Docket No.: 50-271

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Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" and our Safety Evaluation issued by the Commission for the Vermont Yankee Nuclear Power Station.

The Order adds requirements needed to assure the continued safe operation of the Vermont Yankee Nuclear Power Station in the event of the low likelihood, but worst case, loss of coolant accident.

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Order for Modification of License
2. Safety Evaluation

cc: See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

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Enclosed is a signed original of the "Order for Modification of License" and our Safety Evaluation issued by the Commission for the Vermont Yankee Nuclear Power Station.

The Order adds requirements for the operation of the Vermont Yankee Nuclear Power Station to assure the integrity of the torus structure in the event of the low liklihood but worst case loss of coolant accident.

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of VERMONT YANKEE NUCLEAR POWER CORPORATION (Vermont Yankee Nuclear Power Station)	)))))	Docket No. 50-271

ORDER FOR MODIFICATION OF LICENSE

I.

The Vermont Yankee Nuclear Power Corporation (the licensee) is the holder of facility license DPR-28, which authorizes operation of the Vermont Yankee Nuclear Power Station near Vernon, Vermont. This license provides, among other things, that it is subject to all rules, regulations and orders of the Commission now or hereafter in effect.

II.

Results of recent 1/12 scale tests of hydrodynamic characteristics of vapor suppression containment systems of type generally known as Mark I (the so called "light-bulb and torus" shape), have indicated that during the course of a postulated design basis loss of coolant accident, forces exerted on the torus in the upward direction may be sufficient to cause unexpected movement in an upward direction. The Mark I containment design is a feature of some Boiling Water Reactors built by the General Electric Company. For all facilities employing the design except the Vermont Yankee Nuclear Power Station, the upward motion if any, indicated by the data derived from the tests, is within the capability of the torus structures and connected piping. Due to the unique combination of features of the Vermont Yankee design, the estimated upward motion during a design basis loss of coolant accident, for pressure conditions which presently exist in the containment system, would exceed 4 inches. This amount of upward motion is such that under these conditions the integrity of containment and connected piping systems could not be assured. Operation of the facility has been suspended pending assessment of these effects and investigation of techniques to mitigate or eliminate these potentially adverse conditions.

Upon investigation of the test results which led to the indication of upward motion, it appears that by imposing a small differential pressure between the drywell portion of the containment (the "light-bulb" shaped portion) and the wetwell, or torus shaped portion of the system, the upward forces are reduced substantially. Under these conditions upward motion, if any, even in the event of the postulated worst case loss of coolant accident may be reduced to values within the structural capability of the system and its piping. This has been verified by stress calculations performed for connected vital piping, including that of the Emergency Core Cooling System, which indicates that with a pressure differential greater than 1.7 psid piping stresses would not exceed yield stresses.

The licensee has concluded that maintaining a differential pressure between the drywell and the wetwell of 1.7 psid provides adequate assurance that containment integrity will not be jeopardized by upward motions resulting from hydrodynamic forces from the most severe loss of coolant accident. The licensee also proposed subsequent installation of tie down rods to provide mechanical restraint, in lieu of the differential pressure method of providing load reduction, within some 60 days. After discussions with the staff, the licensee established a schedule which would provide for installation within 30 days. (Vermont Yankee submittal dated February 4, 1976.)

The staff has carefully reviewed all the relevant data to ascertain the effect of imposing a differential pressure of 1.7 psid between the drywell and torus. As set forth in the Staff Safety Evaluation (concurrently issued with this Order) the staff has concluded that the data support the conclusion that, with a minimum differential pressure between the drywell and wetwell of 1.7 psid, a best estimate of the upward motion of the torus in the event of the worst postulated loss of coolant accident (the complete double ended offset severance of the 28 inch recirculation line in the limited section between the reactor outlet nozzle and the suction side of the recirculation pump), would be less than one inch. All of the vital piping connected to the torus is capable of withstanding an uplift deflection of at least one inch without overstress. Moreover, the staff has further concluded that the probability of the event required to initiate high containment loadings (that is a complete double ended offset severance of the 28 inch recirculation line) is exceedingly low, particularly in light of the present inservice inspections of the piping welds in this section.

It should be noted that the staff's conclusion with respect to the quantitative effect of drywell pressurization on torus uplift is based on a best estimate, rather than a conservative estimate, in that conservatisms involved in certain assumptions are to some degree offset by uncertainties in other areas. Consequently, the staff concludes that reliance on the pressure differential technique alone should be limited to a short term period of time. Accordingly, the staff concludes that a high priority should be given to installation of the tie down system, with completion by the licensee of its installation within 30 days from the date of this Order. Since maintenance of the pressure differential for the interim period before the tie down installation is completed is of importance to assurance of adequate facility safety, the staff has also concluded that specific limitations on operation in the absence of the minimum differential pressure should be imposed, and that requirements for careful monitoring of differential pressure should be imposed pending further review to assure that the necessary minimum pressure differential is maintained in the containment system.

Based on the foregoing considerations, the staff has concluded that the existing plant licensing requirements and an additional requirement to maintain a differential pressure of 1.7 psid between the drywell and suppression chamber, along with consideration of the low probability of a pipe failure of sufficient size to cause high containment loading collectively provide reasonable assurance that the public health and safety will not be endangered by operation of the Vermont Yankee facility during the short period of time before the required completion of modifications to enhance the structural capability of the containment system. The additional licensing conditions to assure conformance with the pressure differential are set forth as Appendix A to this Order. Further, the staff has concluded that the existing plant licensing requirements and the additional requirement to maintain a differential pressure of 1.7 psid between the drywell and suppression chamber provide assurance that the public health and safety will not be endangered by operation of the Vermont Yankee facility following the completion of the modifications to enhance the structural capability of the containment system.

III.

In view of the foregoing, and in accordance with provisions of the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, the Director of the Office of Nuclear Reactor Regulation has found that additional conditions on facility operation, set forth in Appendix A hereto are required to protect the public health and safety and that the public health, safety, and interest require that the following order be made effective immediately. Pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Part 2 and 50,

IT IS ORDERED THAT:

1. Effective immediately reactor operation shall continue only within the limits of License No. DPR-28 and the Technical Specifications which are a part thereof and the further restrictions set forth in Appendix A, attached hereto.

2. Within thirty days of the date of this Order, the licensee shall complete installation of a system of tie down rods in accordance with the preliminary design contained in Vermont Yankee submittal dated February 4, 1976. The Commission may impose additional requirements including the installation of additional tie down devices if it determines that such additional requirements are necessary to assure adequate design safety margins.

IV.

Within thirty (30) days from the date of publication of this Order in the FEDERAL REGISTER, the licensee may file a request for a hearing with respect to this Order. Within the same thirty (30) day period, any other person whose interest may be affected may file a request for a hearing with respect to this Order in accordance with the provisions of 10 CFR § 2.714 of the Commission's Rules of Practice. If a request for a hearing is filed within the time prescribed herein, the Commission will issue a notice of hearing or an appropriate order.

For further details with respect to this action, see (1) VY application dated February 6, 1976 (non-Proprietary version), (2) Mark I Containment Evaluation, Short-term Program - Final Report, NEDC 20989-1 September, 1975, and (3) Safety Evaluation Report.

These items are available at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

A single copy of item (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 13 day of February, 1976.

APPENDIX A

1. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.7 psid except as specified in A and B below.

   A. This differential shall be established within 24 hours of achieving operating temperature and pressure.

   B. The above differential may be decreased to less than 1.7 psid for a maximum of two hours for the purpose of testing the drywell-suppression chamber vacuum breakers.

2. If the differential pressure of 1.7 psid cannot be maintained, the plant shall be placed in the cold shutdown condition within 24 hours.

3. The pressure difference between the drywell and suppression chamber shall be recorded once per shift.

4. The volume of water in the torus shall be maintained between 68-70,000 Ft^3 and recorded once per shift.

5. The operability of the low differential pressure alarm shall be verified once every seven days.

6. The operability of the alternate circuit of the standby gas treatment system shall be demonstrated once every seven days.

7. From and after the date that one circuit of the standby gas treatment system is made of found to be inoperable for any reason, reactor operation is permissible only during the succeeding three days unless such circuit is sooner made operable.

8. The operability and closure time (equal to or less than 5 seconds) of the three inch torus purge and vent outlet bypass valve shall be verified once every seven days.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING OPERATION WITH THE DIFFERENTIAL PRESSURE CONTROL SYSTEM AND STRUCTURAL MODIFICATIONS TO THE PRESSURE - SUPPRESSION CHAMBER

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Introduction

On January 26, 1976, the Vermont Yankee Nuclear Power Corporation (VYNPC) voluntarily ceased operation of the Vermont Yankee Nuclear Power Station (VYNPS) as a safety precaution. VYNPC took this action to evaluate information and test data received and to conduct appropriate tests of VYNPS's primary containment vessel. The information received by VYNPC from GE included an analysis, based on the test data, which was used to predict the forces that could cause a potential torus uplift during a postulated design basis Loss of Coolant Accident (LOCA). The results of this analysis indicated that the potential uplift during a LOCA could be as much as 4.02 inches (initially determined to be 5.44 inches).* The tests concern the ability of the torus to function as intended during such potential accidents.

The Vermont Yankee Nuclear Power Station is different than facilities of similar design in that: (1) the ratio of break area size to drywell volume is larger, thus permitting faster pressurization of the drywell, (2) the ratio of downcomer vent area to suppression pool area is larger resulting in larger pool swell events, (3) the vent submergence is deeper and (4) facilities similar to Vermont Yankee have anchor bolts restraining the torus whereas Vermont Yankee does not have this feature. The combination of all the above contribute to the magnitude of the potential torus uplift at this facility.

*5.44 inches is the maximum calculated uplift using the original estimated submergence of 4 feet 7 inches; 4.02 inches is the maximum calculated uplift using the actual measured submergence of 4 feet 3.5 inches.

Discussion

During our review of advanced designs of vapor-suppression type containments, (Mark III), containment structural loads were identified that were not considered in the earlier designs of Mark I type containments. The Vermont Yankee containment is of this Mark I type. All utilities with facilities having Mark I containments including the Vermont Yankee Nuclear Power Corporation, were advised by the NRC in April 1975 of this deficiency and were requested to review their plant designs to determine whether this new information would affect the structural adequacy of their containments. Specifically, these containment structural loads were related to the dynamics of the suppression pool responses following a LOCA.

In May 1975, an owners group was formed of all utilities owning plants with Mark I type containments. The purpose of this group was to determine the magnitude and significance of these loads as quickly as possible and identify courses of action needed to resolve any outstanding concerns. The General Electric Company was contracted as the lead technical organization. The group established a short-term and a long-term program. The short-term program is intended to make an assessment of each Mark I plant using current information to determine the significant loads and structural capability of the containments. The long-term program would include large-scale tests and additional refined analyses to evaluate any outstanding concerns which were identified during the short-term program and to confirm the data base for that program.

The short-term program is presently continuing and a report[(3)] was issued in September of 1975 with subsequent amendments to be submitted through March of 1976.

The dynamics of the suppression pool induced loads are described as follows. Following a LOCA in the drywell, the drywell atmosphere will be transferred to the wetwell via interconnecting vents as a result of blowdown mass and energy addition to the drywell volume from the postulated break. Following vent clearing, an air/steam/water mixture will be forced from the drywell through the vent system and injected into the suppression pool approximately four feet below the surface. The steam component of the flow mixture will condense in the pool while the air will be released in the pool as high pressure bubbles. The suppression pool consists of approximately 70,000 ft^3 of water and 110,000 ft^3 of air. The continued addition and expansion of air causes the water pool volume to swell resulting in an acceleration of a ligament of water vertically upward. Initial effects of this water ligament result in impact on torus internals and air compression. Subsequently, due to the effect of buoyancy, air bubbles will rise faster than the pool water mass and will eventually break through the swollen surface and relieve the driving force behind the pool.

For Vermont Yankee, the most significant pool dynamic loads are the upward and downward loads exerted on the torus. The downward load is generated starting immediately after vent clearing and is due to the pressure of the expanding air bubbles within the suppression pool acting on the bottom half of the torus. As discussed above, the expanding bubbles also cause the pool surface to rise, compressing the air at the top of the torus and yielding an upward load. In addition, the impact of the rising pool surface on the vent system ring header can contribute a force component in the upward direction since the header is connected to the bottom of the torus by support struts. The transient loadings due to compression of the air space and vent header reaction must be considered in their proper time phasing to arrive at the total upward load.

During the course of the short-term program, a preliminary assessment of torus upward and downward loads was made on the basis of data from the Bodega Bay tests. These tests, performed in 1962, constitute the original data base for the Mark I containment design. It was determined that the Bodega Bay tests were not fully representative of the Mark I containment with regard to a determination of the effects of the hydrodynamic forces in the torus. However, the data do provide information on pressure suppression system behavior. A series of 1/12 scaled model tests were devised to more accurately define the loading conditions and allow evaluations to be made of torus support capabilities.

The tests were performed by the General Electric Company in December 1975, using 1/12 scale (linear basis) facility representing a segment of a Mark I containment torus. The tests were based upon a drywell pressurization rate that corresponds to the Vermont Yankee design basis loss of coolant accident response; however, the effects of air/steam mixing and steam condensation were not included. The absense of such effects would tend to make the results somewhat conservative.

We and our consultants (BNL) have reviewed the scaling analysis corresponding to the 1/12 scale test rig and believe it to represent a reasonable dimensional and geometric arrangement of the plant configuration. It does, however, only consider two dimensional effects and needs to have the results corrected appropriately by suitable scaling factors. It is in this regard that we are discussing with GE and the Mark I owners the need for additional 1/12 scale tests as well as larger scale tests.

Based on the available test results, upward and downward torus load profiles were derived for each Mark I plant and structural response analyses were conducted. These results indicated that the Vermont Yankee plant had the most severe uplift condition (a calculated torus rise height of 5.44 inches). In calculating the uplift deflection, the stiffness of the connecting piping have been conservatively neglected. Because of such uplift, the licensee could not conclude that there would be no loss of containment function in the event of a design basis LOCA and therefore voluntarily shut down his plant on January 26, 1976.

Following the tests performed in December 1975, and concurrent with the structural evaluations, additional 1/12 scale tests were conducted by G.E. to quantitatively evaluate a potential technique for reduction of torus support loads should it be needed. This method involves maintaining a small overpressure in the drywell of about 1 to 1.5 psi during normal plant operation. Such action would depress the water leg in the downcomers and thereby reduce the drywell pressurization rate following a postulated LOCA. It was believed that this would cause a corresponding attenuation of torus loads.

On the basis of the 1/12 scale pressurized drywell test results the licensee has proposed to operate Vermont Yankee with a drywell to wetwell pressure differential not less than 1.7 psi and a torus water volume between 68,000 ft^3 (current minimum tech. spec) and 70,000 ft^3. The 1.7 psi differential pressure will be maintained by use of components of existing systems. A negative pressure of 0.45 psig will be established and maintained in the wetwell by exhausting through one train of the standby gas treatment system (SGTS). The SGTS is of Class I seismic design and incorporates complete physical and electrical redundancy. An evaluation of the control system for the SGTS components shows that no single failure, except isolation, can prevent the operation of at least one SGTS train. A flow path from the torus to the SGTS is established by opening the 3 inch purge and vent bypass valve and the 12 inch exhaust valve to SGTS. These valves close automatically on isolation signals due to (1) low reactor water level, (2) high drywell pressure, and (3) high reactor building radiation level.

A 1.25 psig pressure will be maintained in the drywell by metering station instrumentation air through the existing containment inerting makeup line and controlling pressure using the existing pressure control valve and its associated pressure transmitter. The air to the torus comes from receivers supplied by one of three available air compressors. Automatic isolation of the air supply is initiated by the same signals as mentioned above. Monitoring of the differential pressure will be accomplished by installing a manometer (normally installed for containment leakage testing) with an accuracy of .01 psi. A differential pressure detector, with read-out and alarm in the control room, and an accuracy of .017 psi will be used. Although the d/P cell is not redundant, backup indication by monitoring drywell pressure and torus pressure is continuously recorded by computer. Backup alarm features are a 1.3 psig low pressure alarm and a 1.5 psig high pressure alarm in the drywell. The peak upward torus loads for these conditions have been determined by the licensee to be 0.62 psig (68,000 ft^3 pool volume) and 0.92 psi (70,000 ft^3 volume). The licensee also provided a curve of torus uplift as a function of the upward pressure which shows that the maximum calculated uplift would not exceed 0.4" for these loads; again, in calculating the uplift deflection, the stiffness of the connecting piping have been conservatively neglected. This estimate did not include the potential reduction in pool swell velocity due to drywell pre-pressurization which would reduce the vent header reaction force contribution to uplift. The licensee has also stated that the containment function would be maintained for upward loads up to 1.5 psi (equivalent to a torus uplift of 1 inch).

The licensee, in parallel to the operational modification, is also proposing installation of structural restraints by the use of tie-downs for the torus support columns. The tie-downs have been designed to resist the uplift force on the torus as determined in the first series of 1/12 scale tests; i.e., on the basis of an unpressurized drywell.

This information was presented by VYNPC in a meeting held in Bethesda, Maryland, with the NRC staff on February 3, 1976. VYNPC submitted this information to the NRC staff for review and evaluation by letter dated February 4, 1976.[1][2]

Evaluation

Containment Differential Pressure Control System Evaluation

Tests were conducted in the Mark I 1/12 scale test facility to determine the sensitivity of the upward pressure load acting on the torus over a range of reduced downcomer water leg height. A total of 14 tests were conducted varying break size, submergence, and drywell over-pressure. The data indicated that uplift loads would be reduced by operation with the pressurized drywell, mainly because of a reduced initial water level in the downcomer pipes which would results in earlier vent clearing.

The upward and downward torus loads currently calculated using this test data are based on data that represent in our opinion "best estimates" rather than conservative estimates in that conservatism involved in certain assumptions are to some degree offset by uncertainties in other areas, as follows:

(1) The torus load profiles specified for plant structural evaluation were developed from 1/12 scale test data without the application of a design margin or an error analysis. Margins have normally been applied to measured loads to establish that the load specification is conservative with respect to the actual data points and to account for uncertainties in the data acquisition.

(2) The forcing function for the 1/12 scale tests, from which the loads were derived, was based on the FSAR drywell pressurization transient and pure non-condensibles venting to the torus. This is clearly a conservatism; however, the methods employed by GE in measuring the loads in the 1/12 scale test and interpreting data introduced compensating uncertainties. The load cell data, which would have provided direct measurements of upward and downward loads could not be used directly because of the structural response of the test facility. This required that the forcing function be determined from averaging pressure transducer readings around the torus circumference, and analytically factoring in the vent header reaction load. Initial estimates made by GE and used in the staff review appear to be reasonable; however the final determination of the accuracy and the confirmation of the results will be achieved during our long-term review.

(3) The loads thus derived from 1/12 scale test data are directly applicable to only a reference plant design to which the test facility was modeled. Loads for other plants must be extrapolated from the base case due to variations in individual plant design parameters. These extrapolations included test data from the earlier Bodega Bay test facility. Our evaluation of these methods and data indicates that G.E. has made reasonable analyses to obtain the necessary torus load information for present usage; however, as noted in (2) above, the final determination of the adequacy of the total data base and the methods used for these extrapolations will be achieved during our long-term review.

We are working closely with the Mark I Owners Group, which includes the licensee, to more accurately quantify the various uncertainties in the existing data base and its application to torus uplift evaluations. There also exist concerns of a general nature related to three dimensional effects and scaling methodology would require additional larger scale testing to resolve. The short and long term programs put into effect by the owners group are designed to provide the information required to resolve these issues; not only for torus uplift loads but for all applicable pool dynamic loading mechanisms. Upon completion of these efforts, an adequately conservative basis will be established for pool dynamic loads to be applied in the final evaluation of Vermont Yankee and other plants with Mark I containments. In the interim period and considering that there are presently margins in the torus structural capability and that installation of tie downs will extend this capability, we believe that the load estimates used by the licensee reasonably reflect loads that may be anticipated in the event of a postulated LOCA, and therefore we would expect uplift of the torus to be less than one inch using the ΔP mode of operation. We further believe that the loads for the tie downs are reasonable.

The licensee has also considered the following effects on containment integrity due to CDPCS operation.

LOCA Blowdown

The licensee has reviewed the blowdown flow rate in light of the slightly higher initial drywell pressure. Since the flow is sonic during most of the blowdown transient, he has concluded there will be no effect due to CDPCS operation. We agree with this conclusion.

Peak Containment Pressure

The licensee has recalculated both the short and long term containment pressure LOCA transient considering a downcomer water leg depression of 3 feet (a differential pressure of approximately 1.7 results in a depression of about 3 feet). The analysis indicates an increase in the peak containment pressure value of about 0.3 percent. We find this level of increase to be negligible and therefore acceptable.

Inadvertent Containment Spray Operation

The licensee has evaluated the effect of the containment pressure response due to inadvertent drywell or wetwell spray actuation. Based on analysis, the applicant has calculated a minimum wetwell pressure of 13.2 psia which is above the design value of 12.7 psia. Therefore, we find the negative pressure differential acceptable.

Containment Isolation

The system requires a three inch line to be open continuously during plant operation. In the event of a LOCA, this line will be automatically isolated by means of one of three signals. They are low reactor water level, high drywell pressure, and high radiation in the reactor building. Based on the relative size of this line, and the addition of a requirement to assure a fast closure time of the three inch bypass valve (less than 5 seconds), we find this mode of operation acceptable during this short-term period.

Structural Capability of the Drywell During DCPCS Operation

The licensee has proposed the use of a differential pressure, ΔP, between the drywell and the torus in order to minimize the effect of torus uplift during the unlikely occurrence of a DBA. This differential pressure is achieved by maintaining a negative pressure of 0.45 psi in the torus and a positive pressure of 1.25 psi in the drywell. The pressure resisting capability of both of these structures is significantly greater than the combined pressures of "ΔP" and DBA in combination with other design loads. This is due to the significant margin of safety (to yield stresses) for the FSAR allowable stresses. Thus, we conclude that no adverse structural effects are anticipated due to the CDPCS operation.

Piping Deformation Capability

The licensee has provided a summary of a piping flexibility analysis for each piping system attached to the torus. This summary presents the stresses in the piping systems due to a one inch (1") upward deflection of the torus. Stresses in all piping lines due to this deflection are below the current ASME Code allowables. In addition, the licensee has inspected all piping attached to the torus to insure that there is a one inch (1") minimum clearance around each pipe, thus precluding impact with other pipes, structures or components.

Based on our review of these analyses and this visual inspection program, we conclude that a one inch (1") torus uplift is within the capability of the attached piping systems and that the required safety function of this piping can be maintained if subjected to one inch (1") torus uplift.

We conclude that operation of the CDPCS will reduce the torus uplift to less than one inch and is therefore acceptable and does not adversely effect other safety considerations.

Torus Support Column Modification

The licensee has proposed a hold down system (to be installed within 30 days) which modifies the torus support columns to resist the anticipated uplift forces during the unlikely occurrence of a DBA. The maximum torus uplift loads for the design have been computed to be 2.4 psi based on an unpressurized drywell. The calculational approach was identical to the methods used above for establishing torus upward loads for the pressurized drywell condition and therefore, our previously expressed conclusions regarding the use of these loads on an interim basis are equally applicable. It is also recognized that the margin provided by the structural fix can be increased by continued maintenance of a pressurized drywell during plant operation.

The torus hold down system consists of tie rods attached on either side of each torus support column. Each tie rod is attached to the column by a welded lug and clevis pin and at the other end to a base plate with another clevis assembly. Each tie rod has a turnbuckle in the middle for tightening. The base plate is held down by expanding bolts which anchor the base plate to the concrete mat.

The structural modifications of the Vermont Yankee torus hold down system are based on the following design parameters:

A. The uplift loads on the outer and inner columns are 240K per column.

B. The design strength of the concrete in the foundation mat is 400 psi.

C. The yield stress for the material used in base plates, lugs, and anchor bolts is 36 ksi and the yield stress for clevis pins is 65 ksi.

D. The pullout strength of a 1-1/2" diameter expanding anchor bolt is 55.5k per bolt for 4000 psi concrete.

The licensee has limited stresses in critical sections to values permitted by 1963 AISC Specification for the Design Fabrication and Erection of Structural Steel for Buildings as indicated in Table 12.2.1 of the FSAR. The stress limits permitted by the AISC specification compare conservatively with respect to those permitted by the ASME B & PV Code, Section III, Division 1, Subsection NF, 1974 for Class 2 and Class MC Component Supports for normal loading conditions.

The fabrication, installation, and construction of the torus hold down system will be performed in accordance with the requirements of the ASME B & PV Code, Section III, Division 1, Subsection NF or similar provisions provided for in the FSAR.

It is our conclusion, based on our review of the above, that the structural modification proposed by the licensee is acceptable. In addition, we conclude that the torus holddown system in conjunction with the differential pressure mode of operation will provide additional assurance that the torus is capable of withstanding the uplift loads due to the unlikely occurrence of a LOCA. In this regard, we will evaluate the need to continue the ΔP mode of operation after completion of the tie-down installation. We may conclude some relaxation of the requirements of operation with the ΔP mode could be made as our evaluation continues; however pending our further review, operation following installation of the tie downs shall be with both the ΔP mode of operation and the tie downs.

Inservice Inspection and Pipe Failure Probabilities

The licensee has performed a recent in-service inspection (ISI) and has presented a probabilistic analysis of large breaks in primary piping systems which will have a significant consequence on the torus integrity. One hundred and two (102) welds in the primary system piping have been identified which have a nominal pipe size of greater than 18" and have the postulated break area exceeding the critical value of 1.77 ft^2. Within the last month, visual liquid penetrant, and ultrasonic examinations conducted in accordance with the 1974 edition of the ASME Section XI Code were performed on 51 of these welds. No unacceptable indications were found by ultrasonic examination. One unacceptable indication was found by surface examination which was subsequently removed by surface grinding.

The licensee used an estimate of the failure probability of these pipe welds based on the expected number of pipe failures per plant per year as given in WASH-1400[4]. Starting with the median probability of 10^{-4} for pipe failures per plant year and taking into account the relevant information related to the Vermont Yankee torus problems, such as the number of pipe welds involved, the pipe size, and the number of days expected until the completion of structural modifications, it has been concluded by the licensee that the probability of a pipe failure which could impose high loadings on the torus is approximately 8.7×10^{-7} based on a sixty (60) day completion schedule for the structural modification.

We believe that the licensee's probability calculations have not been adequately supported in all respects. However, we do believe that the licensee's calculations provide an indication of the general order of magnitude of such failures.

The referenced WASH-1400 values are for piping greater than six inches in diameter which could initiate a pipe rupture in light water reactors of the type now in operation. The appropriate median value given in WASH-1400 for this piping is 10^{-4} per year with an uncertainty spread of from 10^{-3} to 10^{-5} per year.

Because of the lower failure probability for large pipes and taking into consideration the fraction of pipes in a nuclear facility that are very large (>18" in diameter), we believe that a median failure probability for large pipes approaches 10^{-5} per plant year. To further reduce the likelihood of a major pipe failure, all licensees are required to perform periodic in-service inspections. The recent inspection (within the last month) of half the welds in large primary system piping is in addition to such periodic inspections and provides additional assurance concerning piping integrity. In addition, the short period of time (30 days) that the facility will operate prior to the installation of hold-down devices will significantly reduce the likelihood of an unacceptable event.

We conclude that the probability of large pipe failure during the 30 days of operation prior to installation of hold-down devices is on the order of one chance in one million. The probability of a LOCA which leads to a containment failure is significantly lower than this value because of the use of differential pressure to assure containment integrity. On the basis of these low probabilities, we conclude that with differential pressure, the short period of operation prior to installation of the hold-down device is acceptable.

Operating Restrictions

The licensee has identified the upward and downward torus loads as significant loads and has proposed plant operating restrictions and structural modifications intended to limit torus movement such that containment integrity is maintained. Our evaluation of the acceptability of operation with the CDPCS and the structural modifications was based on the proposed operating restrictions being in effect. Therefore, we conclude that the operating restrictions identified in Appendix A to this evaluation should be implemented.

We determined that this action does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the action is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with this action.

Conclusion

We have concluded that operation of the Vermont Yankee facility with the existing plant licensing requirements and the LOCA load reduction resulting from the requirement to maintain a ΔP of 1.7 psid between the drywell and suppression chamber, along with consideration of the short period of time before plant modifications are completed to further enhance the structural capability of the containment system, and the low probability of a pipe failure of sufficient size to cause high containment loading, collectively provide reasonable assurance that the public health and safety will not be endangered. The additional licensing conditions to assure conformance with the pressure differential are set forth in Appendix A to this report.

Dated: FEB. 1 3 1976

REFERENCES

1. VY application dated February 4, 1976 (Proprietary).
2. VY application dated February 6, 1976 (non-Propriety version).
3. Mark I Containment Evaluation, Short-term Program - Final Report, NEDC 20989-1 September, 1975.
4. WASH-1400, Reactor Safety Study, October 1975.

APPENDIX A

1. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.7 psid except as specified in A and B below.

   A. This differential shall be established within 24 hours of achieving operating temperature and pressure.

   B. The above differential may be decreased to less than 1.7 psid for a maximum of two hours for the purpose of testing the drywell-supression chamber vacuum breakers.

2. If the differential pressure of 1.7 psid cannot be maintained, the plant shall be placed in the cold shutdown condition within 24 hours.

3. The pressure difference between the drywell and suppression chamber shall be recorded once per shift.

4. The volume of water in the torus shall be maintained between 68-70,000 Ft^3 and recorded once per shift.

5. The operability of the low differential pressure alarm shall be verified once every seven days.

6. The operability of the alternate circuit of the standby gas treatment system shall be demonstrated once every seven days.

7. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding three days unless such circuit is sooner made operable.

8. The operability and closure time (equal to or less than 5 seconds) of the three inch torus purge and vent outlet bypass valve shall be verified once every seven days.

9. The structural modifications to the torus support columns shall be installed within 30 days.