June 26, 1986

Docket No. 50-410

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Mr. B. G. Hooten
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Dear Mr. Hooten:

Subject: Draft of Section 6 of Safety Evaluation Report Supplement 3

Enclosed for your information is a draft of Section 6 of Safety Evaluation Report (SER) Supplement 3. This section contains the draft SER on your exemption request on downcomers. SER Supplement 3 is expected to be released shortly.

Sincerely,

/S/

Mary F. Haughey, Project Manager
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
As stated

cc: See next page

M Haughey

BWD-3:DBL	LA:BWD-3:DBL	D:BWD-3:DBL
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6/26/86	6/ /86	6/26/86

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6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.3 Short-Term Pressure Response

The drywell and suppression chamber design pressure is 45 psig. In FSAR Amendment 21, the applicant provided the results of a sensitivity analysis based on a change in the number of downcomers from 123, which was used in the original FSAR analysis, to 121, which reflects the as-built plant condition after 2 downcomers were blocked off. The 2 downcomers were eliminated in a design modification to accommodate quenchers which were installed on the RHR heat exchanger relief valve discharge lines. The results of the analysis show that the drywell peak pressure increased nominally from 39.75 psig to 39.86 psig.

The staff has performed a comparison of the NMP-2 values of peak short-term drywell and suppression chamber pressures with a group of plants with the Mark II containment for which the staff performed satisfactory confirmatory analyses using the CONTEMPT LT/028 computer code. This comparison is shown in Table 6.1. Table 6.2 contains an additional comparison of selected NMP-2 containment characteristics with a similar Mark II plant designed by the same architect-engineer, Stone & Webster Engineering. This comparison indicates that the difference in calculated drywell peak pressures between 39.9 psig for NMP-2 and 41.9 psig for Shoreham is 5%. Similarly the difference for peak pressures in the suppression chamber is 12%. These differences are within an acceptable range, given the slight variations in plant parameters shown in Table 6.2. Since the peak pressures calculated for Shoreham have been verified by the staff as being accurate given the postulated accident assumptions, the staff concludes that, by comparison, the values submitted by the applicant for the short-term analysis of drywell and suppression chamber peak pressures resulting from a double-ended rupture of the recirculation line is acceptable.

6.2.1.7 Pool Dynamic Analyses

6.2.1.7.3 Plant-Unique Loads

The following subsections have the same numbers and titles as those in the SER Section 6.2.1.7.3.

(1) Pool Swell Loads

The staff stated in the SER that it had requested the applicant to provide comparisons to demonstrate the conservatisms of the results obtained from LOCTVS and Stone & Webster Engineering Corp. (SWEC) computer codes to those results obtained from the General Electric (GE) PSAM code and GE Topical Report NEDO-10320 and Appendix B of GE report NEDO-20533.

In Amendment 21 to the design assessment report (DAR), the applicant provided the requested information. On the basis of its review of the information submitted by the applicant, the staff concludes that, except for bubble pressures, the comparison shows favorable agreement and, therefore, is acceptable.

With respect to the bubble pressure prediction, the LOCTVS result is 3.5 psi less than bubble pressures calculated by PSAM. The applicant indicated that it has evaluated all safety-related components and structures that are affected by the higher air bubble load and has concluded that such components and structures can withstand the additional 3.5 psi. The structural capability of the NMP-2 downcomers to withstand all loss-of-coolant accident (LOCA) and safety/relief valve (SRV) hydrodynamic loads is discussed in Section 6.2.1.7.4 of this supplement.

(4) Loads on Submerged Boundaries

In the SER, the staff stated that information is needed about the magnitude of the pool swell bounding loads inside and outside the pedestal. In Amendment 21 to the DAR, the applicant stated that a bounding analysis was done to estimate the differential pressure loading across the pedestal wall. This differential pressure was obtained by modifying the containment value by the ratio of pool surface area per downcomer within this area per downcomer in the main pool. On the basis of its review of the applicant's submittal, the staff finds that the applicant's approach to assessing this load is acceptable.

(5) Multi-event Lateral Load

In the SER, the staff indicated that additional information is needed to define how the multi-event lateral load is applied to the diaphragm floor. In a letter dated September 16, 1985, the applicant stated that the diaphragm floor is designed to withstand the moment and shears caused by the multi-event lateral loads at the junction of the downcomers with the drywell floor. The individual multi-event lateral loads are applied simultaneously and in the same direction at all downcomers and, therefore, are added algebraically. On the basis of its review of the applicant's submittal, the staff concludes that this approach is bounding and, therefore, is acceptable.

(6) Condensation Oscillation Loads Inside the Pedestal

In the SER, the staff indicated that the applicant's proposal to use the same time-history segments specified in the Mark II generic condensation oscillation (CO) load definition for the annular pool region, multiplied by 1.25, as the load definition for the cylindrical pool to be acceptable pending approval of the methodology regarding the SWEC computer program.

In Amendment 21 to the DAR, the applicant indicated that the 1.25 multiplier was determined on the basis of the SWEC computer code results that calculated the differential pressure between both regions (cylindrical and annular) of the suppression pool by applying the CO source between 0 and 30 Hz. The average pressure amplitude ratio between the inner and the outer pool varied between 1.04 and 1.24. Therefore, the use of the 1.25 multiplier to define the CO inside the pedestal region is conservative. On the basis of its review of the applicant's submittal, the staff concludes that the use of the 1.25 multiplier is acceptable; the staff now considers this issue closed.

(8) Steam Condensation Submerged Drag Loads

In the SER, the staff indicated that it needed additional information before it could conclude on the acceptability of this load. Reexamination of the DAR revealed that the applicant is using the same methodology that was previously reviewed and found acceptable in the Shoreham SER. Therefore, this issue is closed.

(9) Pool Temperature Limit

(c) Bulk-to-Local Temperature Differences

In the SER, the staff stated that it would require the applicant to perform confirmatory calculations by using data from comprehensive SRV inplant tests, to demonstrate that the maximum local pool temperature specifications will not be exceeded. In a letter dated September 30, 1985, the applicant provided a comparison of the NMP-2 pool geometry to the LaSalle pool geometry, where the SRV inplant tests were conducted. The applicant also provided a comparison between predictions of the LaSalle pool temperature to SRV actuation transient to those measured during an extended blowdown test. On the basis of its review of the information provided by the applicant, the staff concluded that the NMP-2 and LaSalle geometries are very similar. The staff has also concluded that the predicted temperature transients compare favorably with the measured values. Therefore, this issue is now resolved and the use of a local-to-bulk temperature difference of 10°F is acceptable.

(d) Single-Failure Analysis

The applicant stated that the normal shutdown cooling mode could be unavailable as a result of a failure of the suction line isolation valve inside the drywell. For this case, alternate shutdown cooling could be achieved by pumping suppression pool water into the reactor vessel through the residual heat removal (RHR) system and returning water to the pool through manually opened SRVs. The staff finds this alternate mode of removing the decay heat to be acceptable.

(10) Quencher Air Clearing Load

As stated in the SER, the applicant indicated that the acceptance criteria for the T-quencher as set forth in NUREG-0802 is utilized in the design of Nine Mile Point, Unit 2, except for the criterion on frequency range. The applicant concluded that a frequency range of 3 to 9 Hz instead of the staff-recommended value of 3 to 11 Hz, is conservative for NMP-2. To support this conclusion, the applicant provided comparisons of the response spectrum of an extrapolated Karlstein test trace 21.1 which has the highest dominant frequency with the NMP-2 design load. The Karlstein trace was modified by an amplitude reduction factor and dominant frequency factor to account for NMP-2 specific parameters. The comparison indicated that the NMP-2 specification is conservative and, therefore, acceptable. The staff concludes that this issue has been resolved.

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(11) SRV Submerged Structure Load

The SRV air bubble submerged structure drag loads are computed on the basis of a bubble pressure source strength of 1.5 times the Kraftwerk Union (KWU) specification. Since this pressure has been found acceptable for the boundary load specification, its use of submerged structure drag load is also acceptable.

The flow pattern of the fluid about the structure is calculated using the Rayleigh bubble equation for a spherical bubble that uses the pressure field outlined above. The applicant included the NRC-recommended 1.1 factor for bubble asymmetry to the fluid velocity and acceleration. Interference effects of adjacent structures are accounted for in calculating the acceleration drag coefficient in accordance with NUREG-0487, Supplement 1.

The applicant has presented a comparison between the above-described methodology and the previously approved KWU methodology. The results of this comparison indicate that the downcomer responses are within 1% of each other. On the basis of the above discussion, the staff concludes that the use of the Rayleigh bubble equation approach produces equivalent results to the KWU methodology previously found acceptable and, therefore, is acceptable.

In calculating the effective submerged structure drag load for NMP-2, the velocity and acceleration drag terms are modified to include the relative velocity and acceleration of the fluid and the downcomer at each instant in time. This issue is now considered resolved.

(12) SRV Inplant Test

In NUREG-0763, "Guidelines for Confirmatory Inplant Test of Safety-Relief Valve Discharge of BWR Plants," the staff stated in part that inplant tests will be required for those plants in which parameters potentially affecting SRV-discharge performance are deemed to be plant unique. In Section 4 of the report, the staff listed five conditions which, if satisfied (i.e., if applicants are able to demonstrate that the conditions in their plant are similar to the conditions in plants previously tested), will obviate the need for any new tests.

In its letter dated September 16, 1985, the applicant submitted the requested evaluation and justification. The applicant concluded that inplant SRV testing is not required for NMP-2 since the comparison of key parameters for each of the five conditions demonstrates that discharge conditions are sufficiently similar between NMP-2 and LaSalle, where SRV inplant tests were performed. The applicant also stated that other parameters, which differ slightly (such as soil shear wave velocity), do not have a significant effect on SRV loading.

On the basis of its evaluation, the staff concludes that SRV inplant tests are not required for NMP-2.

(13) Wetwell-Drywell Vacuum Breakers

In response to the staff's concern identified in the SER, the applicant provided the following information.

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The vacuum breakers are located inside the drywell and are mounted in piping that connects the drywell to the suppression chamber. Since the vacuum breakers are not mounted on downcomers, they are removed from the direct effects of chugging transients. The vacuum breakers' valves are of the same size and of similar design as the LaSalle valves which have undergone modification and testing to ensure that they can withstand the pool swell phenomena. The staff had previously reviewed and found acceptable the LaSalle vacuum breaker valve tests and design modification. Since the modified valve's design has been incorporated in the NMP-2 plants, and since similarly modified valves have undergone tests at the expected opening and closing velocities for NMP-2, the staff concludes that the design of the vacuum breaker valves for NMP-2 is acceptable and can accommodate the effects of pool swell impact loading following a design-basis LOCA.

(15) Secondary Loads

Following the pool swell process, continued flow through the vent system generates random pool motion. This pool motion creates waves which may impinge upon the downcomers. The staff has determined generically that these loads are considered to be secondary by virtue of their low magnitude when compared with the primary loads discussed in the previous section. However, since the NMP-2 downcomers are unbraced and have a natural frequency of about 0.89 Hz, the random pool motion discussed above may exert loads on the downcomers at a frequency corresponding to the downcomers' natural frequency and consequently amplify these loads. Therefore, the generic conclusion that these loads are secondary by virtue of their low magnitude might not be applicable to NMP-2.

In a meeting on December 20, 1985, the applicant was requested to assess the potential of secondary load of becoming significant load due to resonance. This issue is addressed in the "Load and Load Applications" section of Section 6.2.1.7.4 of this supplement.

6.2.1.7.4 Downcomer Design

The NMP-2 containment design utilizes the BWR Mark II concept of over-under pressure suppression with multiple downcomers (121) connecting the drywell to the pressure-suppression chambers. These downcomers channel the steam resulting from a loss-of-coolant accident (LOCA) from the drywell into the suppression pool.

The NMP-2 downcomers are made of type 304 stainless steel (SA 312-304) pipes, 24 inches in diameter, 3/8 inch in thickness, and 30 to 45 feet in length. Approximately 11 feet of each downcomer is submerged below the high water level of the suppression pool. These pipes are designed to ASME Boiler and Pressure Vessel Code (hereinafter referred to as the Code) rules for Class 2 piping, in accordance with staff criteria on load combinations specified in Standard Review Plan (SRP) Section 3.9.2 and in NUREG-0484, Revision 1, "Methodology for Combining Dynamic Responses."

The downcomer design at NMP-2 is unique in that it does not provide lateral supports at the free ends of downcomers; i.e., at the bottom, the downcomers are free to move in the plane perpendicular to downcomers. All other domestic Mark II plants have employed a bracing system to tie all downcomers together at the bottom to prevent free movement of an individual downcomer pipe. The

design of the unbraced downcomers at NMP-2 is very "soft," i.e., the natural frequency of the fundamental mode is 1.0 to 2.0 Hz. The diameter-to-thickness ratio (D/t) is 64; this exceeds the value of 50 that is generally viewed as the upper limit of the applicability of design procedures for nuclear piping specified in the ASME Code. In a "soft" structure, the deformation is expected to be large; this can invalidate the basic assumptions for performing a linear-elastic structural system analysis. Although there are no clear definitions of "large" deformations (e.g., excessive ovalization and flexure) in the theory, the range of uncertainties in the analysis is expected to become larger and results of the analyses become less reliable as deformation increases.

Because the unbraced downcomer design is unique and because of the concern over the potential loss of structural stability before reaching the design limits, the staff requested the detailed design calculations on the NMP-2 downcomers. The staff performed a preliminary review of the design calculations and in a meeting in Bethesda, Maryland, on December 20, 1985, stated that the design appeared inadequate because the unbraced design did not meet some of the licensing criteria established by the NRC and accepted by the applicant. In the meeting, the staff also presented its specific concerns relating to the applicant's analysis of the downcomer design. Subsequent to the meeting, the downcomer analysis that was discussed at the meeting was submitted in a letter dated December 31, 1985, from C. V. Mangan to E. Adensam. After performing a detailed review of that analysis, a draft safety evaluation report was transmitted to the applicant by letter dated January 8, 1986. On January 15, 1986, a meeting was again held in Bethesda, Maryland, between the staff, the applicant, and the applicant's consultants: Stone & Webster Engineering Corp., General Electric Corp., Stevenson & Associates, and Management Analysis Company. After reviewing the staff's concerns described in the staff's January 8, 1986, letter, the applicant reanalyzed the NMP-2 downcomers on the following bases:

- A time-history analysis was made for the seismic loads.
- Chugging loads were revised according to NUREG-0808.
- Allowable stresses were revised on the basis of the temperatures in the NMP-2 wetwell.
- Damping values were revised.
- The method for combining loads was revised.
- A rigorous ASME Class 1 fatigue reanalysis was completed that superseded the original one presented in the applicant's letters of December 31, 1985, in which the stress intensification factor was not properly considered.

The applicant has also indicated that snap-back tests with deflections of 1.2 and 3 inches were performed to justify the higher damping factors used in the reanalysis. The details of the above reanalysis were submitted by letters dated January 23 and 24, 1986.

On January 24, 1986, the staff met with its consultants to discuss the adequacy of NMP-2 downcomer design in the context of the reanalysis submitted by the applicant on January 23. After a detailed discussion, the staff and the consultants concluded that: (1) the unbraced downcomer design at NMP-2 met the

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licensing criteria for upset and emergency conditions but met the criteria marginally; and (2) the applicant had not adequately demonstrated the design adequacy for the faulted condition. These conclusions along with staff recommendations for the possible resolution were furnished to the applicant by letter dated January 31, 1986. In the material that follows, the staff's specific concerns about the design adequacy of NMP-2 downcomers, the recommendations for resolution, and the bases for the recommendations are discussed.

Design Philosophy

The downcomers are essential elements of the suppression-type containment system and, strictly speaking, are not a piping system. The downcomers channel the steam that can result from a loss-of-coolant accident (LOCA) or other accidents from the drywell into the suppression pool. In fulfilling this suppression function, the downcomers will be subjected to flow-induced and pool hydrodynamic loads in addition to other loads that are considered in the design of structures inside the containment. Both the flow-induced and pool hydrodynamic loads can be influenced by the structural characteristics of the downcomers. These loads have been determined from model testing of a "rigid" downcomer. Therefore, the staff believes that the use of rigid downcomers would obviate the potential problems of resonance, buckling (loss of geometric stability), low-cycle fatigue, and functional capability.

Even though the applicant has demonstrated that the design meets the Code criteria, the applicant has not used an adequate safety factor to accommodate the uncertainties (for example, those associated with the definition of the loading, material properties, imperfections in the geometrical configuration, and method of analysis), since some design convervatisms have been reduced in the reanalysis. In a letter dated January 24, 1986 (from C. V. Mangan to E. Adensam), the applicant noted that Stevenson & Associates observed that "there may be no inherent margin in failure mechanism formation between multi-supported statically indeterminate piping systems and statically determinate simply supported or cantilever supported systems." The staff believes this observation is basically irrelevant because in installing a bracing system connecting adjacent downcomers, thus resulting in a highly redundant (statically indeterminate) space frame, the structural capability of the downcomers would be greatly enhanced. The letter of January 24 further noted that a cantilevered downcomer could be visualized as a pendulum that would be stable under dead and transient loads. If the downcomers act as visualized in the LOCA case, their behavior would be unpredictable and the displacements could be so large as to eventually lead to collapse or break, resulting in functional impairment of the downcomers. The applicant should either demonstrate that this failure mechanism could not occur or should design the downcomer to prevent it from occurring.

Loads and Load Applications

In the resolution of Unresolved Safety Issue (USI) A-8, "Mark II Containment Pool Dynamic Loads," the staff and its consultants evaluated and approved the bases for concluding that certain loads were secondary by virtue of their low magnitude and, therefore, were negligible. These secondary loads included water sloshing during and after the pool swell, seismic sloshing, and fluid/structural interactions. These conclusions were based on results of scale-model tests of pool swell, the chugging phenomenon, and pool response to SRV discharges.

The dynamic characteristics of downcomers were not considered in the modeling and, therefore, possible resonance effects were also not considered. Also, the single downcomer in the test chamber was supported laterally. Therefore, the conclusion that these loads were secondary and negligible may not be applicable to NMP-2 unbraced downcomer design.

In a meeting held on December 20, 1985, the applicant was requested to assess the potential of secondary loads being amplified to become significant as a result of resonance. The applicant reviewed all secondary loads as identified in NUREG-0487 and -0808. In this new light, only two loads were found to be cyclic in nature and, therefore, potentially susceptible to resonance effects: they are seismic sloshing and post-pool-swell loads. The annulus pool seismic sloshing frequency was estimated by the applicant to be 0.13 Hz, which is far from the downcomer resonance frequency of 1.55 Hz. Because of this wide separation, the applicant has concluded that resonance will not occur. The staff concurs with this conclusion.

With respect to post-pool-swell loads, the test data base was reviewed by the applicant, who concluded, and the staff concurs, that water fallback will not effectively excite the sloshing waves. Notwithstanding this conclusion, the applicant computed the frequencies of these waves, if they were to occur, to be between 0.29 Hz and 0.56 Hz. This range is well below the 1.94-Hz downcomer natural frequency in case of a LOCA when the water column inside a downcomer would be displaced by steam. The staff agrees with the applicant that, on the basis of this analysis, resonance will not occur. Therefore, the staff concludes that the applicant has adequately considered all secondary loads. Furthermore, it is noted that in its downcomer design analysis for chugging loads, the applicant utilized GE 800-series in lieu of the GE 700-series tests that had been used in earlier analyses. The applicant performed downcomer analyses considering both the GE 801 and GE 804 chugs. For the remaining 800-series chugs, the applicant was able to demonstrate that the previous analyses using the 700-series or the two 800-series cases were bounding. Since the above approach conforms to the staff acceptance criteria, the staff finds the revised design chugging loads acceptable.

Load Combinations

In FSAR Section 6A.2.2.5, "Design Assessment Report for Hydrodynamic Loads," it is indicated that for all mechanical systems, components, and supports, the structural responses to dynamic loads such as LOCA, SRV, and OBE/SSE (operating basis earthquake/safe shutdown earthquake) are combined by the square-root-of-the-sum-of-the-squares (SRSS) method, and then responses to similar dynamic loads for applicable seismic Category I structures are combined by the absolute-sum method. Even though the downcomers are part of the pressure-suppression system, they have been designed as a mechanical piping system. As a result, the staff has accepted the SRSS method for combining the responses of the above-mentioned dynamic loads in the design analysis of the downcomers. The staff position on the combination of dynamic responses by the SRSS method is given in NUREG-0484, Revision 1.

In reviewing the load combination method presented in the applicant's letter dated December 31, 1985, the staff noted that the SRSS method for response combinations for the NMP-2 downcomer was not in conformance with the staff position provided in NUREG-0484, Revision 1. In a letter dated January 8, 1986, the

applicant was requested to assess its load combination method in accordance with the staff position. In response to the staff's concern, the applicant has revised its methodology for load combinations in accordance with the methodology described in NUREG-0484, Revision 1. This resolved the staff's concern on the load combinations.

Functional Capability

In response to an earlier staff concern on the functional capability of essential piping systems for NMP-2, the applicant made a commitment in its FSAR, as amended, that all essential ASME Code Class 1, 2, and 3 piping system would be designed to meet the functional capability criteria provided in the topical report NEDO-21985 submitted to the staff by GE. On the basis of this commitment, the staff stated in SER Section 3.9.3.1 that "for those piping systems identified as essential that are subjected to loads in excess of Service Level B limits, their functional capability has been evaluated in accordance with the criteria provided in the GE Topical Report NEDO-21985, 'Functional Capability Criteria for Essential Mark II Piping,' dated September 1978, which the staff has previously reviewed and approved."

In the detailed design report (December 31, 1985, letter from C. V. Mangan to E. Adensam) for the NMP-2 downcomers previously submitted, the applicant indicated that the design of the NMP-2 downcomers failed to meet the functional capability criteria presented in GE's report NEDO-21985. The applicant then elected to perform a detailed dynamic stability analysis, which is an option provided in the staff evaluation of the topical report dated February 27, 1981. On the basis of the review of the analysis provided in the December 31 letter, the staff concluded that the applicant did not adequately demonstrate the functional capability of the downcomers, and conveyed its specific concerns to the applicant in its letter of January 8, 1986.

In response to the staff concern, the applicant reevaluated the functional capability of the NMP-2 downcomers (letters from C. V. Mangan to E. Adensam, January 23 and 24, 1985). In this reevaluation, the applicant performed a finite element elasto-plastic shell analysis using the revised limiting loads for the faulted condition. The results were compared to criteria contained in NUREG-0261 on deflection, in GE's report NEDO-21985 on functional capability, and in NUREG-1061, Volume 2, on strain. Note that the strain criteria proposed in NUREG-1061, Volume 2, have not been accepted as a staff position. Furthermore, NUREG-1061, Volume 2, recommended that a factor of safety of 1.5 to 2.0 be applied for the design.

On the basis of the review of the information provided in the applicant's letters of January 23 and 24, 1985, the staff concludes that the applicant has not adequately demonstrated the design adequacy for the faulted conditions; i.e., the downcomer may lose geometrical stability before reaching the calculated stress levels for the faulted condition. The bases for this conclusion are as follows:

NUREG-0261 is based on a small displacement analysis that can not predict buckling. Accordingly, the comparison to the NUREG-0261 results is not meaningful. NEDO-21985 was developed for piping systems. The NMP-2 unbraced downcomers are different from typical piping systems because of the following:

(1) Piping systems have two or more anchors; hence, a single plastic hinge will not lead to gross plastic displacements of the piping system.

(2) Piping systems usually have internal pressure. The stress criteria presented in the NEDO-21985 report includes a pressure term of PD/4t. For piping with a large D/t, the pressure effect may be significant even for a relatively small internal pressure. It is noted that the applicant has not considered the effects from internal pressure and dead weight of downcomers in making comparison to the NEDO-21985 stress criteria. If these two effects were included, the result of the comparison to NEDO-21985 criteria would have changed from being acceptable by a factor of 1.03 to being not acceptable.

Figure 2 in the applicant's letter of January 24, 1986, presents a comparison of the maximum calculated strain of 0.0059 at the limiting moment for NMP-2 downcomers to the strain criterion of $\varepsilon = 0.2\ (t/r)$, where t is the thickness and r is the nominal radius of a downcomer pipe, as suggested in NUREG-1061 (i.e., $\varepsilon = 0.00625$ at $D/t = 64$) as well as the test data from Reddy's paper (1979). The validity of this comparison depends largely on the results presented there. However, in reviewing Reddy's paper, the staff notes that several key parameters relevant to material properties of the test specimens have not been clearly specified; e.g, actual wall thickness, out-of-roundness, type of material. The staff believes that there are considerable uncertainties associated with these parameters that could invalidate their direct applicability to the NMP-2 downcomer design.

Fatigue Evaluation

In the December 31, 1985, letter, the applicant provided its fatigue evaluation of the NMP-2 downcomers. The staff's review of that material raised the concern that because the downcomers as designed have a fundamental mode natural frequency between 1 and 2 Hz, the most significant fatigue damage may incur from the low-cycle/high-stress oscillations. The applicant was requested to clarify its analysis to demonstrate the adequacy of the fatigue design of the NMP-2 downcomers.

In response to the staff concern, the applicant provided a revised fatigue evaluation for the NMP-2 downcomers in its letter of January 23, 1986. The applicant stated that a rigorous ASME Code Class 1 fatigue reanalysis has been performed and the result satisfies the ASME Code Class 1 requirement. The applicant also stated that this revised fatigue analysis is performed for the critical location of the downcomers; i.e., at the junction between the downcomers and the drywell floor, and all postulated loading events that can occur on Mark II plant and can affect the downcomers are considered.

In reviewing the calculations provided in the applicant's January 23, 1986, letter, the staff noted the applicant's analysis method is not a straightforward application of ASME Code rules and, in some areas of calculations, the results were nonconservative as compared with the Code. However, in view of the substantial margin of the calculated cumulative usage factor (CUF) to the Code requirement; i.e., CUF = 0.182 which is significantly less than 1.0, the staff believes that the results provide a sufficient margin to ensure the adequacy of the fatigue design of the NMP-2 downcomers.

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The above conclusions along with staff recommendations were furnished to the applicant by letter dated January 31, 1986 (R. Bernero to B. G. Hooten).

Evaluation of Request for Schedular Exemption

As a result of staff's January 31, 1986, letter (C. V. Mangan to R. Bernero), the applicant submitted a letter dated February 18, 1986, to request a schedular exemption pursuant to the Commission's regulations under 10 CFR 50.12(a) to allow completion of the analysis and any resulting requirement for modification of the installed downcomers in an orderly manner. Specifically, the requested exemption is to permit operation during the time that confirmatory analyses of design margins for the NMP-2 downcomers are being performed. Furthermore, it is requested that the Commission permit any hardware changes to the facility required as a result of this confirmatory evaluation to be completed before startup following the first refueling outage. On the basis of the results of the analysis provided in its exemption request of February 18, 1986, the applicant has concluded that the granting of the schedular exemption would be in accordance with the requirements of 10 CFR 50.12(a). The following material details the applicant's request for a schedular exemption as described in the exemption request and provides the staff's evaluation and conclusion.

Under 10 CFR 50.12(a), the Commission may grant specific exemptions from the requirements of the regulations if (1) the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security and (2) special circumstances are present.

The specific design requirement from which the applicant requested exemption is General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR 50.

GDC 2 requires that the structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions. Furthermore, GDC 2 specifically states that the design bases for these structures, systems, and components shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. It is this particular load combination for which the applicant has requested a schedular exemption to allow additional time to perform further analyses. In the interim, the applicant has requested that the LOCA and safe shutdown earthquake (SSE) loads not be combined because of the low probability of simultaneous occurrence of both events.

In the February 18, 1986, letter, the applicant presented the following technical arguments to support its request for schedular exemption at NMP-2. The applicant contended that (1) the analyses performed support the adequacy of the design; (2) there are margins to failure beyond the ASME Code limits; and (3) there are further unquantified margins based on conservatisms in load combinations and definitions. More specifically, the applicant contended:

(1) The probability of simultaneous occurrence of an SSE and a LOCA is small; therefore, if this load combination were neglected, sufficient additional margins would exist to preclude questions regarding the design adequacy,

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(2) The probability of a large LOCA is now considered to be significantly lower than previously believed.

As discussed in the above evaluation, on the basis of the information presented to the staff to date, the staff does not agree with the applicant's contention that the current analyses demonstrate that sufficient margins are available in the downcomer design to accommodate uncertainties for the LOCA and SSE load combination. The staff in this evaluation addresses the separation of the LOCA and SSE loads for one cycle of operation on the basis of event probabilities and on the materials of construction. The simultaneous occurrence of a LOCA and SSE has two possible scenarios: (1) a double-ended-guillotine break (DEGB) of the recirculation piping (LOCA) simultaneously with an earthquake when both events are unrelated and (2) the seismically induced failure of the recirculation piping during an earthquake. The applicant has presented results from a recent Lawrence Livermore National Laboratory study (the study) on pipe rupture in BWR plants. For the first scenario, the study concluded that the likelihood of simultaneous occurrence of two independent and random events is negligibly low. The staff has estimated that, in the relatively seismically stable region east of the Rocky Mountains, the probability of exceeding the SSE peak acceleration (0.1 g to 0.25 g, depending on the location) is on the order of 10^{-3} or 10^{-4} per year (Reiter, 1983). The probability of a large-break LOCA for boiling-water reactor (BWR) piping is about 10^{-4} to 10^{-8} per year for a large size pipe (> 6.0 inches), independent of seismic event (NUREG-75/014 (formerly WASH-1400); NUREG/CR-3085, -3028, -3600). For BWR piping free of intergranular stress corrosion cracking (IGSCC), the probability of a DEGB (or its equivalence in a longitudinal split) tends to be closer to the lower value. Although the staff recognizes the uncertainties associated with these probabilities and believes a quantitative combination of the two event probabilities may not be meaningful, the staff agrees with the applicant that the probability of simultaneous occurrence of two independent and random events is of extremely low probability.

For the second scenario, the study calculates the probability of LOCA in terms of direct and indirect DEGB. A direct DEGB is pipe failure due to the crack growth at welded joints by either exceeding net section stress for austenitic stainless steels or the tearing modulus for carbon steels. An indirect DEGB is the pipe rupture caused by the seismically induced support failure. That is, an earthquake could cause the failure of component supports or other heavy equipment whose failure in turn would lead to recirculation pipe breaks. The study showed that earthquakes were not a significant contributor to the failure mode of a direct DEGB, especially if the piping was fabricated with an IGSCC-resistant material. The recirculation piping at NMP-2 is made of type 316NG stainless steel that is more IGSCC resistant. The applicant has committed to take additional steps to avoid stress corrosion cracking (see FSAR Section 5.2.3.4.1). With minor exceptions, all other wrought austenitic stainless steels in the reactor coolant pressure boundary are IGSCC-resistant, low-carbon type 304L or 316L. The study showed that the failure probability of a direct DEGB is from 1.5×10^{-9} to 2.5×10^{-10} events per plant year at the 90% confidence limit. The probability of an indirect DEGB induced by an earthquake is about 5.0×10^{-7} events per plant year at the 90% confidence limit.

The staff agrees with the applicant that the likelihood of a large-break LOCA is even more remote at NMP-2 than at some other BWR plants which do not have piping materials that are resistant to IGSCC. Furthermore, even if the piping were of a conventional type of austenitic stainless steel, any potential

degradation from the operating environment during first fuel cycle would be limited because of the limited exposure time (one fuel cycle) to the BWR coolant environment.

In the applicant's exemption request, three categories of special circumstances were discussed: (1) undue hardship, (2) good-faith effort, and (3) "other" or specifically, future rulemaking.

The staff does not believe that the costs directly associated with design and installation of the downcomer bracings would result in undue hardship or other costs significantly in excess of those incurred by others similarly situated, inasmuch as all other BWRs of this design have installed lateral bracing to support the downcomers.

However, although as stated in the SER, confirmatory items for which the information provided by the applicant does not confirm preliminary conclusions (as in the case of the NMP-2 pool loads) will be treated as open, the staff has recognized that concerns directly relating to the structural adequacy of the downcomers were identified late in the review process. Subsequently, the applicant has made good-faith efforts to verify the adequacy of the downcomer design and thereby meet the requirements of GDC 2. In addition, this exemption would be for temporary relief, not to exceed startup following the first refueling period. Inasmuch as the present design presents no undue risk to the public health and safety for the interim period, requiring the applicant to delay operation of the plant for the period of time that additional analysis and/or required modifications to the plant are being completed would present an undue hardship on the applicant. Accordingly, the staff recommends granting the exemption request for the downcomer design as described above.

The staff knows of no imminent rulemaking that would alter the staff's conclusions on the adequacy of the downcomer design and, therefore, the issue of additional rulemaking is considered not applicable and is not addressed.

Conclusions and Recommendations

On the basis of the review of the information provided by the applicant in letters of January 23 and 24, 1986, the staff concludes that the unbraced downcomer design at NMP-2 satisfies the licensing criteria for upset and emergency conditions but the design is marginal. The applicant has not adequately demonstrated the design adequacy for the faulted condition as discussed above. Specifically, the downcomers may lose geometrical stability before reaching the calculated stress levels for the faulted condition.

The staff has reviewed the applicant's request for exemption under the provisions of 10 CFR 50.12. Under these provisions, a finding must be made in accordance with (1) 10 CFR 50.12(a)(1) that the proposed action is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security, and (2) 10 CFR 50.12(a)(2) that the proposed exemption involves special circumstances as defined in 10 CFR 50.12(a)(2)(i) through (vi).

On the basis of the estimates of the probability of the seismically induced pipe rupture at NMP-2 and the short exposure time of the piping to the operating environment during one fuel cycle, the staff concludes that the likelihood of a

LOCA and SSE occurring simultaneously is small during the period for which the exemption was requested, i.e., the first fuel cycle. With decoupling of these loads for the first fuel cycle, the staff finds that sufficient margin exists in the design of the downcomers. For these reasons, the staff finds that the proposed exemption is authorized by law and will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Furthermore, as addressed above and in accordance with 10 CFR 50.12, the staff finds that special circumstances are present. Therefore, the staff recommends that a schedular exemption be granted for NMP-2 downcomers until the end of the first refueling outage. Before startup following the first refueling outage, the applicant should demonstrate the design adequacy of the downcomers with respect to the faulted condition and complete any required modifications to the downcomers.

6.2.3 Secondary Containment

In the SER (NUREG-1047), the staff reported on the applicant's drawdown analysis which functions to bring the secondary containment to a pressure of negative 0.25 inch water gauge. Amendment 23, issued in December 1985, revised the drawdown time from 90 seconds, as previously reported, to 129 seconds. In addition, the capacity of a standby gas treatment system train has been reduced to 3500 cfm from 3600 cfm. To verify the drawdown time, the applicant has committed to perform drawdown tests on the secondary containment every 18 months. The drawdown time acceptance criteria will be reduced below 129 seconds to account for the fact that emergency heat loads are not present in the periodic test, but were included in the FSAR analysis.

The staff has reviewed the changes made by the applicant, discussed above, as well as the proposed inservice testing and finds them acceptable with respect to containment concerns. The effect of the revised drawdown time on the radiological consequences of a LOCA will be discussed in Section 15 in a future supplement to the SER.

6.2.3.1 Bypass Leak Paths

In SSER 2, the staff provided Table 6.1, "Potential Bypass Leakage Paths," which was developed from FSAR information. Amendment 23 makes one change to that table. The drywell floor vent line which terminates in the radwaste tunnel contains a 3-inch valve with Technical Specification leakage of 0.9375 standard cubic feet per hour (scfh) rather than a 6-inch valve as had previously been reported. The staff finds this revision acceptable. See Table 6.3 (revised from SSER 2, Table 6.1).

6.2.4 Containment Isolation System

The containment isolation system includes the containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of LOCA. Staff review of this system included the determination of the number of isolation valves, valve location, the valve actuation signals and valve control features, the valve position under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified in the design of components.

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving

the integrity of the containment boundary to prevent or limit the escape of fission products from a postulated LOCA. The applicant specified design bases and design criteria as well as the isolation valve arrangements to be used for isolating primary containment penetrations.

The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of two isolation valves in series or a closed system and an isolation valve, are provided to ensure that no single active failure will result in the loss of containment integrity. The containment isolation system components, including valves, controls, piping, and penetrations, are protected from internally or externally generated missiles, water jets, and pipe whip.

The basis for staff acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the General Design Criteria (GDC) of Appendix A to 10 CFR 50, and to applicable regulatory guides, staff technical positions, the Standard Review Plan (SRP), and industry codes and standards.

The containment isolation systems are designed to the American Society of Mechanical Engineers Code, Section III, Class 1 or 2, and are classified as seismic Category I design systems.

The containment isolation provisions for the lines penetrating containment conform to the requirements of GDC 55, 56, or 57, except as noted below. As provided by GDC 55 and 56, there are containment penetrations whose isolation provisions do not have to satisfy the explicit requirements of the GDC but can be acceptable on some other defined basis.

Most of those penetrations not satisfying the explicit requirements of the GDC were found acceptable based on their meeting alternative criteria as specified in SRP Section 6.2.4, item II. These alternative acceptance criteria are summarized below:

(1) Lines that must remain in service following an accident and lines that should remain in service during normal operation for safety reasons are provided with at least one isolation valve. A second isolation boundary is formed by a closed system outside the containment. The following penetrations rely on a single isolation valve and a closed system outside containment.

Penetration No.	Description
Z-5A, B, and C	RHR pump suction from suppression pool
Z-6A and B	RHR test return line to suppression pool
Z-7A and B	RHR containment spray to suppression pool
Z-12	HPCS pump suction from suppression pool
Z-13	HPCS test return and minimum flow bypass to suppression pool

Penetration No.	Description
Z-15	LPCS pump suction from suppression pool
Z-17	RCIC suction from suppression pool
Z-18	RCIC minimum flow to suppression pool
Z-19	RCIC turbine exhaust
Z-73	RHR relief valve discharge to suppression pool
Z-88A and B	RHR safety valve discharge to suppression pool
Z-98A and B	RHR relief valve discharge to suppression pool

System piping and valves outside the containment, which are a part of the closed system boundary, are of seismic Category I, Safety Class 2, design; are protected from missiles; and have design temperature and pressure ratings at least equal to those for the containment. Branch lines from the closed system are valved closed and procedurally controlled. Leakage testing of the closed engineered safety feature systems outside containment will be performed in accordance with Section XI of the ASME Code. Relief valve isolation valves listed above seat on accident pressure and contain setpoints greater than 1.5 times the containment design pressure.

(2) On some engineered safety features or a related system, remote manual valves are used in lieu of automatic valves, since these lines must remain in service following an accident. Periodic inspection, testing, and maintenance procedures under normal operating conditions serve to minimize the potential for leakage. For fluid system lines equipped with remote manual isolation valves, the operator in the main control room is provided with information necessary to determine the existence and magnitude of a potential leak. Parameters used to detect leakage are high radiation, high area temperature, high sump level, and reactor vessel and system pressure. By using these parameters, the operator will be able to detect degraded system performance attributable to system leakage and take appropriate action to isolate systems that are potential leak paths.

(3) On some penetrations, the containment isolation provisions consist of two valves in series, both of which are outside the containment. The location of a valve inside containment would subject it to more severe environmental conditions (including suppression pool dynamic loads), and it would not be easily accessible for inspection. An example of this is the purge lines in the drywell and suppression chamber.

(4) Instrument lines that penetrate the primary containment and connect to the reactor coolant pressure boundary (RCPB) are equipped with a restricting orifice located outside and as close as practical to the primary containment, in accordance with Regulatory Guide (RG) 1.11. Those instrument lines that do not connect to the RCPB are equipped with automatic isolation valves whose status is indicated in the control room.

(5) Test connections located before the containment isolation valves in systems containing closed loop boundaries will have two valves in each test, drain, or vent line to ensure that double barrier protection exists in maintaining containment integrity.

Lines penetrating the containment described below do not meet either the explicit requirements of the General Design Criteria or the alternative Standard Review Plan acceptance bases, but either meet acceptable isolation criteria on other defined acceptance bases or require an exemption from the General Design Criteria.

Feedwater Lines

(1) The feedwater line (penetration Z-4) penetrates the drywell to connect with the reactor pressure vessel (RPV). It has three isolation valves. The isolation valve inside the drywell is a check valve. Outside the primary containment is another check valve. Farther away from the primary containment is a motor-operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer prompt primary containment isolation. During the postulated loss-of-coolant accident it is desirable to maintain reactor coolant makeup water from all sources of supply. For this reason, the outermost valve does not automatically isolate upon a signal from the protection system. The motor-operated gate valve meets the same environmental and seismic qualifications as the outboard check valve. The valve can be remotely closed from the control room to provide long-term leakage protection once the operator determines that feedwater makeup is unavailable or unnecessary.

(2) Similar to the feedwater lines is the RCIC/RHR head spray line, penetration Z-22. The head spray line penetrates the drywell and discharges directly into the RPV. It contains testable check valves inside and outside containment.

Upstream of the check valves are a remote manual gate valve (2ICS*MOV126) on the RCIC line and an automatic isolation globe valve (2RHS*MOVI04) on the RHR supply line. The check valves provide a measure of containment integrity in the short term; the gate/globe valves provide long-term leak integrity. All four valves are listed in FSAR Table 6.2-56, "Containment Isolation Provisions for Fluid Line," as being isolation valves. GDC 55, 56, and 57 require that containment isolation valves be located as close as practical to the containment boundary. The RHR reactor head spray line isolation valve is located a piping run of 29 feet 5 inches from the containment and the RCIC isolation valve pipe run is 4 feet 3 inches. The staff believes that these distances are acceptable because by locating the valves there, the applicant is able to reduce the number of penetrations since the RHR head spray and the RCIC line are combined downstream of these valves to form one penetration.

(3) Each of the four main steam line penetrations, Z-1A, B, C, and D, is equipped with a 3/4-inch drain line located before the outermost isolation valve and outside primary containment. These lines each contain a remote manually operated solenoid valve which is normally closed. Downstream of these valves the four 3/4-inch lines join together to form a 2-inch-diameter line which contains the outboard automatic motor-operated, containment isolation valve. This valve, 2MSS*MOV208, is considered to be

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the outboard isolation valve for the four drain lines. Because of this arrangement, the applicant has committed to lock closed the 3/4-inch solenoid valves during normal operation to provide an additional margin of safety since the outboard containment isolation valve is a 36-foot pipe run from the containment boundary.

(4) The standby liquid control system penetration Z-29 contains a simple check valve inside containment and stop check valves on each of two branch lines that feed into the RPV. The stop check valves have a motor operator which acts to keep them closed during normal operation. The system also contains an explosive shear valve that acts as a blind flange during normal plant operation by making a leak-tight seal. Operation of the system requires firing the explosive shear valve to break the seal. The containment isolation provisions are acceptable with a check valve outside containment because the penetration does not communicate with the secondary containment unless the shear valve is fired.

Each of the systems mentioned above meet the General Design Criteria requirements because they satisfy "other defined bases" established by the staff as meeting the GDC requirements but not specifically listed in the SRP. In addition to these systems, the applicant has requested an exemption from GDC 55 for penetrations Z-38A and B, the control rod drive (CRD) hydraulic lines to the reactor recirculation seal purge equipment. GDC 55 does not allow a simple check valve to be used as the automatic isolation valve outside containment. The applicant has proposed to use two simple check valves (spring closing) outside containment in this 3/4-inch line. Furthermore, all three isolation valves (one inboard, two outboard) will be subject to type C leak testing.

The control rod hydraulic system supplies water to the recirculation system for purging of the pump seals. This water cleans and cools the seal area to ensure proper operation during normal plant conditions. Continued recirculation pump seal purge is needed whenever reactor coolant temperature is above 200°F and the pump is not isolated. This prevents premature aging and possible damage to the pump seals from high temperature. The check valves provide containment isolation while permitting seal purge, if available. The check valves are designed so that they are held shut by a spring under no-flow conditions. This isolation valve arrangement for the seal purge line is similar to the arrangements at other BWR-5 plants.

The system leakage boundary leak path does not directly communicate with the environment following a loss-of-coolant accident (LOCA). The system leakage boundary piping components are designed in accordance with Quality Group B standards as defined by RG 1.26, are designed to meet seismic Category I design requirements, and are designed to protect against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features. The system leakage boundary is continually pressurized to reactor pressure and, therefore, system integrity is continually demonstrated during normal plant operations.

In addition, TMI Action Plan Item II.K.3.25, "RCS Pump Seal Design," addresses the importance of providing a source of coolant to the seal coolers by indicating that a loss of seal coolers with resultant seal failure may be the cause for a small LOCA inside containment. For these reasons, the staff believes that automatic isolation valves are not necessary for this system. The benefits gained by providing check valves outweigh the disadvantages, since the

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check valves provide a more reliable flow of coolant to the seals in a plant condition that calls for containment isolation. If automatic isolation valves were used, an isolation signal would isolate the seal purge line. In FSAR Amendment 24, Table II.E.4.2-1 of Section 1.10 was revised to indicate the pump seal purge line is required for seal operation and is considered an "essential" part of the reactor coolant recirculation system. Consequently, the staff concludes an exemption to GDC 55 is justified in this case and the staff recommends granting this request.

In accordance with 10 CFR 50.12(a)(2), special circumstances exist which would warrant issuance of the requested exemption. As discussed above, availability of the reactor recirculation pump seal purge water is necessary to protect the reactor recirculation pump seals. The check valves provide containment isolation while permitting seal purge, if available. Also, as discussed above, the benefits gained by providing check valves outweigh the disadvantages, since the check valves provide a more reliable flow of coolant to the seals in a plant condition which would call for containment isolation. If automatic isolation valves were used, an isolation signal would isolate the seal purge line, thus making seal water unavailable to the reactor recirculation pump seals. Since availability of the pump seal purge water is necessary to protect the seals, granting an exemption to GDC 55 in this case would provide a benefit to the public health and safety that compensates for any decrease in safety that may result from granting the exemption.

The staff informed the applicant that penetration Z-32 represented an unacceptable isolation arrangement because it did not provide for positive isolation for post-LOCA of a nonessential system as required by TMI Action Plan Item II.E.4.2, "Containment Isolation Dependability," and because it deviated from GDC 56 which does not allow use of a simple check valve outside containment. The staff indicated to the applicant that this penetration, N_2 Purge to TIP Indexing Mechanism, would need to be modified to bring it into conformance with the GDC and TMI requirements, because the staff did not believe that an adequate basis existed to consider an exemption. In FSAR Amendment 23, the applicant revised the system by replacing the outboard check valve with an automatic solenoid-operated valve. This revised valve arrangement does meet the provisions of GDC 56 and TMI Action Plan Item II.E.4.2, since the nonessential system receives automatic isolation provisions. The staff finds this change acceptable.

6.2.4.1 Containment Isolation Dependability (TMI Action Plan Item II.E.4.2)

<u>Position</u>

(1) The design of the containment isolation system complies with the provisions of SRP Section 6.2.4; i.e., in that there is diversity in the parameters sensed for the initiation of containment isolation.

(2) Essential and nonessential systems for the purpose of isolation are properly identified.

(3) All nonessential systems are automatically isolated by the containment isolation signal.

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(4) Control systems for automatic containment isolation valves are designed so that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening containment isolation valves shall require deliberate operator action.

(5) Purge valves that do not meet the requirements set forth in Branch Technical Position (BTP) CSB 6-4 should have administrative control that governs "sealed closed" valves during Operational Conditions 1, 2, 3, and 4. Furthermore, these valves are to be verified closed at least once every 31 days.

Clarification

(1) The reference to SRP 6.2.4 in position 1 (above) is only to the diversity requirements set forth in that document.

(2) For postaccident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of GDC 54, 55, 56, and 57, as clarified by SRP Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by SPR Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive the diverse isolation signals.

(3) Revision 2 to RG 1.141 will contain guidance on the classification of essential versus nonessential systems. Requirements for operating plants to review their list of essential and nonessential systems, and an appropriate time schedule for completion, will be issued in conjunction with this regulatory guide.

(4) Administrative provision to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting position 4 (above).

(5) Ganged reopening of containment isolation valves is not acceptable. Isolation valves must be reopened on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

(6) The containment pressure history during normal operation should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuation because of the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification. Applicants for operating licenses and licensees of plants that have operated less than 1 year should use pressure history data from similar plants that have operated more than 1 year, if possible, to arrive at a minimum containment setpoint pressure.

(7) Sealed-closed purge isolation valves shall be under administrative control to ensure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

Discussion and Conclusions

The following discussion summarizes the applicant's response and the staff's evaluation for each item stated above.

(1) Diversity in Parameters. Table 6.4 shows the containment isolation signals and the parameters sensed to initiate each signal. Automatic valves receive two or more of these signals and consequently satisfy the diversity requirement.

(2) Essential and Nonessential Systems. The applicant has evaluated essential and nonessential systems. Table 6.5 lists the essential and nonessential systems as provided by the applicant, along with the basis used for making that determination. The staff finds this list acceptable.

(3) Isolation of Nonessential Systems. All nonessential system lines are automatically isolated by (diverse) containment isolation signals:

Reactor recirculation pump seal purge (Z-38A, B). As discussed in the request for exemption (Section 6.2.4 above) for this system, isolation of these lines is provided by simple check valves. Operating the recirculation pump seal purge line is desirable during pump operation, and whenever the reactor coolant temperature is greater than 200°F, regardless of whether or not the pump is running. Automatic isolation valves are, therefore, undesirable, whereas check valves enhance the operational reliability of the seal purge system. Furthermore, in Amendment 24 to the FSAR the applicant has indicated that the pump seal purge line is an essential part of the reactor recirculation system. Consequently, the staff concludes that the isolation provisions for these penetrations conform to the requirements of Item II.E.4.2.(3). The staff finds this acceptable.

(4) The applicant has indicated that all necessary modifications have been completed so that resetting the containment isolation signal will not result in the automatic reopening of containment isolation valves, i.e., reopening isolation valves requires deliberate operator action.

(5) The applicant has verified that the containment setpoint pressure is the minimum that is compatible with normal operating conditions. Also, ganged reopening of containment isolation valves will not occur.

(6) Containment purge valve operability, including the ability of these valves to close against a LOCA, was addressed in Appendix J to SSER 2, November 1985, and in this SER supplement. The functions to be performed by the purge system are: inerting, deinerting, and pressure control. The 12-inch and 14-inch purge valves will be in use during the operations of inerting and purging. For these functions, there is a limit of 90 hours' use every year described in SRP Section 6.2.4.II.6.n. The pressure control function is accomplished by operation of a 2-inch bypass line which is open to the

standby gas treatment system (SGTS). The 2-inch bypass line taps off the larger purge valve line downstream of the outboard containment isolation purge valve, thus requiring both inboard and outboard valves to be open. The applicant has shown that the SGTS will survive the pressure pulse resulting from a postulated LOCA concurrent with the bypass line open. While the pressure control function takes place, the containment purge valves are partially open; however, flow is eliminated through all but the 2-inch line because of the presence of a closed (fail-closed) 20-inch safety-related valve, 2GTS*AOV101, in the flowpath to the SGTS. Containment isolation is achieved when needed by closing the 12- and 14-inch containment purge valves. To summarize the restriction of 90 hours of operation for the 12- and 14-inch purge valves applies to the functions of inerting and deinerting which take place when 20-inch valve 2GTS*AOV101 is open. The function of pressure control through the 2-inch bypass line, through partially open purge valves, does not have a time limit but is understood by the applicant to be no more than necessary to maintain the containment pressure between the Technical Specification limits. This is acceptable to the staff because the SGTS has been predicted to survive a pressure pulse through the 2-inch line, and the 20-inch safety-related valve discussed above will serve to limit flow through the purge penetration to only the amount going through the bypass line. Finally, any leakage through the closed 20-inch valve would also leak into the SGTS and would be processed by it.

(7) The applicant has indicated that the control logic for the containment purge supply and exhaust lines has been revised to incorporate automatic isolation on high radiation.

Conclusion

On the basis of its review, the staff concludes that the applicant is in compliance with the requirements for containment isolation dependability given in Item II.E.4-2 of the TMI Action Plan.

6.2.6 Containment Leakage Testing Program

Item 1: CRD System Type A Test

In the SER (NUREG-1047), the staff indicated that the control rod drive (CRD) system insert and withdraw line isolation valves need not be type C tested. However, the staff also stated that the CRD system should be vented for the type A test in order to expose the system to containment accident pressure, P_a. In order to meet this requirement, the applicant has proposed to open (vent) the scram discharge volume vent and drain valves during the type A test in lieu of venting the entire CRD system. In addition, 10 CFR 50, Appendix J, type C leak tests will be performed on these valves and the leakage results will be added to the type A test results. The staff finds that this test procedure meets the the Appendix J requirement that all such systems be vented for the type A test and recognizes that the unique aspects of the CRD system preclude conventional venting/draining arrangements. Consequently, the staff finds the proposed test procedure for venting the CRD system during the type A test acceptable. The staff will require that the type C leakage values obtained above be added to the type B and C test allowable leakage of 0.6 L_a

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In a letter dated September 3, 1985, the applicant requested an exemption from the test requirements of 10 CFR 50, Appendix J. Specifically, the applicant requested an exemption from the provisions requiring venting and draining of the CRD hydraulic lines to the scram discharge volume during the type A containment integrated leak test. The staff recognizes that the CRD is a unique system in that it is needed to function in the postaccident condition via operation of the scram system. Appendix J provides relief from the venting requirement for systems such as the CRD system which "are normally filled with water and operating under postaccident conditions." These systems, according to Section III.A.1.d of Appendix J need not to be vented provided the isolation valves are type C tested and the leakage measured is added to the type A test results. The applicant has committed to do so and consequently no exemption need be granted in this circumstance.

Item II: Reverse Direction Type C Testing

Appendix J (10 CFR 50), Section III.C.1, prescribes methods for conducting the containment isolation valve leak rate tests. These requirements state that isolation valves should be leak tested with the test pressure applied in the same direction the valve must function to preclude leakage in the accident condition. Reverse direction testing is permitted if it can be demonstrated that such testing yields results that are equivalent or more conservative than results obtained using same direction as postaccident flow testing. In letter NMP2L-0282 (from C. V. Mangan, NMPC, to A. Schwencer, NRC, December 7, 1984), the applicant provided Table 6.6 which lists the containment isolation valves the applicant proposes to reverse direction test, the valve type, and the justification. The staff has reviewed the bases used as justification for reverse direction testing of these valves and concludes that they are acceptable. Consequently, the staff approves the reverse direction testing of the containment isolation valves listed in Table 6.6.

Item III: Hydraulic Control System for Recirculation Flow Control Valves

By letters dated April 26, 1985, and September 3, 1985, the applicant requested exemption from certain requirements of 10 CFR 50, Appendix J. Specifically, exemptions were requested from both type A and type C leak testing for the hydraulic control system for the reactor recirculation flow control valves because testing these lines would require the system to be disabled and drained of hydraulic fluid.

System Description

The hydraulic control system for the reactor recirculation system flow control valves operates to control the reactor recirculation flow during normal operation and is automatically isolated following a postulated accident. The system provides hydraulic fluid through eight containment penetrations (Z-99A, Z-99B, Z-99C, Z-99D, Z-100A, Z-100B, Z-100C, Z-100D) to the hydraulic operators on the two recirculation flow control valves. The hydraulic lines terminate in the reactor building; therefore, the system does not constitute a potential bypass leak path. The system leakage boundary piping components are designed as Quality Group B between the isolation valves and Quality Group D outside the isolation valves. Although the recirculation flow control valve actuator, which is part of the high-pressure hydraulic system, is not environmentally qualified for operation following the post-LOCA containment temperatures and pressures expected

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in the drywell, the system is designed to withstand a safe shutdown earthquake and is protected against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features. For this system, the applicant has requested exemptions from both type A and type C leak testing because testing these lines would require the system to be disabled and drained of hydraulic fluid. The applicant has stated that testing could be especially detrimental to the proper operation of the system, because possible damage could occur to the system not normally exposed to air in establishing the test condition or restoring it to normal. The staff has evaluated this request and concludes that a basis exists for granting an exemption for this system from both the type A and the type C tests of 10 CFR 50, Appendix J. The staff believes that although the system is not operationally qualified to the post-LOCA containment environment, because it is protected against pipe whip, missiles, and jet forces, there is a reasonable basis for concluding that the system boundary will maintain its integrity and, therefore, will not become a containment atmosphere leak path. In addition, the staff agrees it is not advisable to drain this type of hydraulic line because of possible damage that may result from either establishing the test or restoring the system to proper operation.

Special Circumstances

In accordance with 10 CFR 50.12(a)(2), special circumstances exist which would warrant issuance of the requested exemption. Application of the requirements in this particular circumstance would not be necessary to achieve the underlying purpose of the requirement and the exemption would result in an overall benefit to the public health and safety that would compensate for any decrease in safety that might result in granting of the exemption.

The hydraulic control system lines terminate in the reactor building; therefore the system does not constitute a potential bypass leak path. The system leakage boundary piping components are designed as Quality Group B between the isolation valves and Quality Group D outside the isolation valves. Although the recirculation flow control valve actuator, which is part of the high-pressure hydraulic system, is not environmentally qualified for operation following the post-LOCA containment temperatures and pressures expected in the drywell, the system is designed to withstand a safe shutdown earthquake and is protected against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features. Therefore, although the system is not operationally qualified to the post-LOCA containment environment, because it is protected against pipe whip, missiles, and jet forces, there is a reasonable basis for concluding that the system boundary would maintain its integrity and, therefore, will not become a containment atmosphere leak path. Therefore, the underlying purpose of the leak testing (assuring that the containment leakage is minimized) is sufficiently achieved by the design of the system, thereby meeting the requirements of 10 CFR 50.12(a)(ii).

Type A and C testing of this system would require the system to be disabled and drained of hydraulic fluid. Possible damage could occur to the system not normally exposed to air in establishing the test condition or restoring it to normal conditions. Therefore, not subjecting this system to the increased probability of damage would benefit the public sufficiently to compensate for any decrease in safety that might result in granting of the exemption following the considerations discussed above. Therefore, special circumstance as discussed in 10 CFR 50.12(a)(iv) is met.

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Item IV: Traversing Incore Probe (TIP)

After the SER was printed, the applicant requested an exemption from the type C Appendix J test on the TIP ball valve on the grounds that the system is in operation approximately 4 hours a month, the leakage potential is small, and the dosages incident to the test program itself were high relative to the benefit gained from the test. The staff has evaluated this request and has concluded that an adequate basis does not exist to grant an exemption for the TIP system from type C testing. The potential leakage from the TIP system is not inconsequential and may impact the successful completion of the type A and/or C tests.

In addition, the staff does not believe, on the basis of information provided to date, that the exposure rates of the plant personnel performing the tests are excessively high or significantly higher than normal rates expected to be encountered in the drywell during other routine maintenance operations conducted during the refueling outage. For these reasons, the staff believes that the TIP system must be type C tested in accordance with 10 CFR 50, Appendix J. The Technical Specifications for NMP-2 will require these valves to be type C tested.

The staff has completed its review of the applicant's proposed containment leak test program. The staff finds the test program, as described in the SER and its supplements, to be acceptable. With the exception of the recirculation flow control system, for which an exemption was requested and for which adequate basis exists, the test program reviewed by the staff conforms to 10 CFR 50, Appendix J.

In letters dated March 3, and 5, 1986, the applicant requested additional exemptions from the requirements of 10 CFR 50, Appendix J. Those exemption requests concern the exclusion of leakage of the main steam isolation valves from the acceptance criteria contained in Section III.C.3 of Appendix J, the relaxation of testing requirements for airlock doors, and the exclusion of certain relief valves from type C testing. These exemption requests are under staff review. The staff will discuss the findings of the review of these requests in a future supplement to the SER.

Table 6.1 Comparison of short-term peak pressures

Plant	Drywell (psig)	Suppression chamber (psig)
Nine Mile Point 2	39.9	34
Susquehanna 1 & 2	43.8	28.9
Shoreham 1	41.9	30
WPPSS 2	34.7	27.6
LaSalle 1 & 2	32.4	24.8

Table 6.2 Comparison of selected containment characteristics

Containment characteristics	Shoreham	NMP-2
Downcomers, no.	88	121
Downcomer ID, in.	23.25	23.25
Design pressure, psig	48	45
Free volume ratio (drywell/wetwell)	1.44	1.51

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Table 6.3 Potential bypass leakage paths (revised from SSER Table 6.1)

Line description	Termination region	Bypass leakage barrier	Leak rate* Tech. Spec. (scfh)**
4 main steamlines	Turbine bldg.	Two 21" valves in each line	6
Main steam drain line (inboard)	Turbine bldg.	One 6" valve	1.875
Main steam drain line (outboard)	Turbine bldg.	One 2" valve	0.625
4 postaccident sampling lines	Radwaste tunnel	One 3/4" valve in each line	0.2344
Drywell equipment drain line	Radwaste tunnel	One 4" valve	1.25
Drywell equipment vent line	Radwaste tunnel	One 2" valve	0.625
Drywell floor drain line	Radwaste tunnel	one 6" valve	1.875
Drywell floor vent line	Radwaste tunnel	One 3" valve	0.9375
RWCU line	Turbine bldg.	One 8" valve	2.5
Feedwater line	Turbine bldg.	Two 24" check valves	12
Containment purge system supply line to drywell	Standby gas treatment area	Two 14" valves Two 2" valves	4.38 0.625
Containment purge system supply line to supply chamber	Standby gas treatment area	Two 12" valves Two 2" valves	3.75 0.625

*Test conditions: Air medium; 40 psig and 80°F; leak rate per valve.
**Standard conditions: 14.7 psia and 68°F.

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Table 6.4 Key to isolation signals

Signal	Parameter sensed
A	Low reactor vessel water, Level 3
B	Low reactor vessel water, Level 2
C	High main steam line radiation
D	High main steam line flow
E	High main steam line tunnel area ambient temperature
F	High drywell pressure
H	Steam supply pressure low
J	High reactor water cleanup system equipment area differential or ambient temperatures, or turbine building high space temperature, or reactor water cleanup high differential flow
K	Reactor core isolation cooling high pipe routing or equipment area ambient or differential temperatures, low steam supply pressure. High steam line differential pressure, high turbine exhaust diaphragm pressure
L	High reactor vessel pressure
M	High residual heat removal system equipment area differential or ambient temperatures
P	Low main steam line turbine inlet pressure
R	Low main condenser vacuum
S	Standby liquid control system actuated
T	High main steam line tunnel differential temperature
W	High reactor water cleanup system nonregenerative heat exchanger outlet temperature
X	Low reactor vessel water, Level 1
Y	Standby gas treatment exhaust radiation high
LC	Locked closed
RM	Remote manual switch from control room

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Table 6.5 Essential and nonessential systems

System	Classification	Basis for classification
1. Main steam	Nonessential	Not required for safe shutdown
2. Feedwater	Nonessential	Not required for safe shutdown. Class 1 portion of feedwater line essential. It is desirable to maintain all sources of cooling supply, if available.
3. Reactor coolant recirculation	Nonessential	Not required for safe shutdown
	Essential	Pump seal purge line is required for seal operation
4. Instrument air	Nonessential	Not required in short term for safe shutdown.
	Essential	Required in long term to support LPCI and LPCS by recharging ADS accumulators from tanks outside containment
5. Service air	Nonessential	Not required for safe shutdown
6. Breathing air	Nonessential	Not required for safe shutdown
7. Standby liquid control	Essential	Should be available as backup to the CRD system
8. RHR		
a. LPCI mode	Essential	Safety function
b. Suppression pool cooling mode	Essential	Required to control suppression pool temperature
c. Containment spray cooling mode	Essential	Required to control drywell/containment pressure
d. Reactor steam condensing mode	Nonessential	Not required for safe shutdown
e. Shutdown cooling mode	Nonessential	Not required for safe shutdown
9. Reactor water cleanup	Nonessential	Not required during or immediately following an accident

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Table 6.5 (Continued)

System	Classification	Basis for classification
10. Reactor core isolation cleanup	Essential	Used as a backup to HPCS when the reactor becomes isolated from main condenser
11. Low-pressure core spray	Essential	Safety system
12. High-pressure core spray	Essential	Safety system
13. Reactor building equipment drains	Nonessential	Not required for safe shutdown
14. Containment leakage monitoring	Nonessential	Not required for safe shutdown
15. Reactor building closed loop cooling water	Nonessential	Not required for safe shutdown
16. Reactor containment inerting and purge	Nonessential	Not required for safe shutdown; however, used if available as backup to Category I DBA hydrogen recombiner
17. Containment atmospheric monitoring	Essential	Required for postaccident monitoring of containment pressure, hydrogen, temperature, and level. Radiation monitors are nonessential because they are not required for safe shutdown
18. DBA hydrogen recombiner	Essential	Required for safe shutdown. Following a LOCA, system is used to remove excess hydrogen that would react with oxygen and lead to high temperature and overpressurization that would result in loss of containment integrity
19. Fire protection water	Nonessential	Not required for safe shutdown

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Table 6.5 (Continued)

System	Classification	Basis for classification
20. Reactor building floor drains	Nonessential	Not required for safe shutdown
21. Control rod	Essential	Required for safe shutdown
22. Traversing incore probe (TIP)	Nonessential	Not required for safe shutdown

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Table 6.6 Reverse tested containment isolation valves

Penetration no.	System	Valve ID	Valve type	Justification*
Z-8A	RHR	MOV25A	Split disc gate	1
Z-8B	RHR	MOV25B	Split disc gate	1
Z-12	CHS	MOV118	Split disc gate	1
Z-18	ICS	MOV143	Globe	2
Z-17	ICS	MOV136	Split disc gate	1
Z-19	ICS	MOV122	Split disc gate	1
Z-21A	ICS	MOV128	Split disc gate	1
Z-48	CPS	AOV108	Butterfly	3
Z-51	CPS	AOV109	Butterfly	3
Z-50	CPS	AOV107	Butterfly	3
Z-49	CPS	AOV106	Butterfly	3
Z-55A	HCS	MOV4A	Globe	2
Z-55B	HCS	MOV4B	Globe	2
Z-56A	HCS	MOV6A	Globe	2
Z-57A	HCS	MOV5A	Globe	2
Z-56B	HCS	MOV6B	Globe	2
Z-57B	HCS	MOV5B	Globe	2
Z-58	CPS	SOV122	Globe	2
Z-59	CPS	SOV121	Globe	2
Z-60A	CMS	SOV61A	Plug	4
Z-60C	CMS	SOV63A	Plug	4
Z-60D	CMS	SOV33A	Plug	4
Z-61C	CMS	SOV34A	Plug	4
Z-60E	CMS	SOV61B	Plug	4
Z-60G	CMS	SOV63B	Plug	4
Z-60H	CMS	SOV33B	Plug	4
Z-61F	CMS	SOV34B	Plug	4

* Justification:

1. Split disc gate valves may be tested using a test connection (TC) between the discs. This is a conservative test since both LOCA and non-LOCA seat leakage is measured.

2. Globe valves are orientated to ensure LLRT test pressure tends to unseat the valve, whereas LOCA pressure will tend to seat the valve. This is conservative for testing.

3. On butterfly valves reverse testing will provide equivalent results since the seating area(s) and test pressure force(s) will be equal in either direction.

4. Plug valves are bi-directional plug-type solenoid valves that are oriented so that LOCA pressure will tend to seat the valve and LLRT pressure will tend to unseat the valve.