller penetrations. This means that a "failure" of a small penetration may result in a negligible overall containment leakage rate. Another example would be the "downstream" system design/service. Leakage into a system containing water which is designed for high pressure, for example, would limit the subsequent transport of radionuclides into the environment. The overall "decontamination factor" for the release pathway would be very high. As a result, some penetrations are more important than others in ensuring that the safety function of containment isolation is achieved.

- Alternative Tests - Multiple tests may be performed on specific containment isolation components. In some cases, the isolation function of the component may be verified through test/maintenance programs other than the Appendix J LLRTs. The proposed GGNS program recognizes these situations and utilizes the results of other programs to ensure that the essential containment isolation function is achieved with the required level of confidence.

2.4 Performance Criteria

Specific criteria for the proposed GGNS performance-based Appendix J testing program include acceptance criteria for the tests and the establishment of alternative test intervals.

At present, no changes to the owner's allowable leakage rates are being proposed. These LLRT acceptance criteria are established in a conservative manner. In setting these limits, credit is not taken for the effects of multiple penetration barriers in further reducing overall leakage through the penetration. Therefore, the acceptance criteria for the LLRTs remain at very low values.

The proposed program does, however, include alternative testing intervals for specific components. The alternative testing intervals consider each of the factors identified in Section 2.3. For instance, all containment penetrations have not demonstrated the same degree of reliability. In fact, there is variability in performance on a component-by-component basis. This behavior is not unexpected, and in fact is the basis for the broad spread generally observed in "generic" industry data analyses.

In order to estimate the likelihood that a specific component, or small group of components, is better or worse than average, a statistical analysis of the component/group performance may be conducted. With no prior information concerning component performance, one would have to assume that the component behaves as indicated by generic data. As more component-specific (or group-specific) performance history is accumulated, one begins to see patterns differentiating component-specific performance from the broader generic data. Quantitatively, this is commonly performed in probabilistic safety assessments by means of a procedure known as Bayesian analysis. The failure characteristics of the component or group become more and more specific as more performance history accumulates. In the limit of a large body of component-specific data, the failure characteristics for the component approach those which would be derived via classical statistics. In the opposite limit of very little data, the failure characteristics of the component are given by the generic data.

Mathematically, the failure rate for a component as a function of its performance history is given by:

$$P(\lambda|X) = P_{prior}(\lambda) \cdot P'(X|\lambda)$$

where $P(\lambda|X)$ is the probability that the component failure rate is λ, given operating history X; $P_{prior}(\lambda)$ is the probability that the component failure rate is λ based on generic data; and $P'(X|\lambda)$ is the probability that operating history X would be observed if the failure rate were indeed λ.

Using generic failure rate data for leakage for motor-operated valves, pneumatic valves and check valves, a Bayesian analysis was performed to determine the expected component failure rate as a function of demonstrated time without failures. The results are shown in Figure 2-1. This figure shows the mean failure rate from the probability distribution function derived on the basis of zero observed failures in t component-years of service. For motor-operated valves, for instance, the mean failure rate decreases from the generic value of 1E-7/hr initially to less than 5E-8/hr after fourteen component-years of service with no leakage.

These failure probabilities can be used to estimate the likelihood of penetration leakage under a performance-based Appendix J program.

The probability for penetration leakage is estimated by

$$P_{leakage} \cong \frac{(\lambda_i T)}{2} * \frac{(\lambda_i T)}{2}$$

where the subscripts designate each of the assumed two components forming the penetration barrier, and T is the mean time between tests. The proposed test program includes extending the test interval to every five years after a component successfully passes two consecutive Type B or C tests, and then to ten years after three successive successful Type B or C tests. Assuming the maximum test intervals, the failure probabilities for different types of penetrations are calculated versus time. Results for typical penetrations are shown in Figure 2-2. The calculated probabilities represent projections of the penetration leakage probability, given the known performance up to the current time. For example, the probability between 0 and 2 years represents the mean probability of penetration failure during that interval, given that no failures have occurred at 0 years. The probability between 2 and 5 years is based on the fact that no failures occurred within the first two years. The probability values initially decrease due to the decreasing failure rate estimates for components that have experienced no failures in time t (as shown in Figure 2-1). After five years, the penetration leakage probability increases. This is due to the increase in the next test interval from every two years to every five years. Another increase occurs at eight years, when the component goes from a five-year interval to a ten-year interval. Thereafter, the probabilities decrease, since the test interval remains constant while the failure rate estimate continues to decrease.

Even with the extended test interval, penetration leakage probabilities remain near or below their original values (i.e. the values calculated prior to having any performance history) when the test interval is extended to five years. With the subsequent increase to a ten-year interval, the failure probability for a penetration containing AOVs increases only slightly above its original value for the first such test interval. The probability decreases thereafter. At all times, the probability remains below 1.0E-4. Penetrations containing motor-operated valves and check valves exhibit the same general behavior. The increase in probability when the test interval is first extended to ten years is somewhat greater, but the overall penetration leakage probability remains well below the leakage probability for a penetration containing AOVs. The temporary increase when the test interval is first extended to ten years is still well within the overall uncertainty bands for penetration leakage probability. As demonstrated in Figure 2-2, the overall probability for containment isolation failure is not likely to be negatively impacted by the proposed testing program. The overall performance goal for the testing program, i.e. assurance that containment isolation failure sequences remain negligible and bounded by the IPE assumptions, is achieved by the proposed program.

Figure 2-1
Performance-Based Testing Program
Isolation Valve Failure Rate Estimates

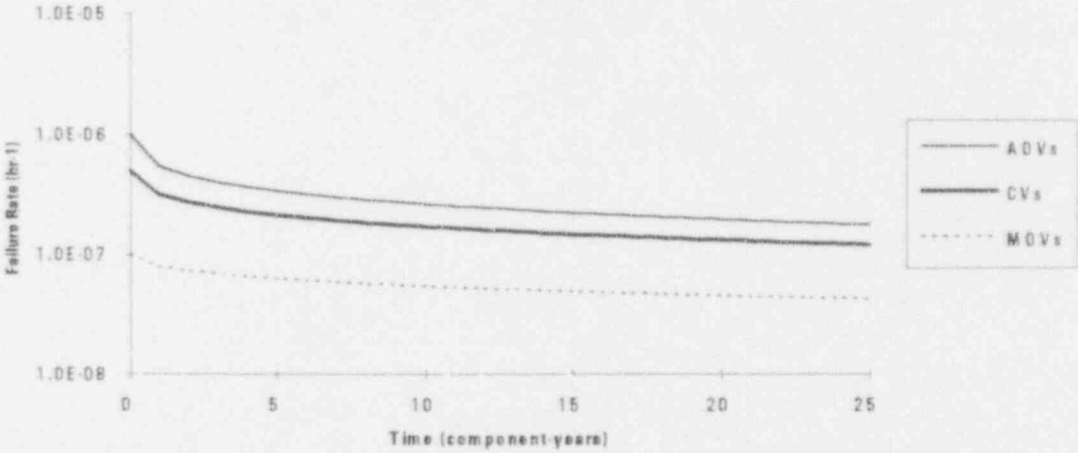
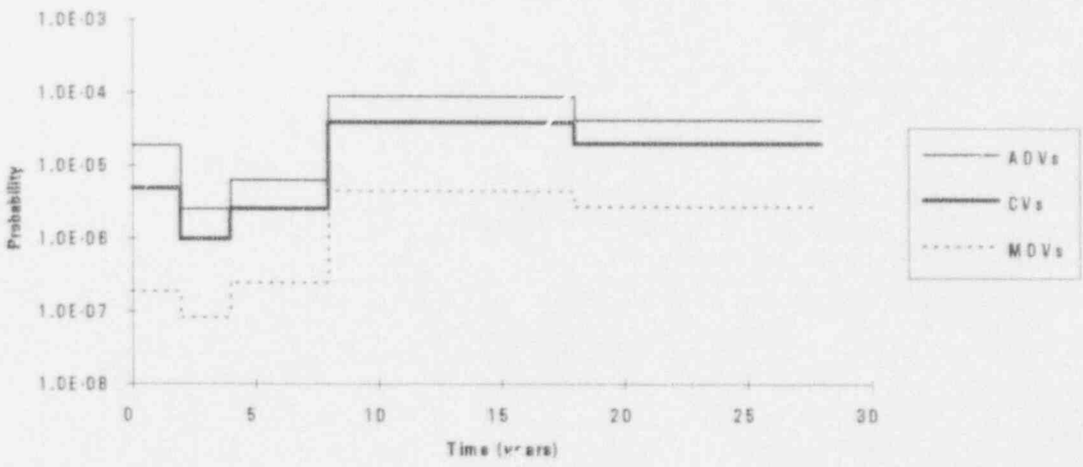


Figure 2-2
Performance-Based Testing Program
Containment Penetration Leakage Probability



2.5 Conclusions

The proposed program changes based on performance criteria are justifiable.

- The overall performance goal of ensuring containment isolation with a high degree of reliability can be demonstrated based on the continued good performance of most containment penetrations.

- Qualitative performance factors are considered in establishing performance criteria. These factors consider: past performance, component design, component service, safety impact and alternative testing programs.

- Quantitative acceptance criteria for testing continue to be set in a conservative manner.

- Quantitative criteria for testing intervals are based on demonstrated component performance and the achievement of the overall performance goal.

The program establishes a rational basis for containment isolation testing, consistent with preserving an adequate level of safety and the evolving practice of performance-based facility management in the nuclear industry.

3.0 Type B and C Testing - Proposed Changes and Basis

3.1 Proposed Changes

"Type B Tests" are defined in Appendix J Section II.G as "tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage limiting boundary for the following primary reactor containment penetrations:

1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.

3. Doors with resilient seals or gaskets except for seal welded doors.

4. Components other then those listed in II.G.1, II.G.2, or II.G.3 which must meet the acceptance criteria in III.B.3."

"Type C Test" is defined in Section II.H as a "test intended to measure containment isolation valve leakage rates. The containment isolation valves are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrumentation valves;

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under post accident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors."

Exemption is requested from the following paragraph in Section III.D.2 for Type B tests intervals:

> "(a) Type B tests, except tests for air locks, shall be performed during reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than 2 years."

Exemption is requested from the following paragraph in Section III.D.3 for Type C tests intervals:

> "Type C tests. Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years."

GGNS intends to commit to a performance-based testing program (described in Section 3.2 below) for the exempted requirements.

In accordance with 10CFR50.90, Surveillance Requirement 4.6.1.2.d associated with Technical Specification 3/4.6.1.2 "Containment Leakage" is being revised to establish a maximum test interval based on performance. Specifically, the maximum test interval will be 2 years for components that pass 1 test or that have failed the previous test, 5 years for components that pass 2 consecutive tests, and 10 years for components that pass 3 consecutive tests. In addition, the revision will add a footnote that states that a failure for an LLRT is one that exceeds the owner's allowable leakage rate and a footnote that states this is an exemption to 10CFR50, Appendix J requirements.

3.2 Proposed Type B and C Testing Program

In lieu of 10CFR50 Appendix J Sections III.D.2(a) and III.D.3, GGNS proposes to perform Type B & C tests at intervals based on the performance of each component. The test intervals will be established for each component by evaluating testing history and adjusting the testing intervals based on certain defined criteria, including engineering judgment.

Test intervals will be established by reviewing the last 3 consecutive Type B/C tests performed and determining if the Type B/C tests for each component had passed or failed. A failure is a Type B/C test that exceeded the owner's allowable leakage rate. To be considered consecutive, tests must be performed in sequence at least 12 months apart with a minimum of 12 months inservice time prior to the test. The test interval assignments and all supporting evaluations will be documented in plant records. If it is determined that a component will be placed on a 2 year interval regardless of its historical performance, interval establishment will not be required.

The owner's allowable leakage rate will be assigned for each Type B and Type C component. The owner's allowable leakage rate assigned to each component is the administrative leakage rate limit and will be specified to be indicative of the potential onset of valve degradation. All owner's allowable leakage rate assignments will be documented.

Specifically, the maximum test interval between Type B and C tests will be as follows:

- Every 2 years for components that pass 1 test or that have failed the previous test
- Every 5 years for components that pass 2 consecutive tests
- Every 10 years for components that pass 3 consecutive tests

Under this approach, penetrations and containment isolation valves that do not demonstrate excessive leakage will have surveillance intervals that reflect their reliable performance. Any valves or penetrations that demonstrate excessive leakage and require repair or replacement will be tested during the next scheduled refueling outage to monitor the effectiveness of the corrective action.

A review of all consecutively passed tests will be performed to determine if the leakage was high, erratic or indicative of a degrading trend. High or erratic leakage could indicate a potential failure prior to the next scheduled Type B/C test. In order to evaluate the probability for failure the responsible engineer will consider the following information:

- Past failures - To determine if the component had failed a previous Type B/C test, if the failure was catastrophic (greater than .60L_a) and if the appropriate corrective action was taken to avoid recurrence.

- Component application\Usage factor - To determine if the component is normally open, normally closed, used for system isolation, used for flow control, or used in any way that could induce a higher wear rate.

- System function - To determine if the component is in a system that is used for normal plant operation, such as main steam, feedwater, etc. and could induce a higher wear rate.

- Component size - To determine if the size of the component has any effect on probability of failure or increases the consequences of failure.

- Operation medium - To determine if the component is in an operating medium that could induce a higher wear rate.

Industry operating experience is reviewed to identify any generic problems including those associated with containment isolation valves and other components subject to Appendix J testing. Any generic problems identified will prompt a review to determine if the problem could affect the Type B/C test performance of any component(s). If the problem could affect test performance, an evaluation will be done and the test interval will be adjusted to an appropriate interval. The problem will be monitored until it is resolved or until the problem is corrected.

A review will be performed on each failure to determine if the failure was generic or isolated. If it is determined that the failure was generic, all other components that are subject to the same failure mechanism will be adjusted to an appropriate interval. All components located in a penetration of a failed component will be evaluated for placement in the same interval as the failed component.

A portion of the components that are on 5 or 10 year intervals will be scheduled for testing each outage to assist in identifying common mode failures. This establishes what amounts to a sampling program for these valves. This staggered testing will help ensure that problems associated with valves of similar design, age or usage are identified on a reasonable frequency. Failure of a valve will prompt a review for generic implications.

An as-found Type B/C test, as appropriate will be performed prior to any maintenance or modification activity performed on a component if the activity could affect the component's leak tightness. Components remaining on 2 year intervals will not require as-found testing during outages during which a Type A test is not performed.

Each maintenance or modification activity that could affect the component's leak tightness is followed by a Type B/C test after the completion of the activity. If the post-work Type B/C test leakage rate for extended interval components was not > +5% of the Type B/C test leakage rate performed prior to the maintenance or modification, and other applicable retests (such as tests required for the Motor Operated Valve Testing Program) are acceptable, re-establishment of component performance will not be required and the component will remain on its current test interval. If the post-work Type B/C test leakage rate for extended interval components was > +5% of the Type B/C test leakage rate performed prior to the maintenance or modification, or other applicable retests were unacceptable, re-establishment of component performance is required and the test interval for the component will be adjusted to a 2 year interval. The test interval may be extended once satisfactory performance is re-established in accordance with the requirements of this program.

3.3 Type B and C Testing History and Preliminary Component Interval Selection

The following provides the testing history for Type B & C tests from Refueling Outage 1 (RFO1) through RFO5. The Type B/C test failures referred to in this Section were Type B/C tests that exceeded the owner's allowable leakage rates. The total number of components tested from RF01 to RF05 has varied. This variance was due to evaluations performed to determine which components are required to be Type B/C tested.

1. The Type B testing success rate from RFO1 through RFO5 is 95%. Of the 92 Type B tested components, 74 (80%) have never failed a Type B test. Of the 18 Type B tested components that have failed at least once, 16 were guard pipe inspection ports.

   The following provides a list by outage of the total Type B tests, failures and percentage passes.

TOTAL TYPE B TEST

REFUELING OUTAGE#	TOTAL COMPONENTS TESTED	TOTAL FAILURES	PERCENT PASSES
RFO1	96	17	82%
RFO2	96	0	100%
RFO3	100	2	98%
RFO4	98	6	94%
RFO5	92	0	100%
TOTAL	482	25	95%

2. The Type C testing success rate from RFO1 through RFO5 is 97%. Of the 297 Type C tested components, 255 (86%) have never failed a Type C test. The following provides a list by outage of the total Type C tests, failures and percentage passes.

TOTAL TYPE C TEST

REFUELING OUTAGE #	TOTAL COMPONENTS TESTED	TOTAL FAILURES	PERCENT PASSES
RFO1	301	13	96%
RFO2	326	8	98%
RFO3	316	16	95%
RFO4	326	9	97%
RFO5	297	6	98%
TOTAL	1,566	52	97%

The Type B & C tested components have been reviewed for preliminary interval assignments based on the program described in Section 3.2 The total number of components assigned to each interval are listed below.

Interval Years

	Fixed 2	2	5	10	Total
Number of Component	12	70	25	276	383

Some components have been selected to remain on fixed 2 year intervals because they are relatively poor performers and are the major contributors to current containment leakage. These are the main steam isolation valves and feedwater valves listed below. These valves currently contribute to 73% of the containment minimum pathway leakage and 48% of maximum pathway leakage. These 12 components are assigned 48% of the total owner's allowable leakage rate. A change to the fixed 2 year interval will only be done if evaluation under of 10CFR50.59 determines the change to be acceptable from a risk perspective, which includes both leakage probability and consequences considerations discussed later in this submittal.

FIXED INTERVAL COMPONENTS

COMPONENT NO.	PENETRATION NO.	SYSTEM	SIZE
B21F022A	005	Main Steam	28"
B21F028A	005	Main Steam	28"
B21F022B	006	Main Steam	28"
B21F028B	006	Main Steam	28"
B21F022C	007	Main Steam	28"
B21F028C	007	Main Steam	28"
B21F022D	008	Main Steam	28"
B21F028D	008	Main Steam	28"
B21F010A	009	Feedwater	24"
B21F032A	009	Feedwater	24"
B21F065A	009	Feedwater	24"
B21F010B	010	Feedwater	24"
B21F032B	010	Feedwater	24"
B21F065B	010	Feedwater	24"

The following provides examples of components that passed 3 consecutive tests which placed them in a 10 year interval but were adjusted to a 2 or 5-year interval after the component was evaluated to the requirements of Section 3.2.

1. Components with test results which were erratic.

COMPONENT NUMBER	PENETRATION NUMBER	SYSTEM	VALVE TYPE	VALVE SIZE
1E12F044A	020	RHR A	GATE	4"
1P41F169A	089	SSW A	CHECK	2"
1P41F169B	092	SSW B	CHECK	2"

2. Components with test results which were indicative of a degrading trend.

COMPONENT NUMBER	PENETRATION NUMBER	SYSTEM	VALVE TYPE	VALVE SIZE
1D23F593	109B	DRYWELL MONITORING	GLOBE	3/4"
1E12F028A	020	RHR A	GATE	18"
1E12F028B	021	RHR B	GATE	18"
1E22F004	026	HPCS	GATE	12"
1G33F028	043	RWCU	GATE	4"
1G33F034	043	RWCU	GATE	4"
1G41F053	054	FPC&CU	GATE	12"

3. Components associated with a generic type failure.

COMPONENT NUMBER	PENETRATION NUMBER	SYSTEM	VALVE/ COMPONENT TYPE	VALVE SIZE
1B21F022B	006	MAIN STEAM	GLOBE	28"
1B21F028A	005	MAIN STEAM	GLOBE	28"
1B21F028B	006	MAIN STEAM	GLOBE	28"
1B21F028C	007	MAIN STEAM	GLOBE	28"
1B21F032A	009	FEEDWATER	CHECK	24"
1B21F032B	010	FEEDWATER	CHECK	24"
1E21F006	031	LPCS	CHECK	14"
1E22F005	026	HPCS	CHECK	14"

4. Components located in a penetration that had a failed component.

COMPONENT NUMBER	PENETRATION NUMBER	SYSTEM	VALVE TYPE	VALVE SIZE
1P52F122	041	SERVICE AIR	CHECK	3"
1G36F106	049	RWCU FILTER/DEMIN	GATE	4"
1P53F003	070	INSTRUMENT AIR	GLOBE	1"
1P60F009	085	SUPPRESSION POOL CLEANUP	GATE	12"
1G33F252	087	REACTOR WATER CLEANUP	GATE	4"
1G33F004	087	REACTOR WATER CLEANUP	GATE	4"

3.4 Basis for Proposed Changes

Modification of the Type B & C test intervals for GGNS is based on continued leak tightness of containment components supported by the following factors:

- Performance-based programs, as discussed in Section 2, provide sufficient rigor and flexibility to ensure continued levels of high performance while allowing a rational allocation of limited resources
- In particular, the GGNS program is comprehensive in considering relevant aspects of component performance (e.g., industry operating experience) which go well beyond mere test results
- The impact of the proposed change is safety neutral as demonstrated by the risk assessment in Section 3.5
- GGNS has demonstrated competence in the implementation of safety significant programs which contribute to strong plant performance while maintaining safety as a primary objective
- Containment component performance is a function not only of Appendix J testing but a multitude of overlapping (and sometimes redundant) practices and programs; at GGNS, the following factors serve to ensure adequate containment performance:

    1) The overall allowable containment leakage rate, L_a, for GGNS is 211,600 standard cubic centimeters per minute (sccm). The Technical Specification 3.6.1.2.b allowable Type B and C test leakage rate is 0.60 La, or 126,900 sccm. Calculated

by maximum pathway leakage techniques, the GGNS current maximum pathway Type B and C test leakage rate for the containment, as of 07/12/93, is 26,179 sccm which is 12% of L_a. This low leakage rate can be attributed to low owner's allowable leakage rates for Type B and C tests, an effective maintenance program and the other programs discussed below.

2) The owner's allowable leakage rate assigned to each component is conservative and was chosen to signal the possible onset of component degradation. Owner's allowable leakage rates for valves tested with air are currently based on the nominal pipe size of the valve times 260 sccm, unless it is specified in other documents (i.e. Technical Specification for Main Steam Isolation Valves). Assignment of allowable leakage rate proportional to the size of the valve is chosen to be small enough to signal that the valve leakage tightness is beginning to degrade. For example, a 1" diameter valve has a leakage limit of 260 sccm and a 10" diameter valve has a leakage limit of 2600 sccm. This method provides a rational method for assigning leakage rates and is conservative when compared to other plants surveyed in the industry. Because of this conservation in the owner's allowable leakage rate calculation, a failure of the owner allowable leakage is not a failure of the Technical Specification Requirement. In fact, the assigned owner's Allowable Leakage provides a substantial margin of safety below the .6 L_a leakage limit specified in the Technical Specification.

3) The components selected for fixed 2 year intervals are components proven to be poor performers and are the major contributors to current containment leakage. The main steam isolation valves and feedwater check valves are fixed 2 year interval components. These valves currently contribute 73% of the containment minimum pathway leakage and 48% of maximum pathway leakage. These 12 components are assigned 48% of the total owner's allowable leakage rate.

4) The Motor Operated Valve Testing Program, in accordance with Generic Letter 89-10, requires monitoring of valve/actuator key performance parameters while at the same time establishes the correct control switch settings to provide additional assurance of valve operational readiness. This program includes requirements to periodically verify operational readiness. This program would assist in predicting potential failures that could contribute to valve leakage, such as a low stem thrust setting which could indicate potential valve seat leakage.

5) American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, (ASME Code) Section XI, Paragraph IWV-3400 requires Category A and C valves to be exercised every 3 months or during cold shutdowns. Category A valves are defined as valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their function. This program assists in identifying failures that could cause valve leakage, such as excessively long stroke times and failures to close fully.

6) Most of the check valves in the Type C test program are also in the inspection program implemented for INPO's Significant Operating Experience Report 86-03. This program requires visual inspection of internal surfaces and parts, or testing using non-intrusive equipment at established intervals, to determine and calculate expected wear rates for check valve internals. This program assists in predicting potential failures that could contribute to valve leakage.

7) Various system surveillances are performed periodically that would identify system problems. These problems could be associated with containment isolation component failures, such as a containment isolation valve failing to close at system pressure.

8) Routine preventive maintenance is performed on components in accordance with the preventive maintenance program and includes inspection of electrical components, meggering of motors, inspection of air actuators for leakage and lubrication of actuators. This program helps to identify problems that could contribute to leakage of containment isolation components and helps to keep the components in fully functional condition.

9) Each valve that is Type C tested is pressure tested and inspected for leakage in accordance with ASME Code Section XI, Paragraph IWA-5211. Class 1 components are pressure tested every refueling outage per Table IWB-2500 and Class 2 components are pressure tested per Table IWC-2500 every 40 months. These inspections detect any external leakage from components, such as through-wall pressure boundary leaks, leaks from mechanical joints including bonnet to body leaks, and packing leaks. External leakage could be an indicator of containment isolation component leakage.

10) Each valve that is Type C tested and is in a system that could contain highly radioactive fluid during an accident is also included in the Leakage Reduction Program, which is required by NUREG 0737. This program requires a walkdown to identify leakage at least once a refueling cycle with the respective system in operation or otherwise pressurized. These walkdowns detect any external leakage from components, such as through-wall pressure boundary leaks and leaks from mechanical joints, including bonnet to body leaks and packing leaks.

11) Accessible containment isolation valves that are not capable of being closed by operable automatic containment isolation valves and are required to be closed during accident conditions are verified to be secured in the closed position monthly during power operation. The verification also includes accessible blind flanges in the containment isolation boundary. Manual containment isolation valves and other containment isolation barriers which are not accessible during power operation are verified to be secured in their isolation positions before reactor startup. These verifications will help to identify any containment penetration abnormalities that could indicate a containment leakage path.

12) The containment penetrations are constructed to ASME Code Section III Class 1 or Class 2 requirements and are examined in accordance with the requirements of ASME Code Section XI.

13) Nuclear industry operating experience reports, such as NRC Information Notices, Nuclear Network Operating Experience Reports and INPO Significant Operating Experience Reports, are examined to determine if problems exist that could affect the performance of similar components at GGNS. The Nuclear Safety and Regulatory Affairs Department's Operating Experience Group reviews and evaluates the operating experience documentation and determines if a problem is applicable to GGNS. If applicable, the documents are distributed to the appropriate GGNS departments and evaluated. An example is NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing", concerning Quad Cities Station's problem with containment penetration bellows. The identification of this problem prompted GGNS to evaluate current testing methods and adjust test methods appropriately.

14) GGNS is a pilot plant for implementation of the new Maintenance Rule. Systems will be monitored under the Maintenance Rule for performance of proper root cause determinations and corrective actions due to Maintenance Preventable Functional Failures (MPFFs). Attention will be focused on those systems which do not meet performance criteria which require 10CFR50.65(a)(1) goal setting monitoring. The function of containment integrity is placed in one system at GGNS. Performance criteria for this system will be no repetitive MPFF. This would focus attention on a particular component or generic grouping of components which have failed repetitively and have not been properly corrected.

15) System Engineers perform weekly system walkdowns, monitor system surveillances, review condition identifications reports written against components and provide corrective action for component failures. The System Engineer provides a single focal point for a system, which allows a concentrated and dedicated review of a system's performance parameters and failures and should help to identify containment isolation component problems.

16) Relief valves are set-pressure tested and seat-leakage tested in accordance with ASME Code Section XI. The valves are tested at least once every 5 years. This program would confirm relief valve operational readiness and/or identify relief valve seat leakage. Seat leakage would be a direct indicator of containment penetration leakage.

17) A root cause evaluation is performed for each failure identified as significant per the GGNS nonconformance program. The root cause evaluation is performed to identify the true cause for the failure so appropriate corrective actions can be performed to avoid the failure in the future. Root cause evaluations performed for Type B/C test failures will contribute to appropriate corrective actions and should increase reliability for the containment isolation component.

18) GGNS' proposed program (see Section 3.2) requires a portion of the components that are on 5 and 10 year intervals to be scheduled for testing each outage. This will allow the components on 5 or 10 year intervals to be monitored for performance, to identify generic problems and to ease outage impact by not testing all components during the same outage.

19) Reducing the amount of testing will reduce the manipulation of valves and components in systems to support testing. Reduced testing will also reduce the possibility for human error when restoring systems to operational configuration and will increase system availability, thereby reducing risk.

20) Maintenance or modification that could affect the leak tightness of a component is followed by Type B/C testing prior to declaring the component operable.

21) All components located in a penetration of a failed component will be evaluated for placement in the same interval as the failed component. Testing all components in a penetration that contained a component that failed will provide added assurance that the penetration will perform its intended function.

3.5 Risk Impact Assessment

3.5.1 Summary

This section summarizes an assessment of the potential risk impact of the proposed alternatives to the Appendix J requirements. The bases for this assessment are existing GGNS-specific risk studies for both normal power and shutdown configurations. This assessment includes the identification of possible impacts of the proposed alternatives on individual component performance, identification of the resulting risk impact on a component-by-component basis, and quantitative estimates of the change in risk, where possible.

The following summarizes the results of this assessment.

- The proposed alternatives do not result in any new potential accident sequences, but do require the reassessment of some accident sequences which had been previously demonstrated to be of negligible frequency. These sequences are shown to still be of negligible frequency.

- An increase in the probability for containment penetration leakage is possible. This is partially offset by a potential decrease in component unavailability due to testing and post-test restoration errors.

- The resulting impact on overall plant risk includes both positive and negative elements. The principal negative impact is the increased potential for containment bypass/isolation failure. The principal positive impact is a reduction in shutdown risk.

- Both the positive and negative risk impacts are small, and well within the uncertainty bands of the present risk analyses. *The overall risk impact of the proposed alternatives is neutral and essentially negligible.*

3.5.2 Identification of Potential Risk Impacts

In order to assess the potential risk impact of the proposed alternatives to Appendix J requirements, available GGNS safety analyses and risk assessments were reviewed. This effort focused on estimating the incremental risk associated with the proposed alternatives within the context of the existing analyses. The principal source for evaluating this incremental risk was the GGNS Individual Plant Examination (IPE), Reference 1, and its supporting calculations and notebooks. In addition, a Nuclear Safety Analysis Center (NSAC) assessment of BWR risk during shutdown, Reference 2, was also reviewed. This NSAC study, which developed a shutdown PRA model, used GGNS as its reference plant.

The potential effects of the proposed alternatives on overall risk include effects on the frequency of core vulnerable events, as well as containment response to those events. From the standpoint of core vulnerable event frequencies, the proposed alternatives may affect: (1) initiating event frequencies, (2) mitigation system availabilities, and (3) shutdown risk. The principal impacts on containment performance are: (4) the probability for containment isolation failure, given a challenge to containment, and (5) the probability for containment bypass. The identified potential impacts of the proposed alternatives in each of the above five areas are described qualitatively below. Note that the proposed alternatives have both positive and negative effects on overall risk. Section 3.5.3 provides an assessment of their quantitative impacts.

1. <u>Initiating Event Frequencies</u>

The proposed changes do not introduce any new accident initiators, since there is no change in plant configuration or testing method. The changes may, however, affect the frequencies of certain accident initiators. Of the initiating events considered in the GGNS IPE, two were identified as possibly being impacted. One of these is the interfacing system LOCA, which may be caused by gross leakage through isolation valves separating the high pressure reactor coolant system from interconnected low pressure systems. The leakage must be sufficient to cause a rupture of the low pressure piping or failure of pipe connections or component seals. The proposed alternatives to Appendix J requirements could increase the time between isolation valve tests. This may contribute to a higher probability for leakage being undetected between tests.

The second initiating event which was identified as possibly being impacted by the proposed alternatives is a LOCA outside containment. This is the result of a pipe break outside containment with failure of the isolation valves to isolate the reactor coolant system from the break. While the proposed changes do not affect the failure probability for the isolation valves to close on demand, they may increase the probability for leakage following valve closure.

2. <u>Mitigation System Availability</u>

The proposed Appendix J program changes could impact the availability of systems required to respond to plant upset conditions. Potential effects include the following:

- Isolation of non-essential loads under accident conditions may require the operation of containment isolation valves. Failure to isolate such loads could violate system success criteria used in the risk assessment. The review of the IPE uncovered no such

conditions. While there are requirements for isolating non-essential loads under certain upset conditions (e.g. the isolation of the non-regenerative heat exchanger from the component cooling water system following a loss of electrical power), the required isolation functions are achieved by valves other than the containment isolation valves. No impact on risk due to this effect was identified.

- Mitigation system success criteria also include requirements that flow not be significantly diverted from the intended flow path. As a general rule, the IPE assumed that a diversion pathway needed to be at least 1/3 the diameter of the main flow path piping. If the proposed Appendix J program changes result in an increased probability for gross leakage, then there could be a detrimental effect in terms of increasing the potential for flow diversion within mitigation systems.

- Failure of isolation valves to close on demand, or valve leakage after closure, could result in an uncontrolled reactor coolant system depressurization. This could subsequently result in the unavailability of steam for the RCIC system and loss of its coolant makeup capability. This would apply to the MSIVs and feedwater isolation valves. However, the failure of these valves is not affected by the proposed changes, since the MSIVs and feedwater isolation valves are maintained on a 2 year fixed frequency.

- Post-test valve restoration errors could lead to: misaligned valves, disabled equipment (i.e., equipment that would not perform its required function on demand), or piping voids which could contribute to a water hammer failure. By conducting the Appendix J Type C tests less frequently, there are fewer opportunities for such errors. The detection of such errors could be delayed, however, if the error would not be observed until the next LLRT. If the error would be uncovered via other system/component tests, then the effect of a reduced test frequency is to reduce the component/system unavailability due to post-test restoration errors.

3. Shutdown Risk

During a Type C test, one or more trains of equipment may be made unavailable. While these train outages are carefully scheduled to maintain adequate redundancy of safety systems during shutdown, the tests do contribute to overall system unavailability. Tests associated with RHR trains A or B may contribute to a loss-of-decay-heat-removal initiator. Other Type C tests may make unavailable trains of makeup water which are required for mitigation of a draindown event during shutdown. The effect of reduced testing is to reduce the testing contribution to component and system unavailability during shutdown.

Another potential risk impact is the possibility of post-test restoration errors contributing to a draindown event during shutdown. This may be the result of failure to restore valves to the closed position following a test. Leakage or inadvertent operation of a second valve in the line isolating the RCS from the draindown pathway, could result in diverting reactor coolant. Such an event would result in a challenge to plant makeup systems and could contribute to core uncovery during shutdown.

4. <u>Containment Isolation Failure</u>

Failure of containment isolation could lead to release of radionuclides from containment under certain accident conditions. As discussed above, one impact of the proposed changes is to potentially increase the probability for containment isolation valve leakage following a release into the containment atmosphere.

5. <u>Containment Bypass</u>

Containment bypass is the creation of a direct pathway from the reactor coolant system to areas outside containment. Interfacing LOCAs, discussed above, are containment bypass pathways. Consequences of a bypass include the direct release of reactor coolant outside containment, and in the case of a severe accident, the direct release of fission products from damaged fuel. The effect of the proposed changes is the same as described for containment isolation failure.

A thorough examination of each containment penetration was performed to identify those penetrations and components which could affect risk in one or more of the above five areas. This initial assessment was qualitative and was intended only to identify potential impacts. Table 3-1 presents results of this examination. Noted for each penetration is the impact of the penetration failure in creating either a bypass or isolation failure condition. Also noted for each component are the identified effects of a reduced Appendix J Type C test frequency in contributing to an initiating event, shutdown risk, and mitigation system unavailability. Penetration or component size was a factor in this qualitative evaluation, with the assumption that leakage through smaller valves would not be sufficient to cause the undesired system degradation, such as an uncontrolled RCS depressurization, flow diversion, etc. Section 3.5.3 of this report assesses the actual positive and negative effects of each identified item on overall risk.

3.5.3 Assessment of Risk Impact

3.5.3.1 Impact of Test Frequency on Valve Performance

In order to assess the impact of the proposed Appendix J program changes on risk, it is necessary to first estimate their impact on individual valve performance. Potential valve failure modes of interest are:

- internal valve leakage
- failure of a valve to open/close on demand
- valve unavailability due to post-test restoration error
- valve unavailability due to testing and maintenance.

Internal Valve Leakage

Internal valve leakage is modeled in this assessment with a constant failure rate model. The failure rates used are listed in Table 3-2. Immediately following a successful Type C test, an isolation valve is assumed to not be leaking. Over time, the probability that the valve develops leakage in excess of acceptance criteria increases approximately linearly with time, reaching a maximum probability at the time of the next Type C test. (Note that this near-linear relationship applies to the probability of leakage, not the magnitude of the leakage.) Since an accident challenging containment may occur at any time between tests, the mean probability that a valve is leaking at the time of a challenge is proportional to the mean time between tests. The proposed testing program changes include a reduction in the frequency for Type B and C tests from a minimum frequency of once per two years to a minimum frequency of once per ten years, based on prior component performance. This decrease in test frequency results in a maximum increase in the leakage probability by a factor of five.

Note that this assessment does not consider: (i) that the components which are allowed to go on the extended testing schedule have demonstrated lower failure rates, (ii) potential effects of component aging which could lead to an accelerated failure rate, or (iii) potential effects of higher failure rates early in component life. The neglect of the first effect results in a conservative estimate of the potential risk impact. The latter two effects may result in failure rates which are not constant over time. The effects of aging are addressed through the planned staggered testing schedule. Aging related failures will be detected through tests conducted on similar components. Early-in-life failures are typically due to manufacturing defects or installation problems. The proposed testing plan would effectively detect such problems, since the current test program (each refueling outage) would apply until the newly installed component successfully passed twc successive leakage tests. The maximum test interval would not be allowed until it successfully passed three successive tests. The constant failure rate model is therefore most applicable for this analysis.

Valve Demand Failures

It is assumed that the proposed modifications to the Appendix J testing program will not impact the failure probabilities for demand related failure modes, e.g., valve fails to open/close on demand. (Note that possible leakage after valve closure is a separate failure mode and is treated as described above.)

Possible effects of the proposed changes include: (i) an increase in on-demand failure probability due to failure mechanisms which could occur over time while the component is in its "normal" standby state, and (ii) a decrease in failure probability due to decreased stress on the component by elimination of component tests. The change in the testing frequency will not have a significant effect on failures occurring between component demands, since valves which must change state to perform their intended functions are cycled as part of other component/system functional tests (see, for instance ,Section 3.4). The Appendix J tests are not intended to demonstrate the functionality of the component in changing state on demand. Similarly, the Appendix J tests do not place a significant number of additional demands on the component. Therefore, the effect of reducing demand-related stress on the component is considered to be negligible.

Post-Test Human Errors

Post-LLRT restoration errors could result in a train of equipment being unavailable when the reactor returns to power following a refueling outage. LLRTs associated with one train of equipment are performed during a scheduled outage of that train. During the LLRT, the train may be tagged out and power removed from the pump, if applicable. Post-LLRT restoration errors could include: failure to restore power to the pump, failure to reposition valves to their normal positions, and failure to properly fill and vent the system piping. These are generally combined together into a single human error event for a train of equipment.

Following the LLRT, a functional test of the train is performed. There is also an independent written verification of the train restoration lineup following the test, and a pre-startup system walkdown to confirm the valve lineup. In addition, many of the systems of interest to this study have trouble alarms, out-of-service alarms or valve position indication in the control room which provide another means to detect post-LLRT restoration errors.

The methodology of the GGNS IPE was used to estimate the probability for failure to restore a train of equipment following the LLRT. The IPE included a general post-test/maintenance train restoration error associated with the RHR, RCIC, HPCS and LPCS systems. This error probability (3.0E-3) was based on a basic human error probability of 3.0E-2 with a recovery factor of 1.0E-1 for post-test verification activities. This general train restoration error probability is assigned to the post-LLRT errors, as well. In addition, a recovery factor of 1.0E-2 is applied for system functional tests performed after the LLRT.

The present assessment also considers a specific failure to restore manual valves in the RHR, LPCS, and HPCS injection lines. These potential errors are considered separately, since they would not be detected through normal system functional tests. A basic error probability of 1.0E-3 is assigned, which includes a recovery factor of 1.0E-1 for post-test verification activities. An additional recovery factor of 1.0E-1 is applied for the independent prestartup valve verification walkdown. No additional credit is taken for valve position indication in the control room.

Post-test restoration errors which have gone undetected at startup, may be detected during the next scheduled functional test of the system. Therefore the component unavailability due to post-LLRT restoration errors can be calculated as:

$$U_{LLRT} = P_{LLRT} * \frac{t}{T}$$

where P_{LLRT} is the probability for failure to restore the train/component following the test, t is the mean time between functional tests, and T is the mean time between LLRTs. Table 3-3 lists the post-LLRT errors which are considered in this study and the bases for their quantification under the current testing schedule and the proposed testing schedule. The effect of the proposed changes is to reduce the unavailability due to post-test restoration errors.

Unavailability Due to Component Testing

The final effect is due to component unavailability during the Appendix J test itself. This unavailability is directly proportional to the testing frequency. The impact of the proposed changes is therefore to reduce this unavailability by as much as a factor of five.

3.5.3.2 Impact on Plant Risk

The impact on overall plant risk is assessed by applying the component failure models described above to the areas of potential impact identified in Section 3.5.2.

1. Initiating Event Frequencies

Interfacing System LOCA

All containment penetrations with the potential for creating an interfacing system LOCA consist of at least two normally closed isolation valves. The initiating event occurs when one of the valves develops a significant leak, while the remaining valve(s) have pre-existing undetected leakage. Interfacing LOCAs are considered for the LPCI, LPCS, HPCS and RCIC systems. In all cases, the isolation valves are water tested for leakage to ensure the integrity of the high pressure/low pressure system boundary in accordance with GGNS Technical Specification 4.4.3.2.2. The proposed changes to the testing requirements of Appendix J do not include any changes to this section of the GGNS Technical Specifications. Therefore, the proposed changes do not alter the frequency for interfacing system LOCAs.

LOCA Outside of Containment

The main steam, feedwater and RWCU systems contain piping outside containment which creates the potential for a LOCA outside containment. Isolation valves close to prevent the loss of reactor coolant through potential breaks in these systems. A LOCA outside containment due to a failure within the RWCU system is considered to be very remote. This is due to the fact that most of the RWCU equipment is located inside containment and that there are a number of check valves and remotely operated valves which could isolate the system should the normal isolation valves fail or leak. The assessment of LOCAs outside containment is therefore limited to the main steam and feedwater systems.

Failure of the main steam and feedwater isolation valves to close on demand is not expected to be impacted by the proposed changes. However, the possibility for leakage after closure could be affected. Changes to the testing frequency for these isolation valves are not anticipated. However, in order to estimate the maximum possible impact, it is assumed here that the test frequency for these valves is decreased to once every ten years.

The probability for failure of these isolation valves is calculated to be very small in the IPE (2.7E-5 per line for the MSIVs and negligible for the feedwater isolation valves). This is dominated by the probability of the MSIVs failing to close on demand. With a potential five-fold increase in the probability of valve leakage after closing, the probability for failure to isolate needs to include the leakage probability following valve closure. The overall isolation failure probability for a single valve i is therefore calculated as

$$P_{I,i} \cong P_D + \lambda \frac{T}{2}$$

where P_D is the probability for failure to close on demand, T is the mean time between LLRTs and λ is the failure rate due to leakage. Figure 3-1 shows the valve arrangement for an MSIV penetration. Also shown is the calculated isolation failure probability for the penetration. The total probability for failure to isolate the MSIVs (any one of four lines fails to isolate) is 1.2E-2. Similarly, Figure 3-2 shows the valve arrangement and calculated isolation failure probability for a feedwater penetration. The total probability for failure to isolate the feedwater penetrations (either of two lines) is 8.0E-6.

These probabilities must be coupled with the frequency of a pipe break outside containment (estimated to be 1E-3/year for steamlines and 1E-3/year for feedwater lines) and the conditional probability of core melt given a LOCA. Since the results are dominated by leakage, the resulting LOCA will have the characteristics of a small LOCA (S2 initiator in the GGNS IPE) or a small-small LOCA (S3 initiator in the IPE), depending on the leakage rate. The conditional probability of core damage given a small LOCA is 8.4E-6. For a small-small LOCA it is 5.3E-7. In order to bound the core damage impact of the proposed changes, it is assumed here that the leak results in a small LOCA initiator. The core damage frequency introduced by the assumed increase in isolation valve leakage probability is therefore only 1.0E-10. This increase is extremely small and represents an insignificant increase in the overall core damage frequency due to LOCAs (3.73E-7/year).

2. Mitigation System Unavailability

Flow Diversion

Table 3-1 identifies potential flow diversion pathways in the LPCI, LPCS, HPCS and RCIC systems. A normally closed motor operated valve in each pathway prevents flow from being diverted from the intended injection path. Leakage through the valve would not create sufficient flow diversion to fail the injection function. Rather, catastrophic failure of the valve would be required. Such failure is extremely unlikely and has never been observed at GGNS. Since a similar effect can result from post-maintenance or post-test restoration errors which are already considered in the IPE, the probability of catastrophic failure of the valves is negligible and not risk significant.

Gross leakage through system relief valves may also cause flow diversion from the LPCI system. This failure mode is detectable during normal system surveillance tests. Therefore, the proposed changes do not impact the probability for system degradation due to flow diversion.

Uncontrolled Depressurization

Failure of the MSIVs could cause an uncontrolled depressurization of the reactor coolant system resulting in the unavailability of makeup capability from the RCIC system. It is highly unlikely that the MSIVs will fail to close on demand. This is due to the high reliability of the valves and the redundancy offered by two MSIVs in each steam line. Furthermore, the proposed changes will not affect the MSIV testing frequency (see Section 3.3).

The probability of leakage through the MSIVs could be affected by the LLRT frequency. However, leakage through the valves would not result in an uncontrolled system depressurization. The resulting transient would consist of a very slow depressurization. The RCIC system would be available to help mitigate the effects of such a transient.

Post-Test Restoration Errors

Post-test restoration errors may impact the availability of equipment in the RHR, LPCS, HPCS and RCIC systems. This in turn affects the availability of these systems to perform their intended functions during an accident. In the case of the RHR system, the LPCI, suppression pool cooling and containment spray functions have potential risk impact and may be affected by post-test restoration errors.

The effect of post-test restoration errors on risk can be estimated through the results of the GGNS IPE. The IPE fault trees contain basic events representing the unavailability of trains of equipment due to restoration errors following test or maintenance. The "importance" of these events is used to estimate the impact of the post-LLRT train restoration errors. The importance measure used in this analysis is the fractional contribution to the core damage frequency which is due to the event. Using the available importance measures for the modeled train restoration faults, the impact of the change in LLRT test frequency can be calculated as

$$\Delta\,(\text{Frequency}) \cong \frac{\Delta\,(U_{LLRT})}{P_{Restoration}} * I_{FV} * \text{Frequency}_{CM}$$

where I_{FV} is the Fussell-Vesely importance measure for the event and $P_{Restoration}$ is the post-test/maintenance restoration error probability from the IPE (3.0E-3).

Table 3-4 lists the importance measures for each system of interest. Also shown is the calculation of the risk impact of the proposed changes using the unavailabilities due to post-LLRT restoration errors from Table 3-3. As can be seen from this table, the effect of a reduced test frequency is a slight reduction in risk due to potential post-test restoration errors. Almost all of the effect is due to the reduction in the post-test restoration error probability for the HPCS system. This is due to the relatively greater importance of the HPCS system in the GGNS IPE, as illustrated by the Fussell-Vesely importance measure. Due to the number and effectiveness of measures to prevent and detect post-LLRT restoration errors, these errors are very low in probability. Therefore the risk reduction effect of further reducing the probability of such errors is very small.

3. Shutdown Risk

GGNS has an Outage Risk Assessment and Management (ORAM) program to help in planning and controlling outage activities. The plant risk model, which is part of this program, was developed as part of an NSAC investigation of shutdown risk. The program allows outage managers to develop risk profiles of the outage. As equipment is removed from service for testing or maintenance, the risk profile changes to reflect the current plant configuration. Risk is measured primarily in terms of core damage frequency and RCS boiling frequency in Modes 4 and 5.

Input to the ORAM program includes the schedule of outage activities. In order to investigate the potential risk impact of the Appendix J LLRTs, sensitivity studies were performed using the outage schedule for the upcoming RF06 outage. Potential risk impact was defined for LLRTs involving the RHR trains, LPCS, HPCS and the RHR shutdown cooling common suction line. These LLRTs are normally performed near the end of scheduled maintenance activities associated with this equipment. The LLRT typically extends the equipment unavailability by an additional 16 to 48 hours. The effect on overall risk during this outage is shown in Table 3-5.

The overall impact of eliminating the equipment unavailability due to LLRTs in this outage is a slight reduction in the core damage risk and a somewhat larger reduction in the RCS boiling risk. Most of the core damage risk reduction is due to elimination of the LPCS LLRT. This is due to the fact that LPCS is removed from service fairly early in RF06 when decay heat levels are relatively high. Most of the RCS boiling risk reduction is due to the elimination of the RHR train A LLRT. Again, RHR train A is removed from service earlier than train B during RF06; therefore, its risk impact is greater.

Another potential risk impact during shutdown is the possibility that an LLRT could contribute to an RCS draindown event. It is postulated that a valve may not be closed following an LLRT and that a second valve in a line connected to the RCS may be inadvertently opened. RCS coolant could be diverted through the open line, requiring makeup to the RCS to avoid the potential uncovery of fuel rods and/or interruption of normal decay heat removal. This risk scenario has not been quantified.

4. Containment Isolation

Containment isolation is required to prevent the release of radionuclides should the containment be challenged, such as in a LOCA or severe accident. The GGNS IPE determined that failure of containment isolation during a severe accident was negligible (i.e., the frequency of such accident sequences was below the reporting guidelines for the IPE). The decrease in isolation valve test frequency could contribute to a higher probability of containment isolation failure, which could negate this previous finding.

Many of the containment penetrations are not expected to be isolated under post-accident conditions since they are part of active systems, are effectively sealed with water thereby preventing the release of radionuclides, or are normally open small instrumentation lines.

Each containment penetration, with the exception of those located below the suppression pool water level, is isolated by at least two independent containment isolation barriers. Those penetrations which have only a single isolation valve meet all of the following design criteria:

- they connect directly to the suppression pool,

- they are provided with a single isolation valve,

- they are submerged thereby preventing the escape of containment atmosphere, and

- the piping outside containment constitutes a closed system providing a second isolation barrier following a single active failure.

As a result, these penetrations will always have a water seal and cannot act as a release pathway unless the integrity of the connected system is compromised. Such an occurrence, coincident with another accident condition, is judged to be incredible. This conclusion is unaffected by the proposed changes. Therefore, the only penetrations of interest from a containment isolation standpoint are those which are protected by at least two valves.

The IPE considers the random independent failures of two valves to isolate due to mechanical faults and/or support system failures. The maximum probability for isolation failure is shown to be less than 1.0E-3. This is dominated by isolation valves failing to close on demand. The resultant release pathway has a nominal full-open penetration flow area.

While the probability of valve leakage after closing may increase with the proposed testing changes, such leakage represents a much smaller effective flow area and therefore a much smaller potential source term. In addition, the relatively restricted flow path through a leaking valve would result in higher decontamination factors, thus further reducing the source term. As a result, releases due to containment isolation failure are still expected to be dominated by failures of valves to close on demand. Since this failure probability is unaffected by the proposed changes, the proposed changes are expected to have little impact on the source term for containment isolation accident sequences. The GGNS IPE demonstrates that such sequences have little risk significance. The proposed changes do not alter this conclusion.

5. <u>Containment Bypass</u>

Containment bypass is addressed in a manner similar to containment isolation. Again, the IPE demonstrates that accident sequences with containment bypass are negligible contributors to overall risk. Many of the lines identified as having the potential for containment bypass are screened out in the IPE based on the fact that:

- they are small instrumentation lines,
- they are isolated by passive check valves with high reliability, or
- they are passive electrical penetrations.

The remaining penetrations with the potential for bypass are analyzed and are generally found to be dominated by probabilities associated with valves failing to close on demand following the accident.

Table 3-6 lists the penetrations with the potential for containment bypass. Also shown are the IPE analysis results for each penetration and an estimate of the bypass probability with the proposed Appendix J program changes. The proposed changes introduce a greater probability for valve leakage. As a result many of the penetrations previously dominated by demand failures now have significant contributions due to leakage after valve closure. This is particularly true for the MSIVs and feedwater isolation valves. The estimated effect of the proposed changes is to increase the bypass failure probability from below 1E-3 to approximately 1.2E-2. Most of this change is due to the contributions of the MSIVs. Since the testing schedule for these valves will remain on the fixed 2-year frequency (see Section 3.3), a second analysis is performed in which the incremental contributions of these valves is set to zero. For this case, the incremental increase in bypass failure probability is approximately 1.2E-4. Combined with a low frequency for containment challenges, bypass sequences would still be below the screening criteria for the IPE.

3.5.4 Conclusions

This assessment has identified potential risk impacts of the proposed alternatives to the Appendix J testing requirements. Significant assumptions and results of this assessment include the following.

- An increase in the probability for containment penetration leakage is possible, *if one neglects the fact that those valves which are allowed to be tested at less frequent intervals have already demonstrated a lower failure rate.*

- The proposed changes do not result in any new potential accident sequences, but do require the reassessment of some accident sequences which had been previously demonstrated to be of negligible frequency. These sequences are shown to still be of negligible frequency.

- There are both positive and negative impacts of the proposed changes on risk. The principal negative impact is the increased potential for containment bypass/isolation failure. The principal positive impact is a reduction in shutdown risk.

- Both the positive and negative risk impacts are small, and well within the uncertainty bands of the present risk analyses. *The overall risk impact of the proposed changes is neutral and essentially negligible.*

3.5.5 References

1. "Grand Gulf Nuclear Station Individual Plant Examination Summary Report", Entergy Operations, December 1992.

2. "Safety Assessment of BWR Risk During Shutdown Operations, EPRI Outage Risk Assessment and Management (ORAM) Program", NSAC-175L, August 1992.

3.6 Occupational Radiation Exposure Assessment

The reduction in occupational dose is difficult to assess. Although doses for personnel that directly support leakage rate testing are available, support personnel such as health physics and operations personnel are not tracked specifically for each activity. Total exposure for personnel that are tracked averages 7.124 man-rem per refueling outage for LLRTs. These exposures may be reduced as much as two thirds under the proposed performance-based program. With this assumption, the total man-rem reduction due to the proposed changes is approximately 140 man-rem over the remaining plant life.

3.7 Cost Reduction Assessment

LLRTs are labor intensive. Approximately 20,000 man hours are expended each refueling outage performing LLRTs. Using similar assumptions to those used for ILRTs (see Section 4.6), an estimated savings of $7,300,000 can be expected over the remaining licensed life for GGNS.

It should be noted that the calculation for LLRT cost reduction assumes no credit for shortening outage lengths. While difficult to quantify, performance of LLRTs may constitute the major hidden workscope that defines an outage critical path. If true, the potential savings due to reduced outage length would be will in excess of other savings cited in this submittal.

Table 3-1
Identification of Potential Risk Impact

					Potential Impact on:			
Penetration	Component	System	Type	Size	Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
001	M23Y007	Containment	O-Ring					Isolation
002	M23Y002	Containment	Airlock					Isolation
003	M23Y001	Containment	Airlock					Isolation
004	F11E015	Fuel Transfer	O-Ring					Isolation
	G41G515	Fuel Transfer	Bellows					
005	B21F022A	Main Steam	AOV	28	[6]	[5]		Bypass
	B21F028A	Main Steam	AOV	28	[6]	[5]		
	B21F025A	Main Steam	XV	0.75				
	B21F067A	Main Steam	MOV	1.5				
	E32F001A	Main Steam	MOV	1.5				
006	B21F022B	Main Steam	AOV	28	[6]	[5]		Bypass
	B21F028B	Main Steam	AOV	28	[6]	[5]		
	B21F025B	Main Steam	XV	0.75				
	B21F067B	Main Steam	MOV	1.5				
	E32F001E	Main Steam	MOV	1.5				
007	B21F022C	Main Steam	AOV	28	[6]	[5]		Bypass
	B21F028C	Main Steam	AOV	28	[6]	[5]		
	B21F025C	Main Steam	XV	0.75				
	B21F067C	Main Steam	MOV	1.5				
	E32F001J	Main Steam	MOV	1.5				
008	B21F022D	Main Steam	AOV	28	[6]	[5]		Bypass
	B21F028D	Main Steam	AOV	28	[6]	[5]		
	B21F025D	Main Steam	XV	0.75				
	B21F067D	Main Steam	MOV	1.5				
	E32F001N	Main Steam	MOV	1.5				

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
009	B21F010A	Feedwater	CV	24	[6]	[5]		Bypass
	B21F030A	Feedwater	XV	0.75				
	B21F032A	Feedwater	CV	24	[6]	[5]		
	B21F063A	Feedwater	XV	0.75				
	B21F065A	Feedwater	MOV	24	[6]	[5]		
010	B21F010B	Feedwater	CV	24	[6]	[5]		Bypass
	B21F030B	Feedwater	XV	0.75				
	B21F032B	Feedwater	CV	24	[6]	[5]		
	B21F063B	Feedwater	XV	0.75				
	B21F065B	Feedwater	MOV	24	[6]	[5]		
011	E12F004A	RHR	MOV	24		[3]	[2]	Isolation
	E12F017A	RHR	SRV	1			[2]	
012	E12F004B	RHR	MOV	24		[3]	[2]	Isolation
	E12F017B	RHR	SRV	1			[2]	
013	E12F004C	RHR	MOV	24		[3]	[4]	Isolation
	E12F017C	RHR	SRV	1			[4]	

Table 3-1
Identification of Potential Risk Impact

					Potential Impact on:			
Penetration	Component	System	Type	Size	Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
014	E12F002	RHR	XV	0.75			[4]	Bypass
	E12F008	RHR	MOV	20	Yes		[2]	
	E12F009	RHR	MOV	20	Yes		[2]	
	E12F308	RHR	CV	0.75				
017	E51F063	RCIC	MOV	10	Yes	[3]		Bypass
	E51F064	RCIC	MOV	10	Yes	[3]		
	E51F072	RCIC	XV	0.75				
	E51F076	RCIC	MOV	1				
018	E12F023	RHR	MOV	6	Yes	[1]	[4]	Bypass
	E12F061	RHR	XV	0.75			[4]	
	E12F342	RHR	XV	0.75			[4]	
	E12F394	RHR	MOV	6	Yes	[1]	[4]	
019	B21F016	Main Steam	MOV	3				Bypass
	B21F019	Main Steam	MOV	3				
020	E12F025A	RHR	SRV	1			[2]	Bypass
	E12F027A	RHR	MOV	18		[3]	[2]	
	E12F028A	RHR	MOV	18		[1,3]	[2]	
	E12F037A	RHR	MOV	12		[1]	[2]	
	E12F042A	RHR	MOV	14	Yes	[3]	[2]	
	E12F044A	RHR	XV	4			[2]	
	E12F107A							

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
021	E12F025B	RHR	SRV	1			[2]	Bypass
	E12F027B	RHR	MOV	18		[3]	[2]	
	E12F028B	RHR	MOV	18		[1,3]	[2]	
	E12F037B	RHR	MOV	12		[1]	[2]	
	E12F042B	RHR	MOV	14	Yes	[3]	[2]	
	E12F107B	RHR	XV	0.75				
	E12F044B	RHR	XV	4			[2]	
022	E12F041C	RHR	CV	12	Yes	[3]	[4]	Bypass
	E12F042C	RHR	MOV	12	Yes	[3]	[4]	
	E12F056C	RHR	XV	0.75			[4]	
	E12F234	RHR	XV	1			[4]	
023	E12D003A	RHR	Flange	18			[2]	Isolation
	E12F011A	RHR	MOV	4			[2]	
	E12F024A	RHR	MOV	18		[1,3]	[2]	
	E12F064A	RHR	MOV	4			[2]	
	E12F290A	RHR	MOV	1.5			[2]	
	E12F303	RHR	XV	0.5			[2]	
	E12F310	RHR	XV	0.5			[2]	
	E12F322	RHR	XV	0.75			[2]	

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
	E12F336	RHR	XV	0.75			[2]	
	E12F338	RHR	XV	1			[2]	
	E12F339	RHR	XV	1			[2]	
	E12F259	RHR	XV	1				
	E12F260	RHR	XV	1				
	E12F261	RHR	XV	1				
	E12F262	RHR	XV	1				
	E12F227	RHR	XV	1				
	E12F228	RHR	XV	1				
	E12F348	RHR	XV	0.75			[2]	
	E12F349	RHR	XV	0.75			[2]	
024	E12D003C	RHR	Flange	18			[4]	Isolation
	E12F021	RHR	MOV	14		[1,3]	[4]	
	E12F064C	RHR	MOV	4			[4]	
	E12F280	RHR	XV	1			[4]	
	E12F281	RHR	XV	1			[4]	
	E12F304	RHR	XV	0.5			[4]	
	E12F311	RHR	XV	0.5			[4]	

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
025	E22F014	HPCS	SRV	1				Isolation
	E22F015	HPCS	MOV	20		[3]	[4]	
026	E22F004	HPCS	MOV	12	Yes	[3]	[4]	Bypass
	E22F005	HPCS	CV	12	Yes	[3]	[4]	
	E22F021	HPCS	XV	0.75			[4]	
	E22F201	HPCS	XV	0.75			[4]	
	E22F218	HPCS	XV	1			[4]	
027	E22D005	HPCS	Flange	14			[4]	Isolation
	E22F012	HPCS	MOV	4			[4]	
	E22F023	HPCS	MOV	12		[1]	[4]	
	E22F035	HPCS	SRV	1			[4]	
	E22F301	HPCS	XV	1			[4]	
	E22F302	HPCS	XV	1			[4]	
	E22F303	HPCS	XV	0.5			[4]	
	E22F304	HPCS	XV	0.5			[4]	
028	E51F031	RCIC	MOV	6		[3]		Isolation
029	E51F068	RCIC	MOV	20		[3]		Isolation
	E51F077	RCIC	MOV	2.5				
	E51F257	RCIC	XV	0.5				
	E51F258	RCIC	XV	0.75				

Table 3-1

Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
030	E21F001	LPCS	MOV	24		[3]	[4]	Isolation
	E21F031	LPCS	SRV	0.75			[4]	
031	E21F005	LPCS	MOV	14	Yes	[3]	[4]	Bypass
	E21F006	LPCS	CV	14	Yes	[3]	[4]	
	E21F013	LPCS	XV	0.75			[4]	
	E21F200	LPCS	XV	0.75			[4]	
	E21F207	LPCS	XV	1			[4]	
032	E21D004	LPCS	Flange	14			[4]	Isolation
	E21F011	LPCS	MOV	4			[4]	
	E21F012	LPCS	MOV	14		[1]	[4]	
	E21F217	LPCS	XV	0.75			[4]	
	E21F218	LPCS	XV	0.75			[4]	
	E21F221	LPCS	XV	0.5			[4]	
	E21F222	LPCS	XV	0.5			[4]	
033	C11F083	CRD	MOV	2				Bypass
	C11F122	CRD	CV	2				
	C11F128	CRD	XV	0.75				
034	M41F011	Cont. Purge	AOV	20				Isolation
	M41F012	Cont. Purge	AOV	20				
	M41F042	Cont. Purge	XV	0.75				
035	M41F034	Cont. Purge	AOV	20				Isolation
	M41F035	Cont. Purge	AOV	20				
	M41F051	Cont. Purge	XV	0.75				

Table 3-1
Identification of Potential Risk Impact

					Potential Impact on:			
Penetration	Component	System	Type	Size	Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
036	P72F122	PSW	MOV	4				Isolation
	P72F123	PSW	MOV	4				
037	P72F121	PSW	MOV	4				Isolation
	P72F165	PSW	CV	4				
	P72F167	PSW	XV	0.75				
038	P71F150	Chilled Water	AOV	4				Isolation
	P71F151	Chilled Water	CV	4				
	P71F232	Chilled Water	XV	0.75				
039	P71F148	Chilled Water	AOV	4				Isolation
	P71F149	Chilled Water	AOV	4				
	P71F246	Chilled Water	XV	0.75				
040	M61F009	Cont. ILRT	XV	0.75				Isolation
	Blank Flange	Cont. ILRT	Flange	6				
	Blank Flange	Cont. ILRT	Flange	[illegible]				
041	P52F105	Service Air	AOV	3				Isolation
	P52F122	Service Air	CV	3				
	P52F258	Service Air	XV	0.75				
042	P53F001	Instrument Air	AOV	2.5				Isolation
	P53F002	Instrument Air	CV	2.5				
	P53F036	Instrument Air	XV	0.75				
041	G33F028	RWCU	MOV	4				Bypass
	G33F034	RWCU	MOV	4				
	G33F070	RWCU	XV	0.75				

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
044	P42F035	CCW	CV	10				Isolation
	P42F066	CCW	MOV	10				
	P42F161	CCW	XV	0.75				
045	G42F067	CCW	MOV	10				Isolation
	G42F068	CCW	MOV	10				
	G42F162	CCW	XV	0.75				
046	E51F019	RCIC	MOV	2		[1]		Isolation
	E51F251	RCIC	XV	1				
	E51F252	RCIC	XV	1				
047	B33F127	Sampling	MOV	0.75				Bypass
	B33F128	Sampling	MOV	0.75				
	P42F161	CCW	XV	0.75				
048	E12F055B	RHR	SRV	6		[1]		isolation
	E12F073B	RHR	MOV	2				
	E12F103B	RHR	CV	1.5				
049	G36F101	RWCU	AOV	4				Isolation
	G36F106	RWCU	AOV	4				
050	P45F067	Drywell Drain	AOV	6				Isolation
	P45F068	Drywell Drain	AOV	6				
051	P45F061	Drywell Drain	AOV	6				Isolation
	P45F062	Drywell Drain	AOV	6			[2]	
054	G41F053	FP Cooling	XV	12				Isolation
	G41F201	FP Cooling	XV	12				

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
056	P11F004	Upper Pool	CV	6				Isolation
	P11F075	Upper Pool	AOV	6				
	P11F095	Upper Pool	XV	0.75				
057	G41F028	FP Cooling	MOV	8				Isolation
	G41F040	FP Cooling	CV	8				
	G41F340	FP Cooling	XV	0.75				
058	G41F029	FP Cooling	MOV	8				
	G41F044	FP Cooling	MOV	8				
060	P45F273	Aux Bldg Drn	MOV	4				Isolation
	P45F274	Aux Bldg Drn	MOV	4				
	P45F275	Aux Bldg Drn	XV	0.75				
	P45F290	Aux Bldg Drn	XV	0.75				
061	C41F150	SLCS	XV	3				Isolation
	C41F151	SLCS	CV	2				
	C41F152	SLCS	XV	0.75				
065	E61F009	Cont Vent	CV	6				Isolation
	E61F010	Cont Vent	CV	6				
	E61F017	Cont Vent	XV	0.75				
066	E61F056	Cont Vent	AOV	6				Isolation
	E61F057	Cont Vent	AOV	6				
	M41F054	Cont. Purge	XV	0.75				
067	E12D003B	RHR	Flange	18			[2]	Isolation
	E12F011B	RHR	MOV	4			[2]	
	E12F024B	RHR	MOV	18		[1,3]	[2]	
	E12F064B	RHR	MOV	4			[2]	
	E12F290B	RHR	MOV	1.5			[2]	

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
	E12F305	RHR	XV	0.5			[2]	
	E12F312	RHR	XV	0.5			[2]	
	E12F321	RHR	XV	0.75			[2]	
	E12F331	RHR	XV	0.75			[2]	
	E12F334	RHR	XV	1			[2]	
	E12F335	RHR	XV	1			[2]	
	E12F212	RHR	XV	1				
	E12F213	RHR	XV	1				
	E12F249	RHR	XV	1				
	E12F250	RHR	XV	1				
	E12F276	RHR	XV	1				
	E12F277	RHR	XV	1				
	E12F350	RHR	XV	0.75			[2]	
	E12F351	RHR	XV	0.75			[2]	
069	P11F130	Refueling	AOV	12				Isolation
	P11F131	Refueling	AOV	12				
	P11F132	Refueling	XV	0.75				
	P11F425	Refueling	XV	0.75				
070	P53F003	Inst. Air	MOV	1				Isolation
	P53F006	Inst. Air	CV	0.75				
	P53F043	Inst. Air	XV	0.75				
071A	E21F018	LPCS	SRV	1.5			[4]	Isolation
071B	E12F025C	RHR	SRV	1			[4]	Isolation
	E12F346	RHR	MOV	1			[4]	
	E12F406	RHR	CV	1			[4]	
	E12F408	RHR	XV	0.75			[4]	
	E12F409	RHR	XV	0.75			[4]	

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
073	E12F036	RHR	SRV	4		[1]	[4]	Isolation
075	E51F078	RCIC	MOV	1.5				Isolation
076B	E12F005	RHR	SRV	1				
077	E12F055A	RHR	SRV	6		[1]		Isolation
	E12F103A	RHR	CV	1.5				
	E12F073A	RHR	MOV	2				
081	B33F125	Sampling	MOV	0.75				Bypass
	B33F126	Sampling	MOV	0.75				
082	M61F010	Cont. ILRT	XV	0.75				Isolation
	Blank Flange	Cont. ILRT	Flange	6				
	Blank Flange	Cont. ILRT	Flange	6				
083	G33F039	RWCU	MOV	6	[6]			Bypass
	G33F040	RWCU	MOV	6	[6]			
	G33F055	RWCU	XV	0.75				
084	P45F098	Drywell Drn	AOV	3				Isolation
	P45F099	Drywell Drn	AOV	3				

Table 3-1

Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [illegible]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
085	P60F009	SP Cleanup	AOV	12				Isolation
	P60F010	SP Cleanup	AOV	12				
	P60F011	SP Cleanup	XV	0.75				
	P60F034	SP Cleanup	XV	0.75				
086	P21F017	Demin.	MOV	2				Isolation
	P21F018	Demin.	MOV	2				
087	G33F001	RWCU	MOV	6	[5,6]			Bypass
	G33F002	RWCU	XV	0.75				
	G33F004	RWCU	MOV	6	[5,6]			
	G33F252	RWCU	MOV	8	[5]			
088	G33F053	RWCU	MOV	4	[6]			Bypass
	G33F054	RWCU	MOV	4	[6]			
	G33F061	RWCU	XV	0.75				
089	P41F159A	SSW	MOV	2				Isolation
	P41F163A	SSW	XV	0.75				
	P41F169A	SSW	CV	2				
090	P41F160A	SSW	MOV	2				Isolation
	P41F168A	SSW	MOV	2				
091	P41F160B	SSW	MOV	2				Isolation
	P41F168B	SSW	MOV	2				
092	P41F159B	SSW	MOV	2				Isolation
	P41F163B	SSW	XV	0.75				
	P41F169B	SSW	CV	2				

Table 3-1
Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
101C	M71F593	Drywell Inst	MOV	0.75				Isolation
101F	M71F591A	Drywell Inst	MOV	0.75				Isolation
102D	M71F591B	Drywell Inst	MOV	0.75				Isolation
103D	M71F592A	Drywell Inst	MOV	0.75				Isolation
104D	M71F592B	Drywell Inst	MOV	0.75				Isolation
105A	E61F596C	Cont H2	MOV	0.75				Isolation
	E61F596D	Cont H2	MOV	0.75				
106A	E61F595C	Drywell H2	MOV	0.75				Isolation
	E61F595D	Drywell H2	MOV	0.75				
106B	E61F597C	Drywell H2	MOV	0.75				Isolation
	E61F597D	Drywell H2	MOV	0.75				
106E	E61F598C	Cont H2	MOV	0.75				Isolation
	E61F598D	Cont H2	MOV	0.75				
107B	E61F598A	Cont H2	MOV	0.75				Isolation
	E61F598B	Cont H2	MOV	0.75				
107D	E61F595A	Drywell H2	MOV	0.75				Isolation
	E61F595B	Drywell H2	MOV	0.75				
107E	E61F597A	Drywell H2	MOV	0.75				Isolation
	E61F597B	Drywell H2	MOV	0.75				
108A	E61F596A	Cont H2	MOV	0.75				Isolation
	E61F596B	Cont H2	MOV	0.75				

Table 3-1

Identification of Potential Risk Impact

Penetration	Component	System	Type	Size	Potential Impact on: Initiating Events [7]	Mitigation Systems [7]	Shutdown Risk [7]	Containment Failure Mode
109A	D23F591	Drywell Sump	MOV	0.75				Isolation
	D23F592	Drywell Sump	MOV	0.75				
109B	D23F593	Drywell Sump	MOV	0.75				Isolation
	D23F594	Drywell Sump	MOV	0.75				
109D	M71F594	Drywell Inst	MOV	0.75				Isolation
	M71F595	Drywell Inst	MOV	1				
110A	M61F014	Drywell Inst	XV	1				Isolation
	M61F015	Drywell Inst	XV	1				
110C	M61F018	Drywell Inst	XV	1				Isolation
	M61F019	Drywell Inst	XV	1				
110F	M61F016	Drywell Inst	XV	1				Isolation
	M61F017	Drywell Inst	XV	1				
113	E30F593A	SP Level	MOV	0.75				Isolation
114	E30F592A	SP Level	MOV	0.75				Isolation
115	E30F594A	SP Level	MOV	0.75				Isolation
116	E30F591A	SP Level	MOV	0.75				Isolation
117	E30F593B	SP Level	MOV	0.75				Isolation
118	E30F592B	SP Level	MOV	0.75				Isolation
119	E30F594B	SP Level	MOV	0.75				Isolation
120	E30F591B	SP Level	MOV	0.75				Isolation

Notes:

1. Leakage through the isolation valve could create a flow diversion pathway. Unless the valve fails essentially full open, this diversion is considered negligible.

2. Testing of the valve makes one train of RHR unavailable for either shutdown cooling and LPCI while at shutdown.

3. Potential impact of a post-test restoration error.

4. Testing of the valve makes the train/systerm unavailable for makeup to the reactor vessel while at shutdown.

5. Leakage could cause an uncontrolled depressurization of the reactor, thereby causing RCIC to be unavailable as a makeup source and negating the need for other depressurization measures.

6. Valve isolates to prevent a LOCA should a downstream line rupture/leak.

7. A blank in this column means that the component has no identified impact on risk.

Acronyms:

AOV: Air Operated Valve

MOV: Motor Operated Valve

XV: Other valve such as test, vent, or drain valves

SRV: Safety Relief Valve

Table 3-2
Failure Rates for Internal Leakage

Component Type	Failure Rate (hr^{-1})	Reference
Motor Operated Valve	1.0E-7	GGNS IPE (NREP)
Pneumatic Valve	1.0E-6	GGNS IPE (EG&G)
Manual Valve	Negligible	
Check Valve	5.0E-7	GGNS IPE (NUREG/CR-2728)
Relief Valve	3.9E-6	GGNS IPE (NUREG/CR-4550)

Table 3-3
Post-Test Restoration Error Analysis

System or Train	Error	Basic Error Probability [1]	Recovery Factors [2], [3]	P_{LLRT}	Functional Test or Inspection Frequency [4],[5]	U_{LLRT} Current Test Schedule	U_{LLRT} Proposed Test Schedule
RHR Train A	Failure to restore train	3.0E-3	1.0E-2	3.0E-5	Q	5.0E-6	1.0E-6
	Failure to reopen manual valve	1.0E-3	1.0E-1	1.0E-4	R	1.0E-4	2.0E-5
	Total Restoration Error			1.3E-4		1.05E-4	2.1E-5
RHR Train B	Failure to restore train	3.0E-3	1.0E-2	3.0E-5	Q	5.0E-6	1.0E-6
	Failure to reopen manual valve	1.0E-3	1.0E-1	1.0E-4	R	1.0E-4	2.0E-5
	Total Restoration Error			1.3E-4		1.05E-4	2.1E-5
RHR Train C	Failure to restore train	3.0E-3	1.0E-2	3.0E-5	Q	5.0E-6	1.0E-6
	Failure to reopen manual valve	1.0E-3	1.0E-1	1.0E-4	R	1.0E-4	2.0E-5
	Total Restoration Error			1.3E-4		1.05E-4	2.1E-5
HPCS	Failure to restore train	3.0E-3	1.0E-2	3.0E-5	Q	5.0E-6	1.0E-6
	Failure to reopen manual valve	1.0E-3	1.0E-1	1.0E-4	R	1.0E-4	2.0E-5
	Total Restoration Error			1.3E-4		1.05E-4	2.1E-5
LPCS	Failure to restore train	3.0E-3	1.0E-2	3.0E-5	Q	5.0E-6	1.0E-6
	Failure to reopen manual valve	1.0E-3	1.0E-1	1.0E-4	R	1.0E-4	2.0E-5
	Total Restoration Error			1.3E-4		1.05E-4	2.1E-5
RCIC	Failure to restore train	3.0E-3	1.0E-2	3.0E-5	Q	5.0E-6	1.0E-6

Notes:
1. Basic error probability includes a recovery factor of 0.1 for post-test verification activities.
2. Recovery factor for a functional test of the train performed after LLRT.
3. Recovery factor for a pre-startup valve lineup verification.
4. Functional test performed quarterly.
5. Valve lineup verification every refueling outage.

Table 3-4
Effect of Post-Test Restoration Errors

System or Train	ΔU_{LLRT}	I_{FV}	Change in Core Melt Frequency (year^{-1}) [1]
RHR-A	-8.4E-5	-	Negligible
RHR-B	-8.4E-5	1.88E-5	-9.1E-12
RHR-C	-8.4E-5	-	Negligible
HPCS	-8.4E-5	1.91E-2	-9.2E-9
LPCS	-8.4E-5	3.68E-5	-1.8E-11
RCIC	-4.0E-6	1.23E-3	-2.8E-11

Notes: 1. Baseline core melt frequency from the GGNS IPE is 1.72E-5/year.

Table 3-5
Shutdown Risk Impact
Refueling Outage RF06

LLRT	Core Damage Risk Impact	RCS Boiling Risk Impact
Baseline Outage	6.9E-7 (total probability for the outage)	3.0E-3 (total probability for the outage)
Common RHR Shutdown Cooling Suction Line	No change	No change
RHR Train A	-3.4E-12	-9.6E-6
RHR Train B	-3.3E-13	-1.4E-6
LPCS	-8.9E-9	-9.8E-7
HPCS	-1.2E-10	No change
All LLRTs Combined	-1.1E-8	-1.2E-5

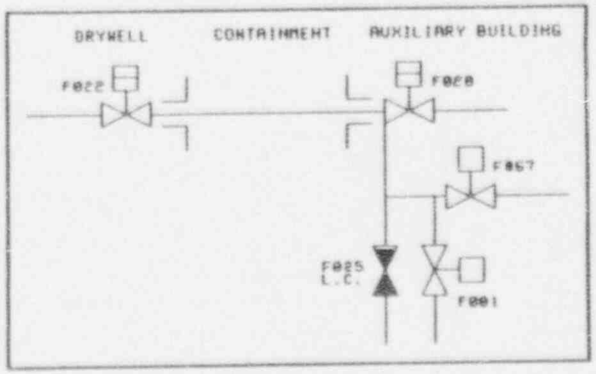
Table 3-6
Containment Bypass Impact

Penetration	Bypass Failure Probability (IPE Results - Current Test Program)	Bypass Failure Probability (Proposed Test Program Changes)	Comments
5 to 8	2.7E-5	2.9E-3	While an increase in the test interval for these valves is unlikely, the current analysis considers a 5-cycle test interval to illustrate the maximum impact.
9 and 10	Not quantified	4.0E-6	While an increase in the test interval for these valves is unlikely, the current analysis considers a 5-cycle test interval to illustrate the maximum impact.
14	4.9E-9	No change	No change due to hydro testing of these valves every refueling outage.
17	3.0E-6	No change	No change due to hydro testing of these valves every refueling outage.
18	Not quantified	No change	No change due to hydro testing of these valves every refueling outage.
19	9.0E-6	5.4E-5	
20 and 21	Not quantified	No change	No change due to hydro testing of these valves every refueling outage.
22	Not quantified	No change	No change due to hydro testing of these valves every refueling outage.
26	Not quantified	No change	No change due to hydro testing of these valves every refueling outage.

Table 3-6
Containment Bypass Impact
(Continued)

Penetration	Bypass Failure Probability (IPE Results - Current Test Program)	Bypass Failure Probability (Proposed Test Program Changes)	Comments
31	Not quantified	No change	No change due to hydro testing of these valves every refueling outage.
33	Negligible	No change	High pressure system; negligible leakage even with isolation failure.
43	1.0E-4	1.2E-4	
47	1.2E-6	1.9E-5	
81	1.2E-6	1.9E-5	
83	Not quantified	1.9E-5	
87	Negligible	No change	High pressure system; negligible leakage even with isolation failure.
88	Negligible	No change	High pressure system; negligible leakage even with isolation failure.

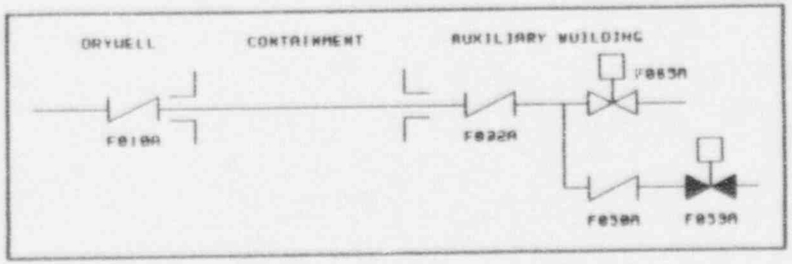
Figure 3-1
Calculation of Probability for
Failure to Isolate MSIV Penetration



Valve	P_D	λ (hr^{-1})	$\lambda T/2$	P_I
F022	3.0E-3	1.0E-6	4.4E-2	4.7E-2
F028	3.0E-3	1.0E-6	4.4E-2	4.7E-2
F067	3.0E-3	1.0E-7	4.4E-3	7.4E-3
F001	3.0E-3	1.0E-7	4.4E-3	7.4E-3
F025	< 1.0E-7	Negligible	-	< 1.0E-7

$$P_{Penetration} = P_{I,F022} * (P_{I,F028} + P_{I,F067} + P_{I,F001} + P_{I,F025}) = 2.9E-3$$

Figure 3-2
Calculation of Probability for
Failure to Isolate Feedwater Penetration



Valve	P_D	λ (hr^{-1})	$\lambda T/2$	$P_{I,i}$
F010A	1.0E-3	5.0E-7	2.2E-2	2.3E-2
F032A	1.0E-3	5.0E-7	2.2E-2	2.3E-2
F065A	3.0E-3	1.0E-7	4.4E-3	7.4E-3
F050A	-	5.0E-7	2.2E-2	2.2E-2
F053A	-	1.0E-7	4.4E-3	4.4E-3

$$P_{Penetration}=P_{I,F010A}*P_{I,F032A}*(P_{I,F065A}+P_{I,F050A}P_{I,F053A})=4.0E-6$$

4.0 Type A Testing - Proposed Changes and Basis

4.1 Proposed Changes

"Type A Tests" are defined in Appendix J Section II.F. as "tests intended to measure the primary reactor containment overall integrated leakage rate" (ILRT).

Exemption is requested from the following paragraph in Section III.D.1(a) for Type A test intervals:

> "Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant service inspection."

Surveillance Requirement 4.6.1.2.a associated with Technical Specification 3/4.6.1.2 "Containment Leakage" is being revised to reflect the 10 year Integrated Containment Leakage Rate interval.

4.2 Proposed Type A Testing Program

In lieu of 10CFR50 Appendix J Section III.D.1(a) as stated above, Grand Gulf Nuclear Station proposes to perform a Type A test once every 10 years.

4.3 Type A Testing History

The testing history for GGNS supports the proposed surveillance interval. Trends of previous test results indicate that the interval extension would not jeopardize the ability of containment to maintain the leakage rate at or below the required Type A limits.

The following is a history of GGNS Type A testing:

- The preservice Type A test was conducted on January 4-5, 1982 and was acceptable (approximately 42% of 0.75 L_a).

- The first periodic Type A test was conducted on November 3-4, 1985 and was considered a failure. The cause of the failure was determined to be excessive leakage through four Type C tested penetrations. Corrective actions were taken to preclude these problems in the future as outlined below. The final leakage was determined to be approximately 57% of 0.75 L_a.

  The isolation valves in two main steam line penetrations had been closed prior to the Type A test using a test switch which limits the speed of closure. This method is used only for partial stroke testing and is not the normal method of closure. Plant procedures have been revised to specify the correct method for normal closure for these valves prior to testing. The isolation valves in a spare Standby Liquid Control System penetration had not been completely closed prior to the Type A test. While both valves required corrective maintenance to restore them to a leak-tight condition, it was determined that a one-time action would prevent this condition in the future.

The inboard isolation valve is a manual stop-check valve which is required to be locked in the open position during normal operation. Since this penetration is a spare penetration with capped ends and is designated for future use, there is no need for the valve to be locked open. Plant procedures were revised to specify the normal position of this valve as locked closed.

The outboard isolation valve is a manual gate valve which was found to be partially open after the Type A test. Attempts to close the valve required forces far in excess of those normally required. The cause was determined to be a lack of internal lubrication on the valve stem. Lubrication of the valve stem is an action which is normally performed prior to initial installation of a valve and periodically thereafter as part of a preventive maintenance program. This valve was entered into the GGNS Preventive Maintenance Program to assure adequate lubrication, and has not since been a problem. The Residual Heat Removal (RHR) test return line valve was another cause of the Type A failure. The original technical specification required Type C testing of this valve with water. Subsequently a determination was made that air testing was required (MP&L letter AECM-83/0540, dated September 12, 1983), and the technical specification was subsequently changed to provide for air testing.

Failure of the Type C test for the subject valve prior to the Type A test required an infinite penalty to the Type A test. Corrective maintenance was performed on the subject valve and a successful "as left" Type C test was performed.

It was determined that a permanent design modification would allow testing of the valves in penetration 24 with water. The modification consisted of extending the RHR test return line approximately 18 inches deeper into the suppression pool and requesting a Technical Specification change to allow Type C testing with water. (MP&L) request AECM-85/0168, dated July 3, 1985, and subsequent issuance of Amendment 4 to the operating license MAEC-85/0314, dated September 18. 1985.) Type C testing of this valve will be performed in the future with water.

- The second periodic Type A test was conducted on April 15-16, 1989 and was acceptable (approximately 54% of 0.75 La).

- The third periodic Type A test will be conducted during RF06 scheduled to commence in October, 1993.

4.4 <u>Basis for Proposed Changes</u>

Factors affecting leak-tightness of the containment may be categorized as 1) active components which are leak rate tested by Type B and C tests and 2) passive components which constitute the containment structure and are tested during the Type A test (i.e., the ILRT).

1. Active Components

Industry experience indicates that the failures associated with Type A tests are generally found on Type B & C tested penetrations. The only Type A test failure at GGNS was associated with Type C tested penetrations. These penetrations would have received increased intervals based on the Type B & C proposed program in Section 3.

Therefore, continued overall leak tightness of the active containment components can be assured by a reliable Type B and C testing program.

2. Passive Structure

Two mechanisms could adversely affect the passive structural capability of containment. The first is deterioration of the structure due to pressure, temperature, radiation, chemical or other such effects. Secondly, modifications can be made to the structure which, if not carefully controlled, could leave the structure with reduced capability.

Absent actual accident conditions, structural deterioration is a gradual phenomenon which, we believe, requires periods of time well in excess of the proposed 10 year ILRT interval. We are unaware of any information developed in the industry which identifies relatively quick-acting degradation mechanisms which could adversely affect containment integrity. Other than accident conditions, the only pressure challenge to the containment structure is the ILRT itself. One of the discords of the current ILRT Interval is the application of test pressure (based on accident analyses) which could potentially affect the passive containment structure over the long term. Increased ILRT intervals could therefore lessen the potential for adverse pressure effects.

10CFR50 Appendix J Section V.A. requires a general inspection of the accessible interior and exterior surfaces of the containment structures and components to be performed prior to any Type A Test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. At GGNS there has been no evidence of structural deterioration that would impact structural integrity or leak tightness.

Modifications made to the containment must continue to meet at least original construction requirements. In fact, modifications which may alter the passive containment structure are infrequent, at best. By their nature, such modification will receive extensive scrutiny to ensure containment capabilities are not diminished. The GGNS design change, 50.59 and similar programs have been demonstrated effective in providing high quality oversight of such safety significant modifications. In addition, 10CFR50 Appendix J Section IV.A requires Type A testing to be performed following any major modification to the primary containment structure boundary. This requirement will be maintained.

The extended Type A testing interval is therefore justified based on:

- Reliability of the passive containment structure

- Maintaining containment boundary modification testing requirements in accordance with Appendix J Section IV.A. and

- The ability of the Type B and C testing program to detect primary sources of containment leakage that cause type A test failures.

4.5 Risk Impact Assessment

The function of the Appendix J Type A test is to ensure that the integrated leakage from containment is within acceptable limits. One effect of the proposed changes is to potentially increase the probability that containment leakage will occur and go undetected between tests. Such leakage may be the result of leakage through containment penetrations, through airlocks or through containment structural faults. The Appendix J Type B and C tests are effective in detecting leakage through containment penetrations. The risk impact of proposed changes to these tests is discussed in Section 3.5. Airlocks are tested with much greater frequency than the Type A test frequency, so that the proposed changes in Type A testing do not impact the airlock performance. Changes to the airlock test program are discussed in Section 5 along with the risk impact of those changes. This section addresses only the risk impact of changes in the Type A tests due to postulated leakage which is not detectable by other parts of the Appendix J test program.

The risk impact of containment structural leakage is to create a release pathway for radionuclides in the event that the containment is challenged, such as in a LOCA or severe accident. Such leakage does not create any new accident scenarios, nor does it contribute to the initiation of any accident. The proposed changes may affect 1) the probability for containment leakage or failure following an accident and/or 2) the consequences of such accidents. The GGNS IPE examined containment response during severe accidents and provides estimates of consequences. The IPE is used as the basis for estimating the impact of changes to the Appendix J Type A test program.

The containment structure is passive. Under normal conditions, there is no significant environmental or operational stress which could contribute to its degradation. Passive failures resulting in significant containment structural leakage are therefore extremely unlikely to develop between Type A tests. No such failures have ever occurred at GGNS.

The post-accident environment within containment may be severe and could contribute to failure of its function. Such environments were considered as part of the IPE. The IPE found, however, that postulated containment failure under severe accident conditions is due to phenomenological effects associated with severe accidents. None of the identified containment failure mechanisms for severe accidents would be impacted by the proposed changes in the Appendix J Type A testing program.

The IPE found that containment isolation failure was not a significant containment failure mechanism and had a probability of occurrence (given an accident which challenges containment) of less than 1.0E-3. This probability was dominated by active failures of containment isolation valves to close on demand. The proposed changes to the Appendix J Type A testing program do not impact the probabilities for such demand failures.The conclusion of this qualitative risk assessment is that Type A tests do not significantly affect the frequency of accident sequences involving releases from containment. This is due to the following.

- Other testing programs will effectively detect containment leakage caused by degradation of containment penetrations.

- Passive failure of the containment structure itself is extremely unlikely.

  Events challenging containment have calculated frequencies of occurrence which are very low.

- Containment failure mechanisms which are dominant in the IPE are associated with severe accident phenomena which are not affected by the proposed changes in the Appendix J Type A test program.

- The containment isolation failure probability is dominated by active component failures, which are not affected by the proposed changes in the Appendix J Type A test program. Even so, containment isolation is not found to be a significant failure mechanism in the GGNS IPE.

4.6 Cost Reduction Assessment

ILRTs are labor intensive and require an estimated 2,000 man-hrs for each refueling outage. By eliminating 6 ILRTs over the remaining licensed life of the plant the estimated savings in labor cost (assuming an estimated $30/hr labor rate) will be $360,000. Using the following assumptions, the outage critical path reduction in cost can be estimated to be $10,000,000 over the remaining licensed life of the plant.

- 3.5 days of critical path time for each ILRT
- $500,000 per day for plant downtime
- 6 ILRTs eliminated over remaining licensed life of plant

5.0 Type B Airlock Test - Proposed Changes and Basis

5.1 Proposed Changes

Type B tests are those previously defined in section 3.1.

Exemption is requested from the following paragraphs in Appendix J Section III.D.2 for Type B airlock test intervals:

> "(b)(i) Air locks shall be tested prior to initial fuel loading and at 6 month intervals thereafter"
>
> "(b)(iii) Air locks opened during periods when containment integrity is required by the Plants Technical Specifications shall be tested within 3 days after opening. For air lock doors opened more frequently then once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings. For airlock doors having testable seals, testing the seals fulfills the 3-day test requirments."

In accordance with 10CFR50.90, Surveillance Requirement 4.6.1.3.a associated with Technical Specification 3/4.6.1.3 "Containment Air Locks" is being revised to reflect the proposed test interval.

5.2 Proposed Airlock Testing Program

In lieu of 10CFR50 Appendix J Section III.D.2(b)(i) as stated above, GGNS proposes to perform Type B tests on each containment airlock every 2 years.

Additionally, in lieu of 10CFR50 App. J Section III.D.2(b) (iii), GGNS proposes to perform the Type B airlock seal test every 30 days.

5.3 Airlock Testing History

The Type B containment airlock test success rate is 97%. The testing history for Type B airlock tests from 1986 to 1993 demonstrates the continued leak tightness of these components. The following provides a list by airlock of the total tests, total failures and pass rate percentages.

CONTAINMENT AIRLOCK TEST

CONTAINMENT AIRLOCK	TOTAL TESTS	TOTAL FAILURES	PERCENT PASSES
1M23Y001	17	0	100%
1M23Y002	15	*1	93%
TOTAL	32	1	97%

* The one failure of 1M23Y002 airlock was due to inadequate tightening of the door seal bolts. The original design of the seal clamps utilized a bolting configuration that allowed

the bolts to back out. The configuration has since been changed to prevent any recurrence of this problem.

The airlock testing histories clearly show that frequent testing of the airlocks and airlock seals (i.e. every 6 months and every 3 months respectively), does not provide benefits to justify the time, manpower, man-rem exposure and loss of airlock availability needed to perform them.

The Type B containment airlock seal test success rate is 100%. Because of the large number of airlock seal tests performed, only the testing history from 1991 to 1993 is provided here. Interviews with engineers involved with the airlock seal tests prior to 1991 verified that, to the best of their knowledge, no airlock seal test has ever failed. The following provides a list by airlock of the total tests, total failures and pass rate percentages.

CONTAINMENT AIRLOCK SEAL TEST

CONTAINMENT AIRLOCK	TOTAL TESTS	TOTAL FAILURES	PERCENT PASSES
1M23Y001	245	0	100%
1M23Y002	244	0	100%
TOTAL	489	0	100%

5.4 <u>Basis for Proposed Changes</u>

The airlock was not included in the performance based testing program for type B and C components because of the 5 year service life of the door seal. As soon as the performance of an airlock was established (by 2 consecutively passed tests), the service life for the seals would expire. The seals would then require replacement with the associated re-establishment of performance. This would effectively result in a continuous 2 year interval.

Several methods are employed at GGNS to monitor airlock performance and to ensure operational readiness.

Programs are in place which will detect excessive wear and degradation and will identify problems that are potential contributors to airlock leakage.

Each test for airlock seals is evaluated and trended by System Engineering to assist in prediction of seal failure. Testing once a month will provide a sufficient number of data points for accurate trending.

In addition, the System Engineer performs weekly system walkdowns, monitors system surveillances, reviews any requests for work and provides corrective action to prevent component failures. The System Engineer also provides a single focal point for the system which allows a concentrated and dedicated review of the system's performance parameters and failures and provides a good mechanism to identify containment airlock problems.

The most probable pathway for airlock leakage is through the door seals which will be tested monthly. The relief valve and flange will be tested in accordance with the proposed Type B & C test program. Airlock seals are monitored for service life and are replaced according to their service life.

System surveillances are performed periodically to identify any problems associated with containment airlock component failures. Routine maintenance is performed on the airlocks under the Preventive Maintenance program. This program aids in identifying problems that could contribute to airlock leakage.

A root cause evaluation is performed for each significant failure per the GGNS nonconformance program. Root cause evaluations are performed to identify the underlying problem causing the failure. Identification of the underlying cause assures that the appropriate corrective action will be taken. Root causes performed for Type B/C test failures will help ensure the appropriate corrective action will be taken to increase the performance of containment airlock components and to prevent any future failures.

Because airlock seal failures occur gradually over an extended period of time and do not occur abruptly, there is generally time to initiate corrective action prior to failure. In the unlikely case of abrupt seal failure, the failure would immediately be detected by the sound of air leakage and would then be repaired or isolated as appropriate.

Finally, reducing the amount of testing will reduce manipulation of components in the airlock (e.g., plugging the relief valve and removing the blind flange to support testing). Reduced testing will also reduce the possibility for human error during restoration of the airlock to its operational configuration and will generally increase system availability.

5.5 Risk Impact Assessment

The containment airlocks provide a passive barrier preventing the release of radionuclides from containment in the event of a LOCA or severe accident. The risk impact of an airlock seal failure is to create a release pathway under such accident conditions. The airlocks do not contribute to any new accident sequences, nor do they affect the probability of occurrence for any accident initiator. The proposed changes to the airlock test program may affect the probability for leakage occurring and going undetected between tests. This will, in turn, affect the probability for containment isolation failure, given a challenge to containment.

In over ten years of airlock testing at GGNS, there has been one test where airlock leakage exceeded the allowable leak rate. This leakage was found to be the result of a design deficiency and has since been corrected. A recurrence of this failure is not possible. There have been no failures of the airlock seals themselves. Based on the testing history at GGNS, the airlock failure rate can be estimated to be less than 2.0E-6 per hour. The probability that the airlock is leaking at a point in time between tests when the containment may be challenged is calculated as

$$P_{Leak} \cong \lambda \frac{T}{2}$$

where λ is the failure rate and T is the mean time between tests. With the proposed seal testing interval of one test every thirty days, the leakage probability is calculated to be less than 7.2E-4.

The GGNS IPE considered containment isolation failure as one of the possible causes for a release from containment following a severe accident. The study concluded that the probability of containment isolation failure, given a challenge to containment, was negligible (less than the screening value of 1.0E-3). As a result, other possible containment failure modes were found to be dominant. The proposed changes to the airlock test program may contribute to an increased probability for containment isolation failure. The probability may approach the screening value used in the IPE to dismiss such sequences as insignificant. This screening value was deliberately set to a conservatively low value in the IPE. The calculated core damage frequency in the IPE is very low (1.72E-5 per year). The frequency of core damage, in conjunction with an airlock failure is therefore extremely small (on the order of 10^{-8} per year). The results of the IPE remain valid, i.e., accident sequences with containment isolation failure remain insignificant.

The proposed changes to the airlock test program do not significantly affect risk. While they may slightly increase the probability for failure of containment isolation, this probability remains very low. Risk results are found to be dominated by other containment failure modes which are unaffected by the proposed changes.

5.6 Cost Reduction Assessment

The airlock test requires approximately 34 man-hours for each airlock. With 2 tests per cycle and an assumed hourly rate of $40.00, $97,920 can be saved over the remaining licensed plant life. Savings of $2,799,360 can be expected for the airlock seal test using similar assumptions.

6.0 Basis for Exemption

EOI has evaluated the proposed changes against the criteria for specific exemptions as described in 10CFR50.12.

The NRC has historically recognized that there are circumstances where on balance it would not be equitable or in the public interest to require literal adherence to regulation. This has been particularly true in situations where literal adherence would not result in an improvement in overall safety or a reduction in risk to the public. The assessments of the proposed changes confirm that the proposed methods are appropriate and provide a technically sound method for accomplishing the regulatory purpose of the rule. No undue risk will result from deviating from the intervals specified in Appendix J. In fact, the better utilization of scarce resources will allow more effective implementation of an overall safety policy and is consistent with the NRC's policy of efficient and effective nuclear safety regulation.

EOI believes that the proposed changes are not in violation of any applicable law, will not present an undue risk to the public health and safety and would be consistent with the common defense and security.

In addition to meeting the general standards of 50.12(a)(1), the proposed changes meet the criteria for special circumstances as described in 10CFR50.12(a)(2)(ii),(iii),(iv), and (vi):

<u>50.12(a)(2)(ii)</u>

The interval requirements specified in Appendix J are not necessary to achieve the underlying purpose of the rule. The stated purposes of containment leakage rate testing are:

1) to assure that leakage through the primary reactor containment and systems and components penetrating primary containment will not exceed allowable leakage rate values as specified in the technical specifications or associated bases

2) to assure that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment

The proposed changes provide an alternative means to achieve the underlying purpose of the regulation. The programmatic controls associated with the proposed GGNS performance-based leakage testing program provide protection equivalent to the prescriptive requirements of Appendix J while allowing for significant reductions in cost and increases in efficiency. The negligible change in risk (both positive and negative) provides equivalent assurance that allowable leakage rates will not be exceeded. All valves that are required to close for containment isolation and that are not maintained closed at all times during power operation are stroke tested in accordance with ASME Section XI, subsection IWV. These requirements are at least as restrictive as those of Appendix J. This ensures that proper maintenance and repairs can be performed as required by the rule.

50.12(a)(2)(iii)

EOI believes that the additional cost associated with literal compliance is not necessary to assure adequate safety protection and therefore represents an undue hardship. In fact, over the long run literal compliance detracts from safety by focusing scarce resources on relatively low risk areas of the plant.

50.12 (a)(2)(iv)

The proposed exemption will, on balance, result in benefit to the public health and safety. The change in safety is negligible and essentially neutral as shown earlier. Resources now being expended on meeting the requirements of Appendix J could be better utilized. At least part of the savings resulting from this exemption will be reinvested to improve safety in more appropriate areas. Therefore, this change will result in an improvement in overall safety and effectively result in a reduction in risk to the public. EOI has a demonstrated commitment to safety and will continue to commit resources as necessary to fulfill that commitment.

50.12(a)(2)(vi)

Plant specific probabilistic risk assessments (PRAs) were not available and were therefore not considered when the regulation requiring compliance with Appendix J was adopted. PRAs have consistently established that overall plant risk is relatively unaffected by containment leakage. Containment leakage is not a factor in the dominant accident scenarios involving containment related contributions to risk. The cost of containment leakage rate testing is demonstrably high. Not only can regulatory burden be reduced substantially, but occupational dose can be reduced without any significant impact on safety.

7.0 Basis for No Significant Hazards Consideration Determination

EOI has evaluated the no significant hazards considerations in regard to this request for a license amendment. In accordance with 10CFR50.91(a), EOI is providing the analysis of the proposed amendment against the three standards in 10CFR50.92(c) below:

1) The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

Two initiating events were identified which could be affected by the proposed changes.

An interfacing system LOCA could be caused by significant leakage of both normally closed isolation valves in systems with high pressure/low pressure interfaces. Interfacing system LOCAs were considered for the LPCI, LPCS, HPCS, and RCIC systems. In all cases the isolation valves are water tested for leakage to ensure the integrity of the high pressure/low pressure system boundary in accordance with GGNS Technical Specification 4.4.3.2.2. The proposed changes to the testing requirements of Appendix J do not include any changes to this section of the GGNS Technical Specifications. Therefore the proposed changes do not alter the frequency for interfacing system LOCAs.

The second event evaluated was a LOCA outside containment. In this case the probability for failure of the MSIVs and the feedwater isolation valves were calculated and combined with the frequency of a pipe break outside containment and the conditional probability of core melt given a LOCA. The increase in core damage frequency is extremely small and therefore does not significantly increase the probability of any previously evaluated accident. It should also be noted that any minor increase in core damage frequency is offset by reductions in core damage frequency due to other aspects of the proposed changes. Moreover, the MSIVs and FWIVs are expected to remain on fixed two year testing intervals. Therefore, the proposed changes will have no effect on the core damage frequency due to a LOCA outside containment.

Failure of, or leakage through a containment barrier may increase the consequences of those accidents previously evaluated. Because the leakage probability for two valves in series to fail is very small and because all lines isolated by a single containment isolation valve always have a water seal and cannot act as a release pathway unless the integrity of the connected system is compromised, there is no significant increase in the consequences of any previously evaluated accident.

Containment bypass can also increase the consequences of evaluated accidents. Accident sequences involving containment bypass have been shown to be relatively insignificant by the GGNS IPE. Nonetheless, the potential for bypass was analyzed. The analysis showed that the probabilities for bypass were dominated by failure to close scenarios, which would not be affected by the Appendix J leakage testing program. Instead, many other programs are in place at GGNS to monitor containment component performance and to ensure that proper maintenance and repairs are made during the service life of the containment. Other routine surveillances are performed periodically to ensure that the valves will close on demand. In fact, all valves that are required to close for containment isolation and that are not maintained closed at all times during power operations are stroke tested quarterly or at a minimum, during each refueling outage in accordance with ASME section XI, subsection IWV.

2) The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

This request involves the reduction in the local leak rate and the integrated leak rate testing frequencies. The method of performing the test is not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Extending the test frequencies has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident or malfunction from those previously analyzed.

3) The proposed change does not involve a significant reduction in a margin of safety.

The safety assessment for the proposed changes concluded that the overall risk impact is neutral and essentially negligible. Any containment isolation barrier allowed to be tested at less frequent intervals will have demonstrated enhanced performance which minimizes the potential for increased leakage. The assessment further shows that there is reasonable assurance that an acceptable level of performance for the containment isolation function can be maintained.

The margin of safety that has the potential of being impacted by the proposed changes involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a which is defined by the GGNS Technical Specifications to be 0.437 percent by weight of the containment air per 24 hours at 11.5 psig (P_a). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (11.5 psig, P_a).

To provide additional conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic integrated leakage rate test and to less than or equal to 0.60 L_a for type B and C leakage rate tests. This is done to account for the possible degradation of the containment leakage barriers between tests. These acceptance criteria ensure that an acceptable margin of safety is being maintained and will not be altered by the proposed changes. The preservation of this margin will continue to provide for potential degradation of the leakage barriers between tests.

No change in the method of testing is being proposed. The tests will continue to be done at full pressure (P_a) or greater. The test pressure for primary containment isolation valves will continue to be applied in the same direction as would be required for the valve to perform its safety function (unless a different direction can be shown to be equivalent or conservative). Primary containment penetrations which require Type B leakage rate tests will be performed in the same manner as before. The type A test will continue to be performed at full pressure (P_a). Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

No change in the owner's allowable leakage rate is being proposed. These conservative leakage rates ensure that the containment leakage remains low.

Although changes to the frequency of testing of main steam line isolation valves is not anticipated at this time, any change would be based on the performance and operating experience associated with these valves. The change would therefore be consistent with the bases specified in the Technical Specifications for these valves (Bases section 3/4.6.1.2). The bases, in effect, specify a performance based approach. The measured leakage rate for all four main steam lines through the isolation valves will not be allowed to exceed the 100 scf per hour required by the GGNS Technical Specifications.

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards considerations.