CENTERIOR ENERGY

PERRY NUCLEAR POWER PLANT
10 CENTER ROAD
PERRY, OHIO 44081
(216) 259-3737

Mail Address:
P.O. BOX 97
PERRY, OHIO 44081

Donald C. Shelton
SENIOR VICE PRESIDENT
NUCLEAR

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
License Amendment Request: Drywell Personnel Air Lock Shield Door Analyses

Gentlemen:

Nuclear Regulatory Commission (NRC) review and approval of a license amendment for the Perry Nuclear Power Plant (PNPP) Unit 1 is requested. On February 15, 1996, PNPP personnel met with the NRC Staff to discuss the proposed amendment. This license amendment involves Updated Safety Analysis Report (USAR) descriptions of analyses which have been performed to demonstrate the drywell personnel air lock shield doors and their supporting structures remain functional when the doors are open during power operation. These USAR changes have been identified as requiring NRC review prior to incorporation, in accordance with 10 CFR 50.59(c). The USAR changes justify plant operation for one cycle, until the sixth refueling outage. A long-term resolution for this issue will be completed prior to restart from the sixth refueling outage.

Attachment 1 provides a Summary, a Description of the Proposed Changes, a System Description, a Safety Analysis, an Environmental Consideration and a Statement Of Exigent Circumstances. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides a copy of the marked-up USAR pages.

Handling of this proposal on an exigent basis is requested. The currently scheduled date for return of the plant to Operational Condition 2 is March 25, 1996. However, it is expected that this restart date may be achieved earlier, and a thirty day Federal Register Notice period could delay plant startup. Communication of outage schedule progress will continue with the NRC Project Manager for PNPP, and issuance of this amendment at least three days in advance of the scheduled startup date is requested, to allow for processing of associated paperwork.

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If you have questions or require additional information, please contact Mr. James D. Kloosterman, Manager - Regulatory Affairs at (216) 280-5833.

Very truly yours,

for D.C. Shelton

Donald C. Shelton

BSF:bf

Attachments

cc: NRC Project Manager
NRC Resident Inspector Office
NRC Region III
State of Ohio

I, Richard D. Brandt, being duly sworn state that (1) I am General Manager, Perry Nuclear Power Plant Department of the Cleveland Electric Illuminating Company, (2) I am duly authorized to execute and file this certification on behalf of The Cleveland Electric Illuminating Company and Toledo Edison Company, and as the duly authorized agent for Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

Richard D. Brandt

Richard D. Brandt

Sworn to and subscribed before me, the 27th day of February, 1996.

Jane E. Mott

JANE E. MOTT
Notary Public, State of Ohio
My Commission Expires Feb. 20, 2000
(Recorded in Lake County)

SUMMARY

The Drywell Personnel Airlock Shield Doors need to be opened during plant startup and shutdown. This is driven by normal activities, such as Main Steam Isolation Valve (MSIV) yoke rod guide adjustments, and inspections for piping or flange leaks. These activities are performed at various reactor pressure vessel (RPV) pressures and power levels.

The original design considerations for the shield doors did not include evaluations of various design basis loading combinations such as suppression pool swell loads and seismic (earthquake) loads on the doors when in the open position. When closed, the doors are shielded from suppression pool swell loads by a concrete shelf, and are held closed with seismic restraints.

Analyses performed using design basis criteria have concluded that with the shield doors in the open position during Operational Conditions 1, 2, and 3, a limited number of the members of the 620'-6" structural steel platform and monorail suspension structure (from which the shield doors are suspended), are subjected to stress levels beyond design basis acceptance criteria, for certain postulated loading combinations. However, additional analyses performed utilizing alternate design criteria have confirmed that the 620'-6" structural steel platform will perform its safety function and the monorail suspension structure will support the doors following accident loading conditions. Specifically, the platform will continue to support systems and components important to safety in a manner that will not affect their ability to perform their intended function, and the monorail structure will continue to support the shield doors such that they do not fall. This is demonstrated within these analyses by showing that the members of the platform and the monorail suspension structure meet alternate acceptance criteria.

Although these structures have been demonstrated capable of performing their intended functions, the supporting calculations identified that the original design basis margin of safety was reduced, which resulted in identification of an unreviewed safety question. Therefore, 10 CFR 50.59 requires that prior NRC review be obtained on the Updated Safety Analysis Report (USAR) changes which discuss the new analyses.

The results of the analyses serve as the justification for this request for amendment of the Perry Nuclear Power Plant (PNPP) licensing basis.

DESCRIPTION OF PROPOSED CHANGES

The USAR will be updated to describe the practice of opening the Drywell Personnel Air Lock Shield Doors during plant startup and shutdown while in Operational Conditions 1, 2, and 3.

The USAR will also be updated to discuss that the analyses of the 620'-6" platform and the monorail suspension structure utilize alternate analytical techniques and acceptance criteria than were utilized in the original design basis structural design. This discussion notes that it is only applicable for the sixth cycle of plant operation, ending with the sixth refueling outage.

Markups of the USAR pages are included as Attachment 3.

SYSTEM DESCRIPTION

The structures of concern for this evaluation are the 620'-6" elevation steel platform inside containment, the monorail suspension structure for the drywell personnel air lock shield doors, and the shield doors themselves. The platform supports plant equipment such as the Control Rod Drive Hydraulic Control Units and Instrumentation and Control panels. It supports piping associated with numerous plant operating systems (both safety and non-safety related). The primary safety function of this platform is to provide support for those systems, structures and components important to safety, allowing performance of their intended design functions. It also provides support to the monorail suspension structure, which in turn supports the shield doors.

A diagram is included as Figure 1, which illustrates the monorail suspension structure and the shield door construction.

The shield doors were designed to provide shielding from radiation streaming through the Drywell Personnel Airlock with the plant at power, to maintain doses of personnel working in containment ALARA (as low as reasonably achievable). The doors are not designed to serve a function for protection of the Drywell Personnel Air Lock doors following an accident. The shield doors also are not designed to provide a barrier against fission products, and opening them during power operation will have no effect on the accident source term. During plant startups and shutdowns, opening of these shield doors is necessary to provide access into the Drywell for various activities.

For Operational Conditions 4 and 5 the shield doors would not be subject to the loads from Safety Relief Valve (SRV) actuation or Loss Of Coolant Accident (LOCA) related loads.

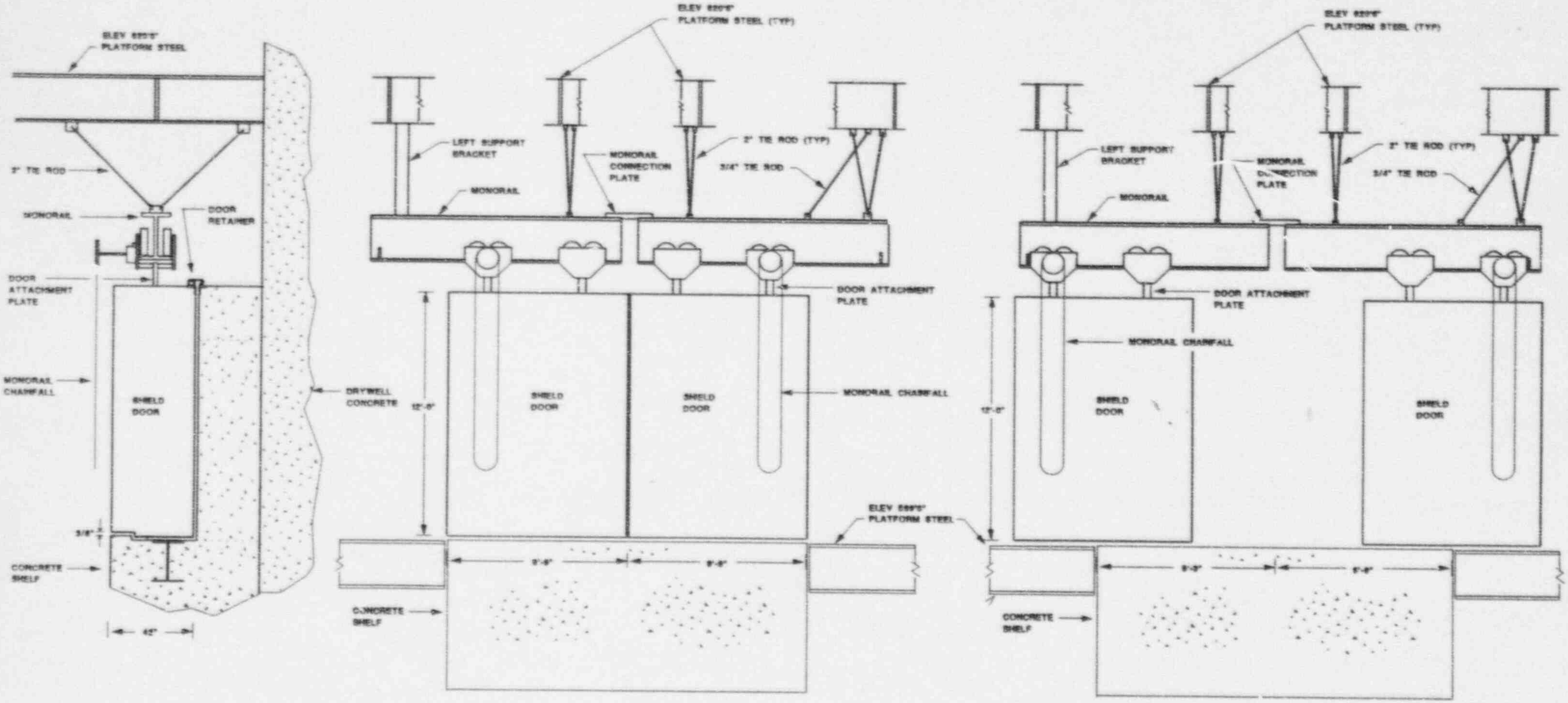
SAFETY ANALYSIS

Opening of the Drywell Personnel Air Lock Shield Doors in Operational Conditions 1, 2, and 3 when RPV pressure is greater than 250 psig can be demonstrated acceptable through review of the structural analyses, and through review of the radiological considerations.

The structural analyses must demonstrate that the doors do not present a falldown concern, and that the structural steel platform from which the doors are suspended will remain capable of supporting the various systems it serves. Several of these systems are addressed within the Technical Specifications, and their "OPERABILITY" must be ensured. Some of the associated systems, structures and components not specifically addressed within the Technical Specifications must remain "functional".

The radiological considerations need to demonstrate that personnel doses can still be maintained ALARA during periods when the doors are open, and that any increased radiation levels experienced outside of the open shield doors do not adversely impact qualification of equipment in the area.

DRYWELL AIRLOCK SHIELD DOORS



SHIELD DOORS CLOSED

SHIELD DOORS OPEN

FIGURE 1

Structural Analysis

The evaluation of this structural system requires definition of the loads applied to the structure (the demand prediction) and evaluation of the capability of the structure to resist these loads (the capacity evaluation). For the Drywell Personnel Airlock Shield Doors, the structural evaluation demand prediction was accomplished by applying various individual loadings to the structure and determining the forces, moments, and/or stresses resulting in the structure for the individual loadings. The capacity was then evaluated by combining the forces, moments, and/or stresses from the individual loads in a predefined method and comparing these combined forces, moments and/or stresses against an established acceptance criteria. For the structural evaluations performed on the platform and monorail suspension structure, two different demand/capacity criteria were considered. These are identified as the Design Basis Evaluation and the Functional Evaluation. Subsequent to the February 15, 1996 meeting with the NRC, an additional Functional Assessment of the monorail suspension structure was performed, which is described further below.

Design Basis Evaluation

The Design Basis Evaluation uses the Perry Nuclear Power Plant (PNPP) design basis input loadings (demand) and PNPP design basis acceptance criteria (capacity). The individual input loadings considered are:

D	Deadweight
SRV_I	Building filtered SRV discharge hydrodynamic inertial loading
OBE_I	Building filtered Operating Basis Earthquake (OBE) inertial loadings
SSE_I	Building filtered Safe Shutdown Earthquake (SSE) inertial loadings
$PS_{I/D}$	Pool swell water impact/drag loading
PS_{LP}	Pool swell water lateral pressure loadings
PS_I	Building filtered pool swell hydrodynamic inertial loadings
$CHUG_I$	Building filtered chugging hydrodynamic inertial loadings

The input seismic response spectra defined for the platform evaluation is an elastic amplified floor response spectra developed using Regulatory Guide 1.60 (Reference 1) input ground response spectra. This is a conservative design basis response spectra input for use in evaluation of the response of systems, structures and components to actual earthquake loadings.

The resulting input combination of applied individual loadings (left side of equation) and the acceptance criteria (right side of equation) are:

$D + SRV_I + OBE_I < S^{(b)}$

$D + SRV_I + SSE_I + PS_{I/D} + PS_{LP} + PS_I < 1.7S$ (Plastic section modulus permitted)[a]

$D + SRV_I + SSE_I + CHUG_I < 1.6S$ (Plastic section modulus permitted)[a]

a) When using the plastic section modulus, the combination of $D + SRV_I + SSE_I$ must meet the specified allowable using the elastic section modulus.

b) $S = 0.6 * F_y$, and F_y = Material yield stress

The loading combinations for this evaluation and the corresponding allowable stress criteria are consistent with Standard Review Plan Section 3.8.3 (Reference 2). These evaluations for the 620'-6" platform and the monorail suspension structure resulted in several members exceeding the design basis acceptance criteria for these loading combinations. For the platform, this consisted of significantly less than 1% of the platform members, and for the monorail structure, the exceedances existed on approximately 50% of the members.

Functional Evaluation

The Functional Evaluation criteria was established based on guidance in Generic Letter (GL) 91-18 (Reference 3) to assess the non-conforming condition for the shield doors. The primary concern in development of this criteria was to ensure that the Drywell Personnel Air Lock Shield Door support structure would remain functional, thereby preventing damage or failure of other systems, structures and components important to safety. This includes the 620'-6" platform from which the shield doors are ultimately supported. The other concern was to ensure the doors would remain in place (that they would not fall down). The ability to open or close the shield doors following a dynamic loading event is not required to meet this functional criteria.

The demand prediction methodology used in this criteria is identical to that used for the Design Basis Evaluation with the exception that for the monorail system and the shield doors, the input seismic (earthquake) response spectra was modified to be an inelastic (non-linear) response spectra in accordance with the methodology suggested by Newmark and Hall in NUREG/CR-0098 (Reference 4) and further recommended by Coats, et. al., in NUREG/CR-1161 (Reference 5).

As presented in NUREG/CR-1161, numerous observations of the actual performance of ductile steel structures subjected to seismic motions have demonstrated the capacity of structures to absorb and dissipate much energy when strained slightly. The energy absorption obtained from a linear elastic response analysis, using a conservative input response spectra and limiting the analytically predicted stresses in a structure to the design allowable stress, predicts only a small fraction of the total energy absorption capability of a structure. Unless corrected for this inelastic energy absorption capability, a linear elastic response analysis cannot account for the inelastic energy absorption capacity of a structure. This significantly under-predicts the structure's ability to withstand a design basis seismic event.

If the total inelastic response is low, the Newmark Inelastic Response-Spectrum Technique (Reference 4) adequately predicts the inelastic response of typical structures when compared to inelastic time-history analyses. The Newmark Inelastic Response-Spectrum Technique was developed by two renowned experts in the design and analysis of systems, structures and components for earthquake loadings, Nathan Newmark and William Hall. Further, it was reviewed and recommended for use by other seismic experts (including Robert Kennedy) in NUREG/CR-1161.

For the evaluation of existing facilities or structures that can deform inelastically to a moderate extent without unacceptable loss of function, both NUREG/CR-1161 and NUREG/CR-0098 recommend the use of the above-described Inelastic Response-Spectrum Technique with a ductility factor of 3.0.

This situation is applicable to the monorail and shield door system. Therefore this approach was applied to the design basis input elastic amplified floor response spectra used in the Functional Evaluation of the monorail and shield door structures and the generation of seismic loads applied to the platform from the monorail system. The seismic evaluation of the platform structure itself employed the design basis elastic amplified floor response spectra.

Two capacity evaluation criteria were developed and applied, one for the platform supporting structure and one for the monorail suspension structure and the airlock shield doors.

Platform Functional Evaluation

For the platform supporting structure, the load combinations applied were essentially the same as those used for the Design Basis Evaluation. This was deemed appropriate and prudent considering the necessary support function provided by the platform for systems, structures and components important to safety. The acceptance criteria was the same as the Design Basis Evaluation acceptance criteria with one exception. The allowable stress limit for the $D+SRV_I+OBE_I$ load case was increased from S (where $S = 0.6F_y$) to F_y.

This demand and acceptance criteria is summarized as follows:

$D + SRV_I + OBE_I < F_y$

$D + SRV_I + SSE_I + PS_{I/D} + PS_{LP} + PS_I < 1.7S$ (Plastic section modulus permitted)[a]

$D + SRV_I + SSE_I + CHUG_I < 1.6S$ (Plastic section modulus permitted)[a]

where $S = 0.6 * F_y$, and F_y = Material yield stress

a) When using the plastic section modulus, the combination of $D + SRV_I + SSE_I$ must meet the specified allowable using the elastic section modulus.

The platform was found to meet the above acceptance criteria for the Functional Evaluation.

Monorail Suspension Structure Functional Evaluation

For this evaluation of the shield doors and the monorail suspension structure, the concern is the potential falldown of the doors, and the loads which will be applied or passed on to the 620'-6" platform structure. Therefore, a more realistic, but appropriately conservative (these conservatisms are further described below), capacity evaluation was developed. The load case combinations were developed based on the considerations put forth in EPRI Report NP-6041-SL, Revision 1, Appendix K (Reference 6). Differences in these load combinations from the Design Basis Evaluation load combinations are:

1) Load combinations involving SRV loads were made by the square root of the sum of the squares (SRSS) as opposed to the simple summation of the individual loads.

2) Pool swell loads were not considered concurrent with the SSE loads.

The random simultaneous occurrence of an intermediate or large LOCA and an earthquake is extremely unlikely. The random occurrence of a large LOCA per NUREG/CR-4550 (Reference 7) is 10^{-4}/yr and the Perry SSE of 0.15g has an annual probability of 6.35×10^{-5} (Reference 8). Therefore, the probability of a random simultaneous SSE and LOCA is approximately 6.4×10^{-9}/yr. Also the piping ruggedness data in Appendix A of Reference 6 indicate that seismically induced intermediate or large LOCAs are not credible events. Hence, the hydrodynamic loads induced by these events (i.e., pool swell, chugging, etc.) need not be considered in conjunction with SSE seismic loadings in a Functional Evaluation. Earthquake induced failure of a small reactor coolant system pipe causing a small LOCA is considered to be a credible event. The hydrodynamic loads which result from a small LOCA are chugging and condensation-oscillation. Earthquake strong motion, especially for an eastern United States earthquake, would be expected to last for less than 10 to 15 seconds (Reference 6), while the significant chugging loads do not begin until 12 to 15 seconds after break initiation. Therefore, it is unlikely these hydrodynamic loadings will occur concurrently with the significant seismic loads.

Considering the relatively short time duration of the peak SRV loading, usually only a few seconds, and its random nature, it is unlikely that the peak SRV and earthquake responses will occur concurrently (Reference 6). While concurrent occurrence of SRV and seismic loadings is a credible event for a Functional Evaluation, it is overly conservative (as used in the design basis) to take the combined loading as the absolute sum of the earthquake and SRV maximum responses. A recent study (Reference 9), cited by Reference 6, using probabilistic load combination methodology has shown that taking the square root of the sum of the squares (SRSS) of the median maximum earthquake and the SRV responses gives a slightly conservative estimate for the median maximum combined response compared to a time-history evaluation. Extensive simulation studies described in General Electric NEDO-24010-03 (Reference 10), cited by Reference 6, have demonstrated the SRSS of the earthquake and SRV 84 percent non-exceedance probability associated with the SRSS combined response is approximately equal to the non-exceedance probability associated with the individual responses. Therefore, for the Functional Evaluation, these loads can be combined on a SRSS basis and still retain conservatism.

The monorail structure Functional Evaluation capacity acceptance criteria were modified from the design basis acceptance criteria in three areas:

1) The allowable stress limit for the load combination involving the OBE was raised from $0.6F_y$ to F_y.

2) For the accident load cases (which utilize the plastic section modulus), only 90 percent of the plastic section modulus was applied in the evaluation.

3) If a capacity stress acceptance criteria was exceeded, the component was still considered functional if the maximum principal strain was less than 20 percent of the ultimate strain of the material.

The resulting demand and acceptance criteria may be summarized as follows:

$D + (SRV_I^2 + OBE_I^2)^{1/2} < F_y$
$D + (SRV_I^2 + SSE_I^2)^{1/2} < 1.7S$ (90% plastic section modulus applied)
$D + PS_{I/D} + PS_{LP} < 1.7S$ (90% plastic section modulus applied)
$D + PS_I < 1.7S$ (90% plastic section modulus applied)
$D + SRV_I + CHUG_I < 1.6S$ (90% plastic section modulus applied)

where $S = 0.6 * F_y$ and F_y = Material yield stress

For each of these analyzed load cases, the monorail suspension structure was found to meet the acceptance criteria for the Functional Evaluation.

Additional Functional Assessment of the Monorail System

The analyses discussed in the previous section demonstrated the functionality of both the platform and the monorail system. To provide an additional level of assurance, engineering assessment of the consequences of assumed (non-mechanistic) failures in the monorail support system was also conducted. First it was assumed that the most highly stressed member, the 3/4 inch tie rod, instantaneously failed as a result of an SSE event. The analysis indicated that should this occur, it would result in significant load transfer to the left hand door support bracket. This load transfer resulted in elastically predicted stresses in this left hand bracket which exceeded the Monorail Functional Evaluation acceptance criteria discussed above.

A further assessment was then performed to confirm the reserve margin of the two inch tie rods. In order to do this, both the 3/4 inch tie rod and the left hand bracket were assumed to simultaneously and instantaneously fail as a result of an SSE event (such a failure is not a credible occurrence). Even in this hypothetical situation, it was demonstrated that the right hand door would remain supported by the two sets of 2 inch tie rods, and the left door would remain supported by one set of 2 inch tie rods, the retainer on the top of the door, and the steel platform at elevation 599'-6". Therefore it was demonstrated that neither door will fall down during an SSE event even if these non-credible door support failures are assumed to occur.

These assessments are conservative for several reasons including:

(1) The results of the initial functional evaluation indicate the components that are postulated to fail have significant margin against such failure; therefore, this type of a failure is not credible.

(2) The failures were assumed to be instantaneous, however, both the 3/4 inch tie rod and the left hand bracket are made of highly ductile carbon steel which will undergo significant inelastic response and will fail by ductile exhaustion, which is not an instantaneous failure mode. Further, this type of failure will absorb significant strain energy which would greatly reduce the system response to an SSE event.

(3) For the second postulated failure case (both the 3/4 inch tie rod and left hand bracket failure), design basis elastic amplified floor response spectra were used in the analysis, i.e., no inelastic response spectra were used.

Structural Analysis Conclusion

The first set of functional evaluation analyses performed using the methodology described above determined that although isolated locations in the 620'-6" platform and portions of the monorail suspension structure exceed the design basis acceptance criteria, the structures remain functional; i.e., continue to provide support to systems, structures and components important to safety. The functional evaluations, although less conservative than design, still contain sufficient conservatism to provide more than adequate assurance of the continued functionality of the platform and monorail suspension structure. Among these conservatisms are:

1) The building-filtered conservative design basis LOCA-related (pool swell, chugging, etc.) spectra were used as input to this analysis; no inelastic considerations were applied to these spectra, i.e., design basis spectra were used.

2) Both the platform and the monorail/door system analyses were conducted as equivalent static linear elastic analyses with conservative boundary conditions. Lower structural loadings would be predicted if non-linear time history analyses were utilized.

3) The structures analyzed are highly ductile steel structures with no components which would exhibit a brittle failure mode. Therefore, in the unlikely event of the occurrence of accident loadings when the doors are in the open position during Operational Conditions 1, 2 or 3, the structures would exhibit a higher level of ductility and energy absorption than assumed in this analysis.

4) Only isolated application was made of the maximum principal strain acceptance criteria.

The additional functional assessments performed also showed that even if the 3/4 inch tie rod and the left support bracket are assumed to be failed, neither shield door will fall down. The right shield door will remain suspended by the 2 inch tie rods. The left shield door will be held upright by the 2 inch tie rods and the 599' steel platform.

Radiation Level Considerations

The USAR contains several statements pertinent to the shield doors and radiation levels in the surrounding area. The radiological purpose of the shield doors is to mitigate radiation streaming from the Drywell through the Personnel Airlock with the plant at power (USAR Section 12.3.2.2.1). USAR Section 12.4.2 reads, "It is anticipated that the general radiation levels for each [radiation] zone will be less than the design values stated for the zone, although isolated higher levels will exist in certain areas within the zone." Finally, USAR Section 12.1.2.3.e under "Design Guidance Given to Individual Designers" reads "Make the design sensitive to the expected procedures of plant personnel under normal operational and anticipated occurrences."

The shield doors are located on the 599' elevation of the Containment at about 105° azimuth. USAR Section 12.3, Radiation Zone Plan B (Figure 12.3-2) delineates this area as a radiation Zone II which corresponds to a maximum dose rate of 2.5 mrem/hr. This value is based on 100 percent reactor power and having the shield doors in the closed position.

Radiological history, specifically Reference 11, Table 4.1, measured radiation levels outside the Drywell Personnel Airlock at 45 percent reactor power with the shield doors open to be 10 mrem/hr neutron and 10 mrem/hr gamma. Therefore, the containment at elevation 599' immediately adjacent to the Drywell Personnel Airlock Shield Doors would have an expected dose of 20 mrem/hr (gamma + neutron) with the shield doors open and the reactor at 45 percent power. These increased radiation levels adjacent to the open Drywell Shield Doors will not affect radiation levels outside the containment Shield Building.

The USAR passages noted above implicitly indicate that a gradient of radiation levels could be expected within the radiation zones and that operational activities would be taking place during plant operation. The area outside of the Drywell Shield Doors is an example of a radiation zone which experiences such gradients and operational activities.

The classification of radiation zones is such that the doses are below the limits of 10CFR20 and will be as low as reasonably achievable (ALARA), predicated on the amount of time the area will have to be occupied during normal operations and anticipated occurrences. With respect to having the shield doors open during plant operation, the minor increase in dose rates will be localized, work will be controlled and the doses can be maintained ALARA in accordance with established radiological procedures.

USAR Section 12.6, Post Accident Radiation Zones Plan B (Figure 12.6-2) does not differentiate between the Containment and the Drywell with respect to post-accident radiation levels. The areas inside and outside of the Shield Doors are both classified as Radiation Zone VIII. This corresponds to a dose rate of greater than 5,000 R/hr post accident. The post accident radiation level classification outside the Drywell Personnel Airlock would not be revised by having the shield doors open. Also, the shield doors do not provide a barrier against fission product release and opening them during power operation will have no effect on the accident source term.

A review of the Equipment Qualification Equipment List was performed for equipment located in zone CT-2. A single hydrogen igniter was identified as potentially being exposed to the increased dose rates during the brief periods of time that the Shield Doors are open. The increased dose from plant startups and shutdowns will have no effect on the qualification of this equipment.

The integrated dose for this area of the containment over the 180 days following a LOCA is 3.6×10^7 Rads (Reference 12). Equipment qualification is performed to address both expected normal operational conditions and post-accident conditions, plus an additional margin of 10% of the accident dose. The integrated dose experienced by increasing the dose rate from 2.5 mrem/hr to 20 mrem/hr during the brief periods of plant startup and shutdown over the life of the plant is a small fraction of the LOCA integrated dose margin, and will have no effect on the qualification and operation of the hydrogen igniter in the area.

Radiological Conclusion

There are no radiological concerns with having the shield doors open under procedural controls when the plant is in Operational Conditions 1, 2 and 3.

References

1. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, December 1973.

2. NUREG-0800, "Standard Review Plan."

3. Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and On Operability," November 7, 1991.

4. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," May 1978.

5. NUREG/CR-1161 "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria", May 1980.

6. EPRI Report NP-6041-SL, "Nuclear Power Plant Seismic Margin," Revision 1, August 1991.

7. NUREG/CR-4550, "Analysis of Core Damage Frequency: Internal Events Methodology," Volume 1, Revision 1, January 1990.

8. EPRI Report NP-6395-D, "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," April 1989.

9. R.H. Sues, M.K. Ravindra, C.S. Putcha, and R.H. Kincaid, "Combination of Seismic and Hydrodynamic Loads for the LaSalle County Station Probabilistic Risk Assessment", October 1985.

10. General Electric NEDO-24010-03 "Additional Demonstration of Statistical Basis for the Square-Root-of-Sum-of-the-Square Method", August 1979.

11. "Neutron and Gamma Dose and Energy Spectral Measurements Inside the Perry Unit 1 Reactor Containment Building," Battelle Pacific Northwest Laboratories, December 17, 1987.

12. Environmental Drawing B-022-021.

ENVIRONMENTAL CONSIDERATION

The proposed license amendment has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above and in Attachment 2, the proposed change does not involve a significant hazards consideration, does not increase the types and amounts of effluents that may be released offsite, and does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, it has been concluded that the proposed USAR change request meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

STATEMENT OF EXIGENT CIRCUMSTANCES

The general issue of whether opening the shield doors at power was acceptable was first raised at the end of the fourth refueling outage. Documentation of the interim acceptability of opening the doors under administrative controls was developed at that time, with plans for subsequent analysis to resolve long-term acceptability issues. Preliminary analyses indicated acceptable resolution under the provisions of 10 CFR 50.59a and b. Analysis of the issue continued throughout 1995 to determine the final long-term resolution. Based on continued examination of the issues involved, the basis for some of the assumptions in the calculations was questioned in early January 1996. Also in early January, an outside specialist was brought in to assist in the evaluations. The identification that revised engineering analyses would result in an unreviewed safety question and would therefore require processing of a license amendment was made at a Plant Operations Review Committee meeting on February 9, 1996.

In parallel with completion of the engineering analyses and initial development of a license amendment request letter, a meeting was held with the NRC staff on February 15, 1996. Information from that meeting is incorporated into this license amendment request. This request has been processed in a timely fashion, based on completion of the engineering analyses and the necessary reviews prior to submittal.

It is requested that the Federal Register Notice be published with provisions for exigent processing. This will help to ensure that an adequate public review period is provided, while also ensuring that processing of the amendment will not be the sole item restraining plant restart from the current refueling outage, which is currently scheduled for March 25 1996. Such a restraint would result in a costly extension to the outage with no corresponding benefit to safety, since analyses have shown that plant systems, structures and components will remain capable of performing design basis accident safety functions. As noted above, this request has been submitted in a timely fashion following final completion of the engineering analyses which had led to the an unreviewed safety question determination. Therefore, publishing of this request under the exigent provisions of 10 CFR 50.91(a)(6) cannot be avoided.

Communication of outage schedule progress will continue with the NRC Project Manager for PNPP. To ensure that final processing of the associated paperwork does not hold up return of the plant to Operational Condition 2, issuance of the amendment at least three days prior to the scheduled startup date is requested.

SIGNIFICANT HAZARDS CONSIDERATION

This proposed license amendment would authorize a revision to the Perry Nuclear Power Plant Updated Safety Analysis Report (USAR) to describe the practice of opening the Drywell Personnel Air Lock Shield Doors during plant startup and shutdown, while in Operational Conditions 1, 2, and 3. The USAR will also be updated to discuss that the structural analyses of the 620'-6" platform and the monorail suspension structure (which support the Shield Doors) utilize alternate analytical techniques and acceptance criteria than were utilized in the original design basis structural design. This discussion notes that the alternate analyses are only applicable for the sixth cycle of plant operation, ending with the sixth refueling outage. These USAR changes have been identified as requiring NRC review prior to incorporation, in accordance with 10 CFR 50.59(c). This amendment does not affect pages in the Operating License or the Technical Specifications; it involves only the USAR changes.

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in the Commissions regulations, 10 CFR 50.92. This regulation states that a proposed amendment involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed change has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   An assessment was made of functionality given occurrence of the loads imposed on the shield doors. This assessment involves the 620'-6" steel platform, the monorail suspension structure and the shield doors themselves. Although certain structural members of the 620'-6" platform exceed the design basis acceptance criteria, these members were found to be acceptable when reviewed for functionality using alternate acceptance criteria. This demonstrates that the various supported systems and components that are important to safety will remain OPERABLE (for Technical Specification systems) or functional (for non-Technical Specification systems, structures and components). Even if the 3/4 inch tie rod (which provides lateral stability) and the left support bracket (a vertical load bearing member) were assumed to be failed, the shield doors would remain in a upright position and not fall. The monorail suspension structure and shield doors do not provide support to other systems. There are no interferences, and opening the shield doors has no effect on other systems. Therefore, there will be no increase in the probability of an accident due to the monorail suspension structure or shield doors, with the doors placed in the open position during Operational Conditions 1, 2, and 3.

   The primary purpose of the shield doors is to mitigate radiation streaming from the Drywell through the Personnel Airlock into the adjacent areas of the Containment, to maintain doses to personnel working inside containment ALARA (as low as

reasonably achievable). Opening the doors during power operation will have no effect on the postulated accident source term, and the shield doors do not provide a barrier against fission products. Therefore, allowing the shield doors to be opened during plant startup and shutdown while in Operational Conditions 1, 2, or 3 will also not increase the consequences of an accident previously evaluated in the USAR.

Based on the above, the proposed changes do not significantly increase the probability or the consequences of any accident previously evaluated.

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

   The proposed changes do not involve physical modifications to the plant. There are no interferences with piping or other system components when the doors are placed in the open position during Operational Conditions 1, 2, or 3. Given the initiating events postulated for the various load combinations, none result in a new type of accident. The increase in radiation levels in the immediate vicinity of the open shield doors with the plant at power was verified to have no effect on the qualification and operation of systems, structures, or components important to safety. Since the platform and the monorail suspension structure will continue to provide support for the shield doors, i.e., the doors will not fall from the support structure, no new initiators of accidents are introduced.

   The 620'-6" platform will continue to function with the shield doors open. The equipment supported by the platform will continue to perform their safety related design functions. Although components of the platform and the monorail suspension structure exceed design basis acceptance criteria, analyses have shown that, based on a functional assessment, the monorail suspension structure will continue to function and the doors will remain upright. With no additional loads imposed on other equipment and the continued functioning of the monorail suspension structure, there will be no "different" accidents, since there will be no change, degradation, or prevention of actions described or assumed in any analyzed accident. The radiological consequences and the fission product barriers are not affected.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The NRC has accepted the Perry structural steel design (Safety Evaluation Report, NUREG-0887) based on the Structural Acceptance Criteria in Standard Review Plan Section 3.8.3. Analyses were subsequently performed considering the shield doors to be in the open position during plant operation. Several members and connections of the 620'-6" platform and monorail suspension structure exceed the allowable stresses based on those acceptance criteria, and therefore a determination was made under the provisions of 10 CFR 50.59 that there was a slight reduction in the margin of safety. However, as described below, the proposed change has been reviewed and determined not to involve a significant reduction in a margin of safety, as discussed in 10 CFR 50.92.

Those members which had exceeded the design basis allowables were found to meet the Functional Evaluation acceptance criteria. This demonstrated functionality of the platform and the monorail structure; i.e., the platform would continue to support systems, structures, and components (SSCs) important to safety, the SSCs would remain functional, and the shield doors would not fall down. Analytical conservatisms within the Functional Evaluations remain to provide adequate assurance of continued function of the affected SSCs.

Placing the shield doors in the open position during Operational Conditions 1, 2, or 3 is not inconsistent with the guidelines of the Technical Specifications for High Radiation Areas and the Radiation Protection Program. The open shield doors will not affect radiological limiting conditions or action limits for plant effluents as described in the Technical Specifications or Operating License. It does not affect the radiological bases as described in the Technical Specifications or Operating License. It does not affect the margin of radiological safety. The offsite radiation doses to members of the public are not increased.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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