SAFETY ANALYSIS

RELATED TO

OPERATION WITH

THE NINE MILE POINT UNIT 2

MAIN STEAM ISOLATION BALL VALVES

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I. INTRODUCTION AND SUMMARY

This report describes the results of an analysis relating to use of the ball valves as main steam isolation valves at Nine Mile Point Unit 2. The possibility of main steam ball valve leakage in excess of Technical Specification Surveillance Requirements results after repeated operation of the valve due to delamination of the tungsten carbide coating, which is believed to be caused by local high contact stress between the ball seat and ball. Niagara Mohawk has evaluated this situation and has determined it may proceed with the planned start-up testing program and operation of Unit 2 after repairing or refurbishing the affected valves as necessary to meet Technical Specification leakage requirements. The factors that we considered in reaching this decision are summarized below:

1. The plant is and can be operated in conformance with regulations, regulatory guidance, the Operating License and Technical Specifications.

2. The history of leakage of the ball valves is at least comparable to a wye pattern globe valve.

3. The maximum ball valve leakage reasonably expected during this period of continued operation has been assessed (both by testing and calculation) and, using NUREG 1169 shown to be acceptable from a direct radiological health and safety viewpoint.

4. The Unit 2 FSAR analysis for radiological releases after a LOCA significantly overestimates the actual doses. Further, the assessment of the X/Q values used is conservative.

5. The probability of a large break LOCA is lower than previously believed.

This report shows that the ball valves are adequate for this service, meet the regulatory requirements and regulatory guidance and requirements of the Operating License and Technical Specifications.

II. HISTORY OF MAIN STEAM VALVE PERFORMANCE

Main steam isolation valve leakage has been a Nuclear Regulatory Commission Generic Issue for some time. In 1983, the staff reprioritized Generic Issue C-8, "MSIV Leakage and LCS Failures," as high priority. The results of the evaluation on this issue were reported in NUREG 1169. Niagara Mohawk has performed a comparison of the Unit 2 design to NUREG 1169 as described in Section III.B in order to assess the capability of the ball valves and the design bases of Nine Mile Point Unit 2.

Many BWR licensees have reported difficulty meeting the allowable leakage rate limit for periodic local leak rate tests (LLRTs). NUREG 1169 discusses a survey of MSIV performance at BWRs for the years 1979 through 1981 which found that 18 of 25 operating BWRs had MSIVs which failed to meet the maximum permissible leak rate limiting condition for operation during one or more surveillance tests. NUREG 1169 also indicates that during this time, a number of MSIV test failures exceeded 150 scfh for valves supplied by all three globe valve MSIV vendors. Although there has been general improvement in the performance of these globe valves, not all the problems have been resolved.

Niagara Mohawk was concerned with main steam isolation valve leakage in the early 1970's. On Unit 2 we decided to install a new type of valve for this application; namely, a ball valve. Each ball valve has a double sealing surface to prevent leakage past the valve. Even though the current problems of wearing and delamination have been experienced, the Unit 2 ball valves have still shown reasonably good leak tightness. The worst case leakages experienced on the installed main steam isolation valves (described in our final 10CFR50.55(e) Report dated October 20, 1986) to date have not exceed 16.9 scfh (see Attachment A test data). This value was obtained while testing through the valve. This compares well with a conservative calculation for estimating leakage through a scored valve which shows the leakage could be 29 scfh through the valves. This calculation is provided in Attachment B.

Further, our October 20, 1986 letter sets forth several attributes about the ball valve design which result in the conclusion that the valve is adequate for the service intended. Generally, main steam isolation valves have a low number of valve strokes, and it is this movement that causes wearing between the ball and seat. Previously, we provided test results to the Nuclear Regulatory Commission which documented that for the estimated 75 valve strokes during the first plant operating cycle, the test valve met its Technical Specification leakage criteria. Although some of the Unit 2 valves in the plant have continued to experience the delamination phenomena, all of the installed valves still show reasonably good leak tightness whether they are wearing or not. Further, the repeatibility of leak tightness, in our opinion, is better than the industry experience with the wye pattern globe valve (see Attachment A test information).

The startup program up to the 100-hour warranty run consists of about 35 weeks (250 days) and concludes with a main steam isolation valve closure and leak tightness check. During this time, we plan an additional 7 and estimate 10 unanticipated trips for a total of 17 main steam isolation valve strokes in addition to those which have occurred to date. Our new test results support continued plant operation at least through the first refueling outage, and show that the ball valves are at least comparable to the wye pattern globe valves which are in use throughout the industry and are capable of performing their intended safety function, and meet all regulatory, operating license and technical specification requirements.

III. REGULATORY COMPLIANCE REVIEW

A. Conformance to 10CFR50

One of the conditions of all operating licenses for water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in 10CFR50 Appendix J. These test requirements provide for pre-operational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for such tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the Technical Specifications and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

Therefore, Appendix J requires an initial determination that the main steam isolation valve leak rate is established within allowable limits, and that periodically testing will be performed to maintain the function during the service life. The Nine Mile Point Unit 2 ball valves will meet these requirements.

The periodic testing requirement in 10CFR50 Appendix J requires that Type C testing be performed at least once every 2 years. Further, 10CFR50 Appendix J indicates that this should occur at each reactor shutdown for refueling. Niagara Mohawk is proposing to test the main steam isolation valves in full conformance with these requirements. Also, an additional test is planned within 30 days of the completion of the 100-hour warranty run. Further, Niagara Mohawk has committed to additional testing as described in our October 20, 1986 letter because of the unique application of the ball valves for main steam line isolation. This testing includes additional prototype testing scheduled for the spring of 1987. The prototype testing includes the following attributes:

- Verification of the mechanical integrity of the valve and the actuator for the expected operating and test cycles.

- Demonstration of valve leak tightness for the expected valve duty cycles.

- Demonstration of the ability to close the valve within the Technical Specification limits under normal operating pressure, temperature and steam conditions.

- Verification of the conservatism of the between-the-seat leak test method as an alternative to across-the-valve seat leakage tests.

- Provision of baseline data for the evaluation of (1) the long-term suitability of the valve and (2) potential design and material improvements.

- The prototype test report, which will address the confirmation of the valves' acceptability for the first operating cycle, is to be provided to the Nuclear Regulatory Commission by May 15, 1987.

Further, 10CFR50 Appendix A General Design Criteria 54 and 55 require the main steam isolation valves to close to perform a containment isolation boundary function. During the past year, we have actuated the main steam isolation ball valves over 1000 times for testing purposes. During these tests, a small amount of tungsten carbide coating was abraded from the ball surface. This small amount of tungsten carbide coating has never interfered with full closure of any valve. This is the evidence that the valves will close upon a valid actuation signal, even in a degraded condition. In addition, the monthly surveillance testing involves 6 degree closing and opening, thereby providing periodic confirmation of valve operability.

Therefore, Unit 2 is in conformance with the regulations and the valves meet their licensing bases.

B. Evaluation of Nine Mile Point Unit 2 Utilizing NUREG 1169

The Nuclear Regulatory Commission established a review team to address MSIV leakage. Their review indicated that "the overall risks from the accident (LOCA) sequences in which MSIV leakage is a significant factor are low ... and alternate management schemes produce significant dose

reductions." To assess the ability of the ball valves to meet the safety requirements related to isolation and leakage control, Niagara Mohawk has performed a comparison to the Nuclear Regulatory Commission NUREG 1169 which shows that the results of this study are applicable to Nine Mile Point Unit 2. The comparison study is described in the following sections.

<u>Summary</u>

The purpose of NUREG 1169 was to determine: 1) the adequacy of industry efforts to identify and correct causes of excessive MSIV leakage, 2) the basis for any change in the allowable MSIV leakage rate, 3) the need for a safety-grade Leakage Control System (LCS), and 4) the specific areas of regulations and guidance that may be necessary to implement the findings. The approach was to evaluate the effects of MSIV leakage in terms of offsite doses following a LOCA, using realistic assumptions concerning the equipment, facilities and site characteristics available to mitigate the effects of a LOCA.

NUREG 1169 indicated that alternate treatment methods (discussed in the following sections) are highly effective in trapping the radioactivity such that the MSIV leak rate could be increased significantly without exceeding any dose limitations. Further, the NUREG indicates that the reliability of the leakage control system is actually lower than some of the alternate treatment methods. Therefore, the installation of LCS will not increase the safety of the plant and a MSIV leakage rate in excess of the Technical Specification will be adequately mitigated by existing alternate treatment methods.

This report demonstrates that the physical layout and design of Unit 2 is sufficient to ensure the health and safety of the public even were the Unit 2 MSIVs to leak in excess of their Technical Specification surveillance limits. This was demonstrated by comparing the reference plant in NUREG 1169 (WNP-2) and NMP2 and a calculation that shows it is acceptable. This calculation is provided as Attachment C. The calculation shows that a total for all main steam lines of up to 150 scfh leakage would not result in doses to the public and plant operators in excess of regulatory requirements specified in 10CFR100 and 10CFR50 Appendix A General Design Criteria 19, respectively.

COMPARISON OF PLANT PARAMETERS TO NUREG 1169

This section provides the comparison between the reference plant (WNP2) and NMP2.

Table I shows the comparison of the critical parameters affecting the radiological consequences of a LOCA. Tables II through VI show a detailed comparison between the reference plant for NUREG 1169 analysis and NMP2.

Some of the parameters governing the progression of a LOCA are the thermal power, thermal-hydraulic design, and the ECCs. Review of this data in Tables I, II, and III shows nearly identical design for both plants. Therefore, the expected source-term for NMP2 would be identical to NUREG 1169.

Another important concern is the MSIV leak rate. NMP2 Technical Specification surveillance leak rate is currently 30 percent lower than that assumed in NUREG 1169.

NMP2 steam lines are expected to cooldown at a somewhat higher rate than the reference plant, primarily due to higher thermal conductivity of the insulation and larger number of supports. This is expected to enhance particulate removal due to thermophoretic aerosol deposition.

Any leakage through the paths described in NUREG 1169 when applied to NMP2 would result in the radioactive releases being released either from the main stack or from the turbine building (dispersion factors are represented by the radwaste/reactor building (RW/RB) vent in Table I). In either case, the dispersion factors are comparable to those used in the NUREG 1169 analysis.

Since Unit 2 and the reference plant have similar systems, the NUREG 1169 probabilistic risk assessment scenario is judged to be equally applicable. In addition, NMP2 has an auxiliary boiler steam supply system to support operation of the Steam Jet Air Ejector Offgas pathway for MSIV leakage control (after isolation of the condenser from the reactor). This path is highly effective in removing and delaying radioactivity prior to release.

EVALUATION OF LEAKAGE CONTROL METHOD

This section compares the alternative leakage treatment methods discussed in NUREG 1169 to the as-built condition of the Nine Mile Point Nuclear Station - Unit 2.

The alternate leakage treatment methods contained in NUREG 1169 are:

1. Isolated Condenser

2. Mechanical Vacuum Pumps

3. Steam Jet Air Ejectors-Offgas System

4. Isolated Steam Lines

Each of these are discussed in relation to NMP2 as follows:

1. Isolated Condenser

   This leakage treatment method takes advantage of the main condenser to hold up the release of fission products from the main steam isolation valves (MSIV) and the main steam lines (MSL). The condenser itself is isolated from the turbine building and the outside environment. The method addressed in NUREG 1169 has two variations. The first path (using the turbine bypass valves) requires operator action to open the

bypass valves. The second method uses the main steam drain valves in lieu of the turbine bypass valves to connect to the condenser. The NUREG mentions that in some plants, the steam line drains are of a fail-open design, requiring no operator action.

The Nine Mile Point Unit 2 main steam line drains automatically open on loss of air power, first stage turbine pressure, or loss of signal. This is, as noted in NUREG 1169, a completely passive system; the main steam lines communicate with the condenser without operator action. It is also possible to connect the main steam lines to the condenser by way of the turbine bypass valves, but this would require operator action to initiate the turbine electrohydraulic (EHC) system. In all cases, the NMP2 isolated condenser leakage will migrate through the low pressure turbine seals into the turbine building and into the environment as described in NUREG 1169.

2. Mechanical Vacuum Pumps

The NUREG addresses the use of both condenser mechanical vacuum pumps and the gland seal and exhaust system blowers. By use of this equipment, the condenser is kept at a lower pressure than the surrounding environment. Thus, MSIV leakage will migrate through the main steam lines to the condenser, assuming that the MSL drains are open and/or the turbine bypass valves are open.

The mechanical vacuum pumps or gland seal exhausters will discharge the leakage to the stack (elevated release point). The NMP2 system design is the same as that described in the NUREG.

3. Steam Jet Air Ejector - Offgas System

NUREG 1169 describes a highly desirable mode of operation for control of MSIV leakage which uses the plant's existing steam jet air ejectors, steam seal system, and off-gas system to first collect the leakage and then discharge this leakage through the off-gas system where it is filtered, treated, and delayed. In addition, the discharge from the off-gas system is then sent to the stack (elevated release point).

The NMP2 installation meets all the recommendations for the most desirable steam jet air ejector-offgas system operation; that is:

> The NMP2 design incorporates two electric boilers, either one of which can produce a sufficient amount of steam to re-establish operation of the steam jet air ejectors and the gland seal and exhaust system if offsite power is available.(1) Therefore, the Unit 2 design can accomplish necessary filtering and delay of the radioactive gases.

(1) Further, if one of the Auxiliary Boilers were in operation prior to the LOCA, steam would be immediately available (as long as power is available). If one of the Auxiliary Boilers was not in operation prior to the LOCA, a delay (as long as 12 hours) in steam being available could occur.

4. Isolated Steam Lines

NUREG 1169 evaluates the condition in which the main steam lines are sealed off from the condenser, turbine, and environment. The main steam lines become a cavity to isolate the MSIV leakage from the environment. In this case, the turbine stop and control valves and bypass valves can pass some leakage which will eventually migrate to the environment. The NMP2 installation is again as described in the NUREG. However, this scenario has a low probability since the main steam drains automatically open upon loss of air or power.

NUREG 1169 contains a probabilistic analysis of each of the above paths including leakage control system pathway. Comparing the NMP2 plant with the analysis contained in the NUREG indicates that there are no differences in the NMP2 installation except as noted below:

a. Isolated Steam Line Flow Path

This condition has a low probability at the NMP2 plant, due to the fact that the main steam line drains automatically open on loss of air, power, or turbine first stage pressure. The more likely path is the isolated condenser.

b. Off-gas System Availability

The probability tree for the system using the steam jet air ejectors with the off-gas system and the steam seal and exhaust system is actually better for NMP2 because of higher availability than that contained in the NUREG. This improvement is described below:

1) Two electric boilers are installed, each of which can supply 40,500 lb. of steam per hour. With either boiler, the steam jet air ejectors, the steam seal and exhaust system, and the off-gas system can be maintained through the event.

2) the main steam line drains are a passive system, thus increasing their probability of opening during a LOCA.

3) In the event of a Loss of Offsite Power (LOOP) and LOCA with the NMP2 installation, the condenser vacuum, steam jet air ejectors, off-gas system, and gland seal exhaust system can be re-established once power is restored.

RADIOLOGICAL ANALYSIS IN CONFORMANCE WITH NUREG 1169

Based on the NUREG 1169 comparison with NMP2, the isolated condenser path is the conservative scenario for radiological consequences. The allowable MSIV leakage was determined based upon a simplified main condenser model for the beta skin and whole body gamma doses, while a direct comparison ratio method was used to determine the thyroid doses.

The beta and gamma dose evaluation model utilized hold up of the MSIV leakage in the main condenser and subsequent release of radioactivity to the environment. No credit was taken for hold up of noble gases in the main steam lines, drain lines or turbine building. Additionally, the volume reduction due to steam condensing in the piping or components prior to being released was not considered. The above conservative analysis provided results indicating that the most restrictive radiation limit was the Control Room beta dose. The analysis demonstrates that a maximum MSIV leak rate of 150 scfh total for all steam lines (as compared to a rate of 53 scfh* based upon testing) would not result in personnel doses in excess of regulatory limits. The maximum leakage rate could be increased to 500 scfh from the main steam lines with appropriate beta shielding (such as overalls). The calculation of radiological impact is provided in Attachment C.

* The tested rate represents individual MSIV leak rates of 17 scfh per valve. Credit was taken in three of four lines which would have two MSIVs closed in series resulting in 12 scfh per line. One of the four lines was conservatively assumed to have one MSIV closed. Therefore, the combined leakage rate would be (3 x 12) + 17 = 53 scfh

SUMMARY OF COMPARISON TO NUREG 1169

The following section describes the summary and conclusion to NUREG 1169:

NMP2 has in place what NUREG 1169 titled, "The Highly Desirable Mode of Operation," that is, a method of collecting, treating, and discharging from the stack all leakage from the main steam isolation valves. This is accomplished by:

a. A passive steam line drain system.

b. Electric boilers capable of providing steam to the steam jet air ejectors, off-gas system, and turbine gland seal and exhaust system.

c. In the event of a LOCA and/or LOOP, NMP2 has the capability to re-establish the condenser vacuum, the operation of the steam jet air ejector, the operation of the gland seal and exhaust system and the offgas system once off site power is restored.

Following simultaneous LOCA and LOOP, the NMP2 plant would automatically align itself in the condition defined as Isolated Condenser Pathway. It is unlikely that an isolated steam line pathway would occur for NMP2, a somewhat less effective method to control MSIV leakage.

The following conclusions are drawn from this analysis:

1. NMP2 has the capability to effectively control the MSIV leakage in a way similar to NUREG 1169.

2. NMP2 meets the NUREG description of the most desirable operating mode "steam jet air ejector offgas" which is available following a LOCA or temporary loss of offsite power.

3. NMP2 main steam lines are expected to cool down at a higher rate, thus further reducing offsite dose.

4. The radiological analysis has determined that leakages up to 150 scfh for all four mainsteam lines (or 500 scfh with beta shielding) would not result in personnel doses in excess of regulatory limits.

Therefore, considering the availability of the alternate methods for controlling the MSIV leakage, the presently installed ball type MSIVs are capable of performing their safety design function.

IV. ADDITIONAL CONSIDERATIONS

A. Summary

The following additional information is provided to demonstrate that continued operation through the first refueling outage would not present an undue risk to public health and safety.

1. The probability of a large break LOCA is lower than previously believed, as stated in SSER 3.

2. A portion of this period involves the initial startup testing with some operation below 100% power, during which several factors are applicable:

a. the core fission product inventory is lower than previously estimated (using the equilibrium value of the FSAR dose calculations).

b. there is additional operator response time available at lower power levels

c. the safety related system capacity (flow, heat removal, etc.) needed to mitigate an accident at low power is significantly lower than provided in the design

3. The application of regulatory guidance overestimates the actual radiological doses.

4. The assessment of the X/Q values is conservative.

B. <u>Discussion</u>

These conclusions are discussed in more detail below:

1. The probability of a large break LOCA is considerably lower than previously believed. This is particularly true for NMP Unit 2 because of the materials used for construction of the recirculation system.

In the NRC Safety Evaluation Report, Supplement 3 the staff agreed that "the probability of a large LOCA is now considered to be significantly lower than previously believed." This conclusion is based upon the following information:

The leak before the break (LBB) concept has led the staff to initiate rulemaking to modify GDC-4, excluding the double ended guillotine break (DEGB) from the set of design basis accidents. A new rule for pressurized water reactors ("PWRs") has been issued, and a similar rule for BWRs is scheduled for issuance.

Inasmuch as BWRs potentially have greater susceptibility to intergranular stress corrosion cracking (IGSCC), the Piping Review Committee recommended in NUREG-1061 (recommendation A-4, p.xi, Volume 5) that the recirculation piping in BWRs be replaced with alloys resistant to IGSCC, for example, with Type 316NG to reduce the probability of a DEGB. However, recirculation piping for NMP Unit 2 is already constructed of Type 316NG.

2. During the startup period, with some operation below 100% power, there are several factors which are related to safety as described below:

   a. The initial source term inventory of the core during the period from initial criticality to five percent power is low. Decreased fission product production and a relatively short

period of time for radionuclide buildup results in reduced core inventory. The DBA analyses utilize a core inventory equivalent to 1000 hours of operation at 105% of rated power.

b. Additional time is available during low power operation for the reactor operators to correct the loss of important safety systems or to take alternate courses of action.

c. During low power operation, the required capacity for mitigating systems are significantly lower than the design capacity. For example, the decay heat at 5 percent power which is generated by the LOCA is substantially less than that analyzed in the FSAR. Since the safety systems important to safety are designed for the mitigation of design basis events at 105% of rated power, there is ample margin to ensure minimal risk following a postulated low power accident.

3. The application of regulatory guidance overestimates the actual radiological doses. There are many factors which are neglected or which only partial credit is taken, that would substantially reduce these doses. These factors include:

a. Only partial credit for the plateout of isotopes in the core and containment is accounted for in accordance with the standard review plan.

b. The release from the containment to the environment (containment leakage not associated with the main steam isolation valves) is overestimated in the analysis. Actual leakage is below the design values used, as required by Technical Specifications.

c. The meteorological factors (wind speed and direction) which are discussed in more detail in item 4 below result in reduced doses.

d. The breathing rates and length of exposure time and availability of potassium iodide and protective clothing for the operating crew results in a overestimate of dose compared to realistic values. Potassium iodide could be used to reduce thyroid doses.

4. The assessment of the meteorological data and values is conservative. Dispersion near buildings is strongly affected by disturbances created by buildings. For releases close to buildings such as for the control room intakes, conventional equations cannot be used because the dispersion coefficients and mean velocities vary in space. For Nine Mile Point Unit 2 a plant specific evaluation was used which takes into account building wake effects. These assessments discussed in FSAR Section 2.3.4 use conservative wind speed, direction, joint distribution frequency, temperature and lateral and vertical plane diffusion values.

The differences between Design Basis and realistic meteorological parameters can be seen in FSAR Section 2.3.4.2 and Appendix F, Tables 2F-2 through 2F-11, which show large differences in X/Q.

V. CONCLUSION

Nine Mile Point Unit 2 is in conformance with the regulations and regulatory guidance regarding main steam isolation valve leakage. The probability of the occurrence of a LOCA is low; the radiological inventory and decay heat is lower than previously estimated. The application of regulatory guidance and the estimate of meteorological parameters overestimates the radiological doses. Continued operation with the ball valves is acceptable.

TABLE I

RADIOLOGICAL CONSEQUENCE PARAMETERS

PARAMETER	NMP2	NUREG-1169
Plant Type	BWR-5 Mark II	BWR-5 Mark II
Power (MWth)	3323	3323
Combined Technical Specification Leakage Rate for MSIVs, volume percent/day	<0.19	0.27
MSIV Leakage Pathways*	o Isolated Condenser o Isolated Steam Lines o Mechanical Vacuum Pumps o SJAE - Offgas System	o Isolated Condenser o Isolated Steam Lines o Mechanical Vacuum Pumps o SJAE - Offgas System
Steam Line Details		
Pipe Size	28"	30"
Wall Thickness	1 2/8"	1 3/8"
Insulation Thickness	4"	4"
Insulation Thermal Conductivity, Btu/ft-hr°F	0.03	0.02
Number of Pipe Supports	36+	29
Support Spacing	1' min 25' max	8' min 35' max

* Section III provides detailed discussion

\+ These supports are typical of one line; supports for headers and valves are not included

TABLE I

RADIOLOGICAL CONSEQUENCE PARAMETERS

PARAMETER		NMP2	NUREG-1169
Turbine System Supplier		GE	GE
Condenser Volume, ft^3		123,000	120,000
Condenser Horizontal Deposition Area, ft^2		214,000	252,000
Dispersion Factors*	RW/RB Vent	Stack	
EAB 0-2hr	19.00×10^{-5}	2.97×10^{-5}	7.50×10^{-5}
LPZ 0-8hr	1.78×10^{-5}	1.03×10^{-5}	2.80×10^{-5}
8-24hr	11.90×10^{-6}	0.88×10^{-6}	3.45×10^{-6}
1-4 days	4.93×10^{-6}	0.37×10^{-6}	1.59×10^{-6}
4-30 days	1.40×10^{-6}	0.10×10^{-6}	1.02×10^{-6}

* NUREG-1169 data taken from the WNP2 FSAR Chapter 15. This data have been used for a variety of release points in WNP2 FSAR.

TABLE II

NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

	NMP2	NUREG-1169
THERMAL AND HYDRAULIC DESIGN		
Design power, MWt (ECCS design basis)	3,463	3,468
Steam flow rate, millions lb/hr	14.27	14.30
Core coolant flow rate, millions lb/hr	108.5	108.5
Feedwater flow rate, millions lb/hr	14.56	14.26
System pressure, nominal in steam dome, psia	1,020	1,020
Average power density, kW/l	49.15	49.15
Minimum critical power flux ratio (MCPR)	1.24	1.24
Coolant enthalpy at core inlet, Btu/lb	527.5	527.6
Core max exit voids within assemblies	76.2	76
Core average exit quality, % steam	13.10	13.5
Feedwater temperature, °F	420	420
Design Power Peaking Factor		
Maximum relative assembly power	1.40	1.40
Axial peaking factor	1.40	1.40
REACTOR VESSEL DESIGN		
Material	Low-alloy steel/ stainless clad	Carbon steel/ stainless clad
Minimum base metal thickness (cylindrical section), in	6.1875	6.75
Minimum cladding thickness, in	1/8	1/8
Design pressure, psig	1,250	1,250
Design termperature, °F	575	575
Inside diameter, ft-in	20-11	20-11
Inside height, ft-in	72-5	72-11

TABLE II

	NMP2	NUREG-1169
REACTOR COOLANT RECIRCULATION DESIGN		
No. recirculation loops	2	2
Design pressure		
Inlet leg, psig	1,250	1,250
Outlet leg, psig	1,650(1) 1,550(2)	1,650(1) 1,550(2)
Design temperature, °F	575	575
Pipe diameter, in	24	24
Pipe material, AISI	316K	304/316
Recirculation pump flow rate, gpm	47,200	47,250
No. jet pumps in reactor	20	20
MAIN STEAM LINES		
No. steam lines	4	4
Design pressure, psig	1,250	1,250
Design temp. °F	575	575
Pipe material	Carbon steel	Carbon steel

(1) Pump and discharge piping to and including the discharge block valve.
(2) Discharge piping from discharge block valve to vessel.

TABLE III

EMERGENCY CORE COOLING SYSTEM
DESIGN CHARACTERISTICS

	NMP2	NUREG-1169
Low Pressure Core Spray System		
No. loops	1	1
Flow rate, gpm	6,350 @ 128 psid	6,250 @ 122 psid
High Pressure Core Spray System		
No. loops	1	1
Flow rate, gpm	1,550 @ 1,130 psid 6,350 @ 200 psid	1,650 @ 1,110 psid 6,250 @ 200 psid
Automatic Depressurization System		
No. systems	1	1
No. relief valves	7	7
Low Pressure Coolant Injection		
No. loops	3	3
Flow rate, gpm/pump	7,450 @ 26 psid	7,450 @ 20 psid

TABLE IV

CONTAINMENT DESIGN CHARACTERISTICS

	NMP2	NUREG-1169
Primary containment[1]		
Type	Over & under pressure suppression Mark II	Over & under pressure suppression Mark II
Construction	Reinforced concrete steel liner	Steel free standing
Drywell	Frustum of cone, upper portion	Frustum of cone, upper portion
Pressure suppression chamber	Cylindrical lower portion	Cylindrical lower portion with elliptical bottom
Pressure suppression chamber - internal design pressure, psig	45	45
Pressure suppression chamber - external design pressure, psig	4.7	2
Drywell - internal design pressure, psig	45	45
Drywell - external design pressure, psig	4.7	2
Drywell free volume, ft^3	303,418	200,540[2]
Pressure suppression chamber free volume (min), ft^3	192,028	144,184
Pressure suppression pool water volume (max), ft^3	154,794[4]	112,177[3] [4]

TABLE IV

	NMP2	NUREG-1169
Submergence of vent pipe below suppression pool surface, ft	9.5 min 11.0 max	11.67 min 12.00 max
Design environmental temperature of drywell, °F	340	340
Design environmental temperature of pressure suppression chamber, °F	270	275
Downcomer vent pipe pressure loss factor	1.37(5)	1.9
Break area/total vent area	0.0108	0.0105
Calculated maximum pressure after blowdown to drywell, psig	39.7	34.7
Calculated maximum pressure in suppression chamber, psig	34.0	28.0
Calculated maximum initial pressure suppression pool temperature rise, °F	50	35
Leakage rate, % free volume/day at 45 psig	1.1	0.5
Reactor Building		
Type	Controlled leakage, elevated release(6)	Controlled leakage, elevated release
Construction		
Lower levels	Reinforced concrete	Reinforced concrete
Upper levels	Steel superstructure and siding	Steel superstructure and siding
Roof	Steel decking	Steel decking

TABLE IV

	NMP2	NUREG-1169
Internal design pressure, psig	0.25	0.25
Design inleakage rate, % free volume/day at 0.25 in H_2O	100	100

(1) Where applicable, containment parameters are based on design power.
(2) Maximum water in suppression pool.
(3) Does not include water in the pedestal.
(4) At high water level.
(5) Includes entrance and pipe friction.
(6) For accident conditions.

TABLE V

ELECTRICAL POWER SYSTEM DESIGN CHARACTERISTICS

	NMP2	NUREG-1169
Offsite Power System		
Outgoing lines (No.-rating)	1-345kV	1-500kV
Incoming lines (No.-rating)	2-115kV	1-230kV 1-115kV
Onsite ac Power System		
Normal station service transformers	1	2
Reserve station service transformers	3(1)	2
Standby diesel generators	3(2)	3(2)
4,160V ESF buses	3(2)	3(2)
ESF buses	3-600-V(2)	3-480-V(2)
dc Power Supply		
Batteries (No.-volts)	6-125V(3) 4-24V	4-24V 5-125V(3) 1-250V
Buses (No.-volts)	6-125-V(3) 2-24V	2-24V 5-125V(3) 1-250V

(1) Includes one auxiliary boiler transformer.
(2) Includes an HPCS diesel generator.
(3) HPCS battery and bus included.

TABLE VI

POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

	NMP2	NUREG-1169
Design power, MWt	3,463	3,468
Design power, MWe, gross	1,202	1,205
Generator speed, RPM	1,800	1,800
Design steam flow, lb/hr	14.3×10^6	15.0×10^6
Turbine inlet pressure, psia	985	970
Turbine Bypass System		
Capacity, percent of turbine design steam flow	25	25
Main Condenser		
Heat removal capacity, Btu/hr	$7,830 \times 10^6$	$7,702 \times 10^6$
Circulating Water System		
No. Pumps	6	8
Flow rate, gpm/pump	105,000	82,000
Condensate and Feedwater Systems		
Design flow rate, lb/hr	14.917×10^6	14.260×10^6
No. condensate pumps	2 running, 1 stdby.	3 running
No. condensate booster pumps	2 running, 1 stdby.	3 running
No. feedwater pumps	2 running, 1 stdby.	2 running
Condensate pump drive	ac power	ac power
Condensate booster pump drive	ac power	ac power
Feedwater pump drive	ac power	Turbine
Heater drain pumps	3 running	

RESULTS OF

MSIV LEAKAGE TESTING

ATTACHMENT A

MSIV LEAKAGE TESTING

INTRODUCTION

This section describes testing performed at Nine Mile Point Unit 2 to demonstrate the leakage characteristics of the main steam isolation valves (MSIV).

ASSEMBLY AND PRE-OP TESTING

As described in the 10CFR50.55(e) Final Report related to the Main Steam Isolation Valves dated October 20, 1986, the MSIVs were reassembled with a revised seat design configured to minimize ball/seat contact stresses.

Pre-operational testing of the valves was performed, followed by Type "C" leak tests between the valve seats. Results indicated that all valves had acceptable leak rates with the exception of valve 6B. A tabulation of the valve stroking performed for pre-operational testing and the Type "C" test results are shown on Table 1.

Disassembly of valve 6B revealed considerable damage to the carbide coating on the ball and to the mating seat surfaces.

The bonnets of valves 6C, 6D and 7D were also removed and the ball conditions noted. Slight damage to the coating of these balls was evident. A remote video inspection of the remaining valves 7A, 7B, 7C and 6A revealed no apparent damage.

LEAKAGE TESTING

To quantify potential increases in valve leakage as a function of progressive coating damage, a test program was initiated. The test utilized the ball from valve 6B in the body of valve 7D. The ball from 6B was selected because it had sustained the most coating damage and, as such, was likely to produce the most conservative condition.

A combination of full and partial closures was used in the test program to bound valve stroking requirements up to the first refueling outage. Full fast closure of the valve represents planned strokes during heatup and up to the first refueling outage. Partial closure represents the Technical Specification Surveillance for the Reactor Protection System trip test performed monthly. A conservative estimate of anticipated valve operations for the first plant operating cycle is shown on Table 2. Valve stroking during the leak test program enveloped these requirements. Figure 1 shows the schedule of valve strokes during leakage testing and the subsequent Type "C" test results. Each "set" of strokes consisted of two full (90°) fast closures plus two partial (6° from full open) closures followed by a Type "C" test performed through the valve (across the seats).

CONCLUSION

The testing performed demonstrates that the leakage rate does not substantially increase when the valve is stroked with the ball wet or dry even after some ball coating damage has occurred.

It is anticipated that during normal plant operation, some amount of moisture due to steam condensation will be present on the outer surface of the ball. This film of condensate could assist in reducing damage to the ball, similar to the wet stroke tests described above. The effectiveness of this condensation will be confirmed during prototype testing underway at the valve vendor's facility.

The 6B ball used during the test had the most damage experienced recently since being refurbished and, as such had the potential for the greatest leakage. The remaining valves have substantially less or no damage and are anticipated to have leakage rates less than the test valve. All valves disassembled for evaluation will be reassembled and tested to meet Technical Specification limits.

TABLE 1

MSIV PRE-OP TEST RESULTS

MSIV	Full Closures (Fast)	Partial Closures (6°)	Type "C" * S.C.F.H.
6A	14	6	1.8
6B	10	11	8.9
6C	10.5	8	2.8
6D	13	6	1.8
7A	13.5	7	0.3
7B	9.5	9	0.6
7C	12	4	1.1
7D	8.5	8	1.1

* Leakage measured between the seats.

TABLE 2

MSIV OPERATIONAL SEQUENCE USED

TO ESTABLISH VALVE TEST CYCLES

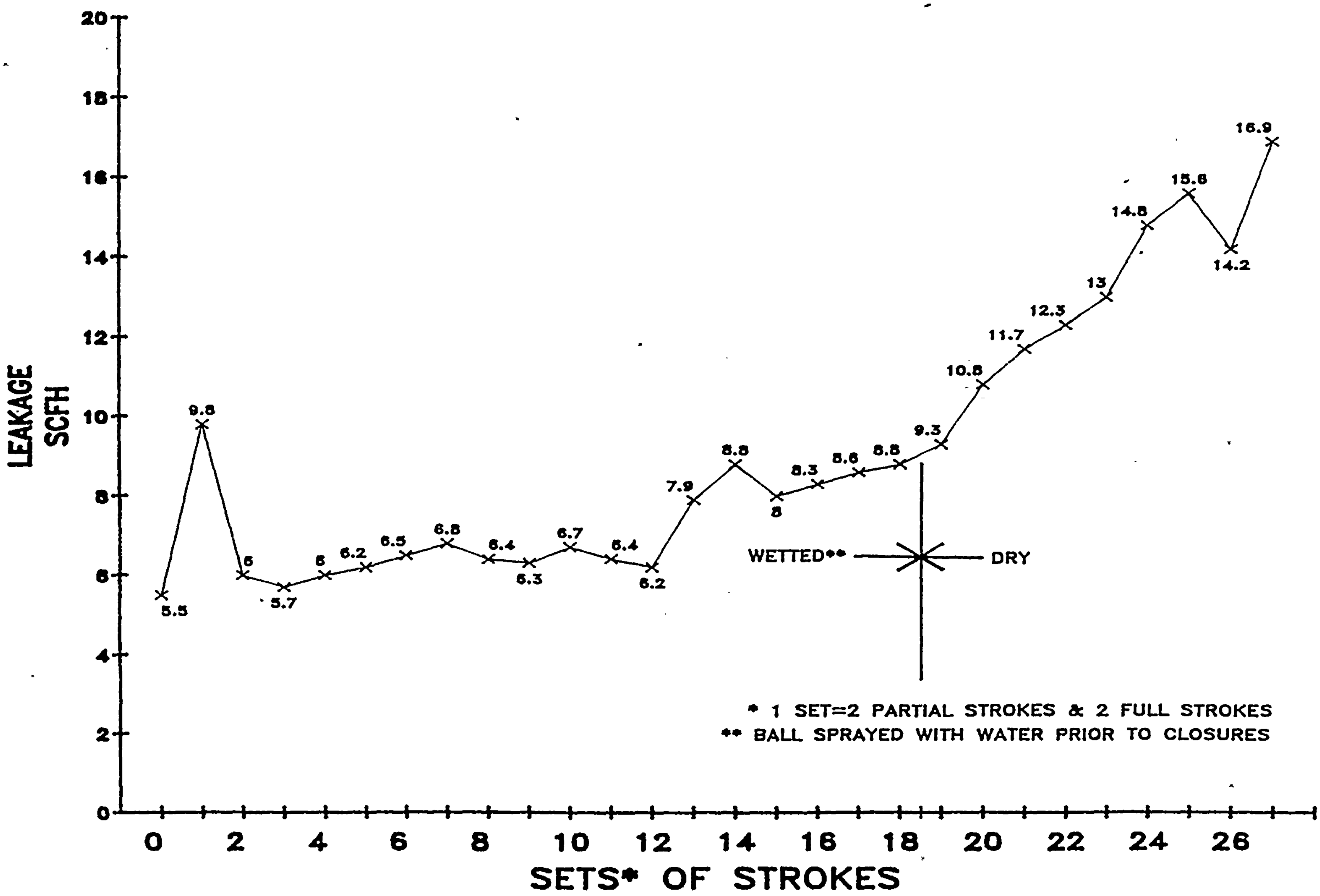
A.	Valve checkout prior to Type "C"	1 open/close (fast closure for 6A, B, C, & 7D actuator function check)
B.	Valve opening for plant heatup	1 full open
C.	Unanticipated trips	10 close/open (fast closure)
D.	Surveillance	1 cycle (partial closure 6°)
E.	Planned trips	5 close/open (fast closure)
F.	Surveillance	1 cycle (partial closure, 6°)
G.	100 Hour Warranty Run	1 closure (shutdown for Type "C" test)
	SUBTOTAL	FULL CYCLE-17 PARTIAL CYCLE-2
H.	Valve opening or plant heatup	1 full open
I.	Unanticipated trips	5 close/open (fast closure) (1st month)
J.	Unanticipated trips	5 close/open (fast closure) (2nd month)

K.	Surveillance	3 cycles (partial closure, 6°) (3-5 month)
L.	Mini-Outage	1 closure (shutdown)
	SUBTOTAL	FULL CYCLE-11 PARTIAL CYCLE-3
M.	Valve opening for plant heatup	1 full open
N.	Unanticipated trips	1 cycle (fast closure)
O.	Surveillance	3 cycles (partial closure 6°) (6-8 months)
P.	Unanticipated trips	1 cycle (fast closure)
Q.	Surveillance	3 cycle (partial closure 6°) (9-11 month)
R.	Unanticipated trips	1 cycle (fast closure)
S.	Surveillance	3 cycles (partial closure 6°) (12-14 month)
T.	Unanticipated trips	1 cycle (fast closure)
U.	Surveillance	3 cycles (partial closure 6°) (15-18 month)

V.	Planned trip	1 closure (shutdown)
	SUBTOTAL	FULL CYCLE-5 PARTIAL CYCLE-12
	TOTAL ESTIMATED CYCLES	FULL CYCLES-33 PARTIAL CYCLES-17
	TOTAL TEST CYCLES	FULL CYCLES-54 PARTIAL CYCLES-54

MSIV TYPE "C" TEST RESULTS
THROUGH THE VALVE

FIGURE 1



MSIV BALL VALVE

LEAKAGE RATE CALCULATION

ATTACHMENT B

STONE & WEBSTER ENGINEERING CORPORATION

CALCULATION TITLE PAGE

*SEE INSTRUCTIONS ON REVERSE SIDE

▲5010.64 (FRONT)

CLIENT & PROJECT	PAGE 1 OF 22
NIAGARA MOHAWK / NINE MILE POINT UNIT 2	PLUS: 1 ATTACHMENTS 12 pages
CALCULATION TITLE (Indicative of the Objective): LEAKAGE RATE CALCULATION THROUGH DAMAGED MSIV BALL/VALVE SEAT.	QA CATEGORY (✓) ☒ I - NUCLEAR SAFETY RELATED ☐ II ☐ III ☐ ______ OTHER

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* APPROVALS - SIGNATURE & DATE Tung Wang 1/14/87

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REF.

STATEMENT OF REVIEW

This calculation has been reviewed in accordance with CHOC-EMDM-82-12 and was found to be adequate.
The method of review was: (list the appropriate items)

a. Review of Calculation

b. Comparison with a similar previous calculation

1. Shil Lee 1/14/87 a 0
REVIEWER DATE METHOD REV.

2. REVIEWER DATE METHOD REV.

3. REVIEWER DATE METHOD REV.

The statement below applies to Nuclear Safety Related QA Category I calculations only.

This calculation has been INDEPENDENTLY reviewed in accordance with CHOC-EMDM-82-12 and was found to be adequate.
The method of review was: (list the appropriate items)

a. Comparison with prequalified method and assumptions ______
(prequalified document number(s))

b. Addressing the key questions appearing in EAP-5.3, and EAP-3.1 (latest revisions)

1. Y-S-Sun 1-14-87 b 0
INDEPENDENT REVIEWER DATE METHOD REV.

2. INDEPENDENT REVIEWER DATE METHOD REV.

3. INDEPENDENT REVIEWER DATE METHOD REV.

REVISON STATUS TABLE

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JOB ORDER NO. 12177

REV NO.	PAGE NO.	REASON	REVISOR/DATE	NON-INDEPENDENT REVIEWER/DATE	INDEPENDENT REVIEWER/DATE	APPROVAL/DATE
0	-	Original	NA			

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1.0 OBJECTIVE

The objective of the calculation is to calculate the anticipated leakage rate through the damaged Valve seats of the Main Steam Isolation Valve (MSIV). Leakage rates are calculated for the simulated plant condition in which pressurized (40 psig) air or steam leak through the upstream and the downstream seats in series as depicted in figure 1

P.6

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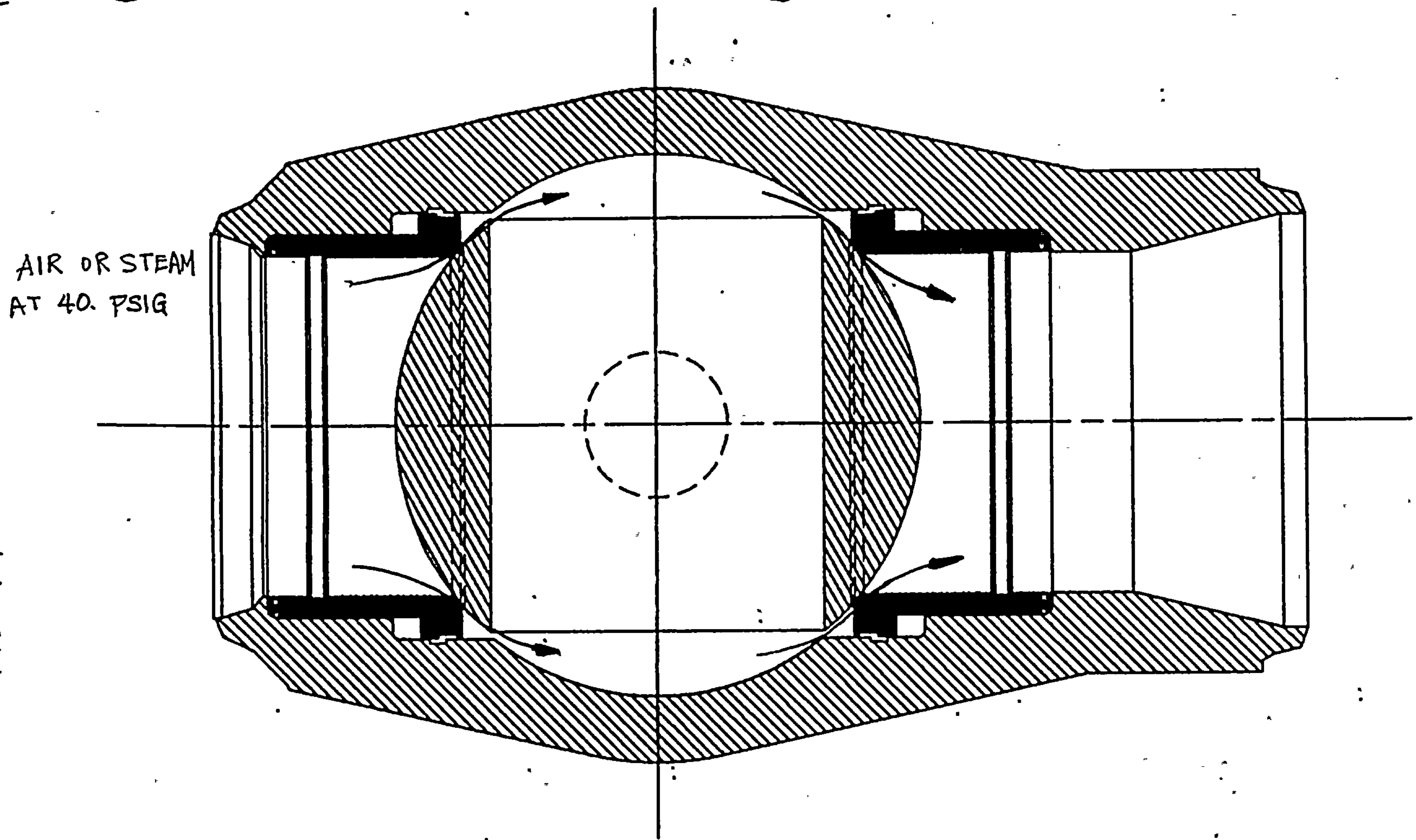

Figure 1 STEAM LEAK THROUGH SEATS

NO STEAM
FLOW-VALVE
FULLY CLOSED

'RE 5-5

NMMSIVMD7^

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2.0 Assumptions

1. The leakage path through the gouges and the associated dimensions are conservatively represented as in figure 3

2. Leakage flow is one-dimensional, frictional but adiabatic [illegible]

3. All leakage paths are assumed identical, i.e., the damage at the top portion and the bottom portion of the seats as well as those on the other seats are conservatively assumed to be the same so that the total leakage rate is two times the leakage rate per channel, per seat.

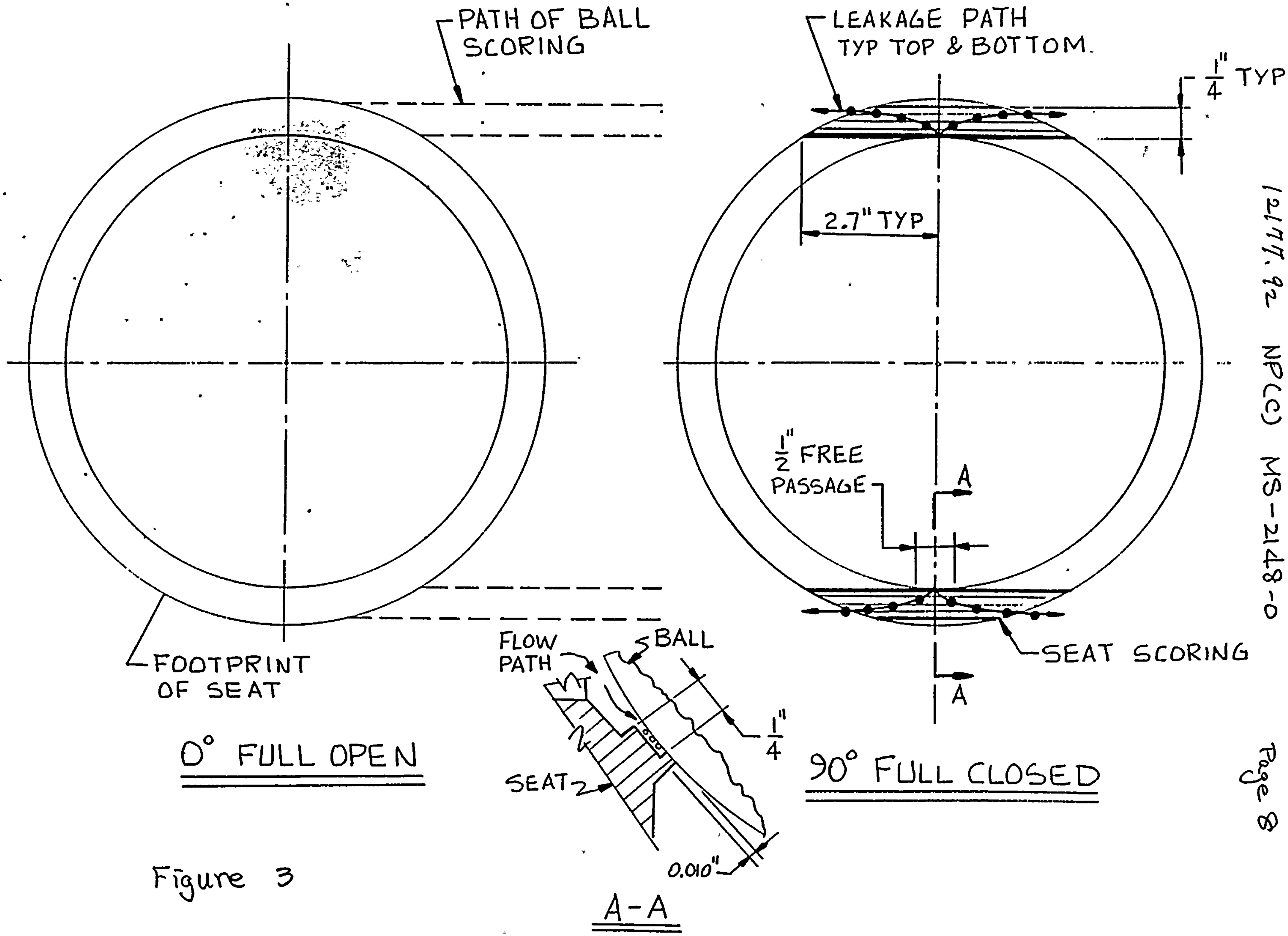

Figure 3

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3.0 METHOD OF CALCULATION

Field observation of the dis-assembled MSIV showed gouges with groves appearing on the upper and the lower portion of the seats (see figure 3). These gouges provide leakage paths for pressurized air or steam under test or postulated LOCA conditions. To provide a conservative estimate on the leakage rate, the flow paths are assumed as rectangular channels. The dimensions of the channel are conservatively estimated from the field observation of the damaged seats. The channels are assumed to be identical at both the upper and the lower parts of the seats and for both the upstream and the downstream seats.

Air or steam flow are considered as one-dimensional, adiabatic, frictional without appreciable heat transfer.

The equations of motion are given as (see, for example, reference 6.1)

$$\frac{dp}{p} = \frac{KM^2[1+(K-1)M^2]}{2(1-M^2)} 4f \frac{dx}{D_e}$$

$$\frac{dV}{V} = \frac{KM^2}{2(1-M^2)} 4f \frac{dx}{D_e}$$

$$\frac{d\rho}{\rho} = -\frac{KM^2}{2(1-M^2)} 4f \frac{dx}{D_e}$$

Where p = pressure

ρ = density

V = velocity

K = specific heat ratio

$M = \frac{V}{c}$ = Mach number

c = sound speed

f = Fanning friction factor

D_e = equivalent circular tube diameter

x = length of the channel from entrance

The system of equations are solved iteratively using a verified one-time use computer program until the upstream and the downstream pressures reach the specified values, i.e., 40 psig and 0 psig respectively.

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The total leakage rate per seat for the simulated plant condition is 2 times the leakage rate per channel, as there are 2 flow paths through the seat: (see figure 1)

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4.0 DESIGN INPUT

See section 2.0 Assumption and section 7.0 Analysis for numerical constant used in this section.

4.1 Simulated Plant Condition – Air flow

4.1.1 Flow from Upstream Pipe to Valve Body

Length of Equivalent Pipe Segment = .1106 FT.

Diameter of Pipe = 0.0016 FT

Friction Factor = .0175

Upstream Pressure = 54.7 psia

Upstream Density = 0.2639 lbm/ft^3

Specific Heat Ratio = 1.4

Downstream Pressure = (see note on next page)

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4.1.2 FLOW FROM VALVE BODY TO DOWNSTREM PIPE.

LENGTH OF EQUIVALENT PIPE SEGMENT = .1106 FT

DIAMETER OF PIPE = .0016 FT.

FRICTION FACTOR = .0175

UPSTREAM PRESSURE = (SEE NOTE)

UPSTREAM DENSITY = (SEE NOTE)

SPECIFIC HEAT RATIO = 1.4

DOWNSTREAM PRESSURE = 14.7 PSIA.

NOTE: PRESSURE AND DENSITY IN VALVE BODY WERE CALCULATED BY ITERATION SO AS TO MATCH THE FLOW RATE BETWEEN THE TWO PATHS.

4.2 SIMULATED PLANT CONDITION - STEAM FLOW.

4.2.1 FLOW FROM UPSTREAM PIPE TO VALVE BODY.

LENGTH OF EQUIVALENT PIPE SEGMENT = .1106 FT.

DIAMETER OF PIPE = .0016 FT.

FRICTION FACTOR = .0175

UPSTREAM PRESSURE = 54.7 PSIA.

UPSTREAM DENSITY = .1284 lbm/ft^3 (REF. 6.2).

SPECIFIC HEAT RATIO = 1.317 (REF. 6.2).

DOWN STREAM PRESSURE: (SEE NOTE ABOVE)

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4.2.2 FLOW FROM VALVE BODY TO DOWNSTREAM PIPE.

LENGTH OF EQUIVALENT PIPE SEGMENT = .1106 FT

DIAMETER OF PIPE = .001.61. FT.

UPSTREAM PRESSURE = (SEE NOTE ON PREVIOUS PAGE)

UPSTREAM DENSITY = (SEE NOTE ON PREVIOUS PAGE)

SPECIFIC HEAT RATIO = 1.317

DOWN STREAM PRESSURE = 14.7 PSIA.

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5.0 RESULTS SUMMARY

The leakage rates for simulated plant condition are summarized in following table.

FLOW MEDIUM	TOTAL LEAKAGE SCFH	TOTAL LEAKAGE lbm/Hr	REMARK
AIR AT 40 PSIG 100°F	29.09	2.219	
STEAM AT 40 PSIG Saturated	19.92	1.519	

6.0 REFERENCES:

6.1 A.H. SHAPIRO, "The Dynamics And Thermodynamics OF Compressible Fluid Flow" Vol. I The Ronald Press Co., 1953.

6.2 CRANE, TECHNICAL PAPER No. 410, "FLOW OF FLUIDS THROUGH VALVES, FITTINGS AND PIPE, 1976.

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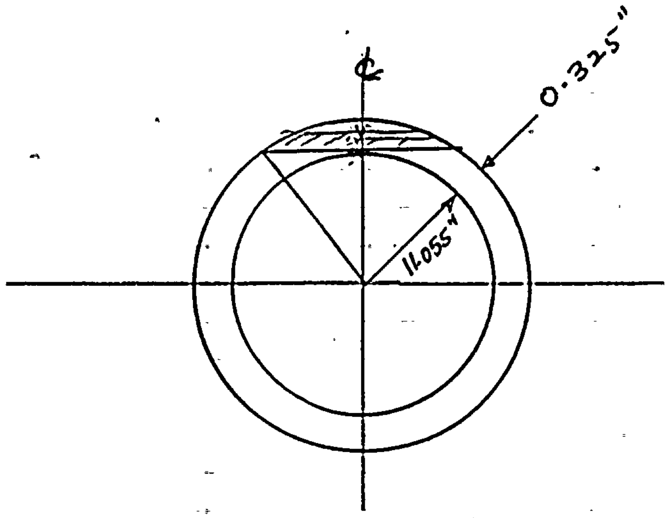
7.0 ANALYSIS

7.1 LENGTH OF FLOW PATH, TOP OR BOTTOM SEAL.



$$\text{FLOW PATH LENGH} = 2 \times \sqrt{(11.055+.325)^2 - 11.055^2} = 5.4''$$

SINCE FLOW IS SYMMETRICAL ABOUT ℄ 2.7" LENGTH IS USED, AND FLOW IS MULTIPLIED BY 2. TO GET FULL CHANEL FLOW

DIMENSION OF EQUIVALENT CIRCULAR PIPE

$$= \frac{4 \times \text{Flow Area}}{\text{Wetted Perimeter}}$$

RECTANGULAR CHANNEL OF DIMENSION $.25'' \times .010''$

$$De = \frac{4 \times .25 \times .010''}{2(.25 + .010)} = 0.0192 \text{ in.}$$

FRICTION FACTOR IS CALCULATED BASED ON ROUGHNESS OF CAST IRON PIPE, RELATIVE ROUGHNESS = .00085 (REF. 6.2)

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$$\epsilon/D = \frac{.00085 \times 12}{.0192} = .531 .$$

Fig A-23 OF REF. 62, USE MAXIMUM FRICTION FACTOR OF .07 BASED ON $\epsilon/D = .05$, WHICH IS CONSERVATIVE.

$$\text{FANNIG FRICTION FACTOR} = \frac{.07}{4} = .0175$$

PROPERTIES OF AIR AT 54.7 PSIA AND 100°F.

$$\text{Density of AIR} = \frac{144 P}{R T.} \quad lbm/ft^3$$

$$\rho = \frac{144 \times 54.7}{53.3 \times (460+100)}$$

$$= .2639 \ lbm/ft^3.$$

PROPERTIES OF AIR AT STANDARD CONDITION OF 14.7 PSIA AND 60°F

$$\rho = \frac{144 \times 14.7}{53.3 \times (460+60)} = 0.0763 \ lbm/ft^3$$

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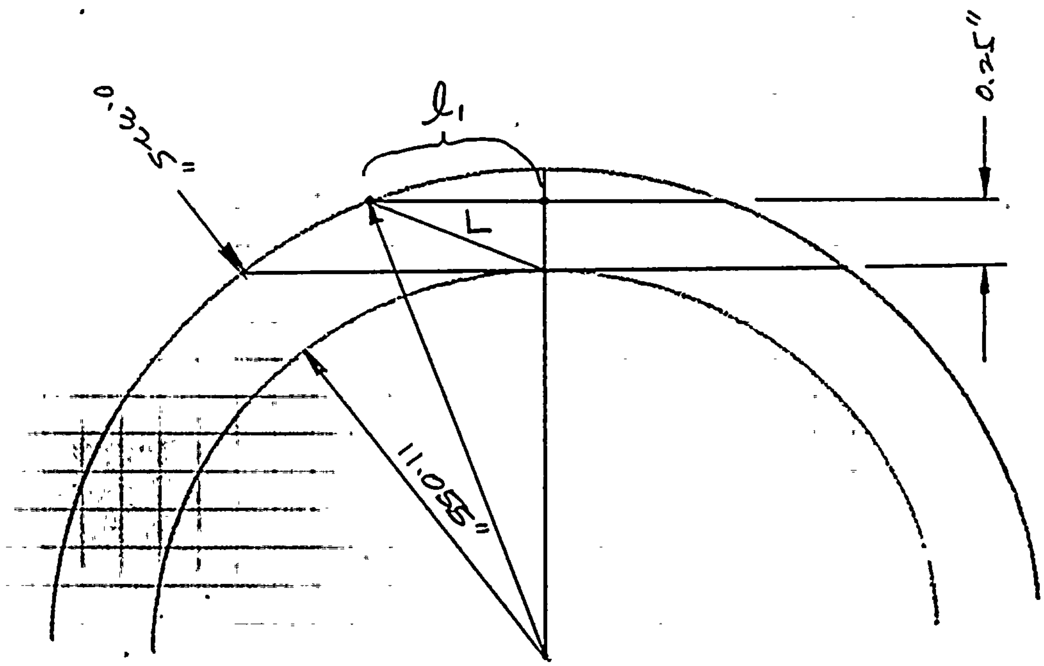
7.2 Calculation of Shortest Flow Path

$$l_1 = \sqrt{(11.055+0.325)^2 - (11.055+0.25)^2}$$

$$= 1.3043677''$$

$$L = \sqrt{1.3043677^2 + 0.25^2}$$

$$= 1.328'' = 0.1106 \text{ ft}$$



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7.3 COMPUTED LEAKAGE FLOW RATE.

LEAKAGE FLOW RATES ARE CALCULATED PER HALF THE CHANNEL BY THE COMPUTER PROGRAM IN lbm/HR. THESE ARE CONVERTED TO STANDARD CUBIC FEET PER HOUR HERE.

1. SIMULATED PLANT CONDITION. AIR FLOW.

   FLOW RATE = 0.555 lbm/HR (RUN # 3280)

   FLOW/CHANNEL = 0.555 × 2/.0763 = 14.547 SCFH

2. SIMULATED PLANT CONDITION, STEAM FLOW.

   FLOW RATE = 0.38 lbm/HR (RUN # 3511)

   FLOW/CHANNEL = .38 × 2/.0763 = 9.96 SCFH

IMPACT OF INCREASING THE MSIV LEAKAGE

FLOWRATE ON POST LOCA DOSES

UTILIZING CREDITS BASED UPON NUREG 1169

ATTACHMENT C

STONE & WEBSTER ENGINEERING CORPORATION

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*SEE INSTRUCTIONS ON REVERSE SIDE

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NIAGARA MOHAWK POWER CORP./NINE MILE POINT - UNIT 2	

CALCULATION TITLE (Indicative of the Objective):	QA CATEGORY (✓)
IMPACT OF INCREASING THE MSIV LEAKAGE FLOWRATE ON POST-LOCA DOSES AT EAB, LPZ, CONTROL ROOM, AND TSC UTILIZING CREDITS BASED ON NUREG-1169	☑ I - NUCLEAR SAFETY RELATED ☐ II ☐ III ☐ ____ OTHER

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12177	PR(C)	28-M	NA	18A

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CLIENT/PROJECT	QA CATEGORY / CODE CLASS
NIAGARA MOHAWK POWER CORP. / NINE MILE POINT - UNIT 2	I / NA

SUBJECT/TITLE IMPACT OF INCREASING THE MSIV LEAKAGE FLOW RATE ON POST-LOCA DOSES AT EAB, LPZ, CONTROL ROOM, AND TSC UTILIZING CREDITS BASED ON NUREG-1169

OBJECTIVE OF CALCULATION THE PURPOSE OF THE CALCULATION IS TO EVALUATE THE IMPACT OF INCREASING THE MSIV LEAKAGE FLOWRATE ON THE POST-LOCA DOSES AT THE EAB, LPZ, CONTROL ROOM, AND THE TSC TAKING CREDITS OF NUREG-1169. THIS ANALYSIS IS BASED ON THE ISOLATED CONDENSER (STEAM LINE CONDENSATE DRAINS OPEN) LEAKAGE TREATMENT METHOD OF NUREG-1169.

CALCULATION METHOD/ASSUMPTIONS SEE METHOD SECTION - P.5

SOURCES OF DATA / EQUATIONS

SEE PAGES 7 TO 9

CONCLUSIONS THE MOST LIMITING MSIV LEAKAGE FLOWRATES ARE BASED ON BETA DOSE LIMITS AT THE CONTROL ROOM WHICH ARE AS FOLLOWS:

TOTAL ALLOWABLE LEAKAGE* = 150 SCFH (BETA LIMIT OF 30 REM)

TOTAL ALLOWABLE LEAKAGE* = 500 SCFH (BETA DOSE LIMIT OF 75 REM W/ PERSONNEL BETA SHIELDING)

FOR OTHER ALLOWABLES - SEE P. 38 & 39

* BASED ON LEAKAGE FROM FOUR MAIN STEAM LINES

REVIEWER(S) COMMENTS		DATE
	PREPARER M. Hasan / M. H. Tofani	1/14/87
	REVIEWER/CHECKER R. J. Kliniewski	14 Jan 1987
	INDEPENDENT REVIEWER R. J. Kliniewski	14 Jan 1987

OBJECTIVE

The purpose of the calculation is to evaluate the impact of increasing the MSIV leakage flow rate on the post-LOCA doses at the EAB, LPZ, control room, and TSC taking credits based on NUREG-1169. Maximum allowable MSIV leakage flowrates based on gamma, beta, and thyroid dose limits are calculated taking credit for non-category I structures and components. Any potential affects on personnel exposures due to equipment failure resulting from changes in the releases is not considered.

This analysis is based on the isolated condenser (steam line condensate drains open) leakage treatment method of NUREG-1169 (Ref 6).

METHOD

NUREG-1169 (Ref 6) is reviewed to determine credits which can be taken to reduce doses with increased MSIV leakage rates. Note that this analysis is based on the isolated condenser case of NUREG-1169. This case uses the large volume of condenser to holdup the release of activities leaking from the MSIV and down the main steam line and into the drain lines to the condenser hotwell. Several DRAGON runs are made to compare the dose reductions possible. The DRAGON model which provides the maximum dose reduction is used further to determine the maximum allowable increase of the MSIV leakage rate based on gamma and beta* doses at the EAB, LPZ, and the control room. Note that for NMP2 thyroid dose evaluation, the NUREG-1169 values at the LPZ are utilized since

* Evaluated only for control room and TSC

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CALCULATION SHEET

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the iodine plateout model has not been established for NMP2.

The core inventory used in DRAGON models is from Ref. 5.

The total beta* and gamma doses at the EAB, LPZ, and the control room versus MSIV leakage rate are plotted.

The maximum allowable increase in the MSIV leakage rates is determined from the plots corresponding to the regulatory limits.

* Evaluated only for control room & TSC

DATA/ASSUMPTIONS

(1) The reactor is at 105% of full power at the time of the accident.

(2) 100% of the core noble gases and 25% of the core halogens are available for release from the primary containment to the environment. (Ref 1)

(3) Core inventory at 105% power is taken from p.19 of Ref 5.

(4) Main steam tunnel K/Q's are used for release from the turbine building. The K/Q's are as follows: (p.20 & 21, Ref 5)

EAB 0-2 hrs 1.90-4 Sec/M^3

	Control Room	LPZ
0-8 hrs	1.29-3 Sec/M^3	1.78-5 Sec/M^3
8-24 hrs	9.90-4 Sec/M^3	1.19-5 Sec/M^3
24-96 hrs	3.37-4 Sec/M^3	4.93-6 Sec/M^3
96-720 hrs	9.92-5 Sec/M^3	1.40-6 Sec/M^3

(5) Minimum drywell free volume = 2.85+5 CFT (p.22, Ref 5)

(6) Control room (emergency control room ventilation system pressure envelope) free volume = 3.81+5 CFT (p.22, Ref 5)

(7) Emergency control room ventilation system flowrates: (p.22, Ref 5)
Outside air intake = 1500 CFM
Recirculation = 750 CFM

(8) Control room filters halogen removal efficiency = 99% (p.22, Ref 5)

(9) The actual leakage rates from the drywell corresponding to 6 SCFH leakage through a MSIV (based on isentropic-one valve - worst case): (p. 708, Ref 7)

0-8 hrs	0.197-3	Fraction/day
8-24 hrs	0.191-3	Fraction/day
24-96 hrs	0.180-3	Fraction/day
96-720 hrs	0.141-3	Fraction/day

(10) Condenser is assumed to be at atmospheric conditions from T=0 to 30 days post-LOCA.

(11) Condenser outflow rate due to daily and weather front related barometric changes is assumed to be ≈ 100 cft/hr from T=0 to 30 days post-LOCA. (p. F-3, Ref 6).

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(7) The volumetric flow of noble gases into the condenser is assumed to be the actual flow rate of the air/steam mixture leaking through the MSIV. In other words no credit is taken for the condensation of steam causing the reduction in flow rate. This is conservative.

(8) Condenser free air volume
= 123,000 CFT (Ref 22)

*(9) It is assumed that the reference plant analyzed in NUREG-1169 (Ref 6) has similar operating and design characteristics as that of NMP2.

(10) NUREG-1169 reference plant (WNP2) X/Q's: (Ref 23)

EAB	0-2 hr	7.50-5	Sec/M³
LPZ	0-8 hr	2.80-5	Sec/M³
	8-24 hr	3.45-6	Sec/M³
	1-4 days	1.59-6	Sec/M³
	4-30 days	1.02-6	Sec/M³

* CONFIRMATION REQUIRED - A preliminary comparison study has been performed. However, the results of the study are not yet documented formally.

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CALCULATION

First three cases are analyzed using the DRAGON code:

(1) Drywell leaking directly to the control room

(2) Drywell leaking to the turbine bldg. and to the control room (no holdup credit in the condenser is taken)

(3) Drywell leaking into the main condenser via the steam line drains and out to the control room (no dilution/holdup credit taken for the turbine bldg.)

Note that the above dose evaluation is performed only for the control room for determining the model which will provide the maximum dose reduction. This is done since control room beta and gamma doses turned out to be most limiting based on review of results of Ref 5 and 10.

(1) First DRAGON Run (Input Card Image on p.41)

MSIV leakage rate used = 50 SCFH/valve

Total leakage for 4 MSIV = 200 SCFH

Actual leakage from drywell based on drywell pressure and temp conditions:

0-8 hrs $= 0.197\text{-}3 \times \frac{200}{6}$

$= \underline{6.56\text{-}3}$ Fraction/day

8-24 hrs $= 0.191\text{-}3 \times \frac{200}{6}$

$= \underline{6.37-3}$ Fraction/day

24-96 hrs $= 0.180-3 \times \frac{200}{6}$

$= \underline{6.00-3}$ Fraction/day

96-720 hrs $= 0.141-3 \times \frac{200}{6}$

$= \underline{4.70-3}$ Fraction/day

Multipliers used in DRAGON to reduce leakage

8-24 hrs $= \frac{6.37-3}{6.56-3} = 0.97$

24-96 hrs $= \frac{6.00-3}{6.56-3} = 0.91$

96-720 hrs $= \frac{4.70-3}{6.56-3} = 0.72$

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(2) <u>Second DRAGON Run</u> (Input Card Image on p.42)

Drywell leakage rates thru MSIV are same as the first run.

Turbine bldg gross volume to be used in the second DRAGON run is calculated based on the following criteria:

(1) The turbine bldg volume where the mixing and dilution of activities released from the condenser will most likely occur.

(2) The area where the turbine bldg outleakage will occur. The turbine bldg. ventilation system is assumed to be not operating.

The likely place where the release from the condenser will occur is the turbine seals which are located at the operating floor el 306'-0". Based on a review of Ref 9 through 18, concrete walls are above and below el 306'-0". However, it is estimated that taking

the volume above el 306'-0" will give a reasonable value and this should offset the areas below and above el 306'-0". Based on the above and Ref 8, the following zone volumes are included:

Zone	2	=	2,375,210	ft^3
"	3	=	148,195	"
"	4	=	164,300	"
"	5	=	148,195	"
"	6	=	30,969	"
"	8	=	198,128	"
"	49	=	15,415	"
"	50	=	9,863	"

Total Gross Volume

= 3,090,275 cft

Assuming 80% is free volume

Turbine Bldg. Free volume to be used in DRAGON Model

= 0.8 X 3,090,275 cft

= 2,472,220 cft

The turbine bldg. outleakage is based on the amount of inleakage required to maintain slightly negative pressure inside the bldg which is equal to 8,700 CFM (Ref 19). The use of this value is considered to be conservative for use as outleakage.

Fraction/day leakage of the turbine bldg. free volume

$$= \frac{8700 \frac{FT^3}{MIN} \times 60 \frac{MIN}{HR} \times 24 \frac{HR}{DAY}}{2{,}472{,}220\ FT^3}$$

$$= \underline{5.068}\ \text{Fraction/day}$$

NOTE: Any inaccuracies noted in the turbine building release model are of little concern, since the turbine building provides very little holdup of noble gases, and was not included in the final model.

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(3) Third DRAGON Run (Input Card Image on p. 43)

Drywell leakage rates through MSIV's are the same as the first run.

The condenser outleakage is calculated as follows:

Cond. Outleakage = Total MSIV Leakage at standard conditions (D/A #10) + Condenser outflow due to barometric changes (D/A #11)

$= 200 + 100$

$= 300$ CFH

Condenser Air Volume

$= 123{,}000$ CFT (D/A #8)

Fraction/day of Condenser volume

$$= \frac{300\ \text{CFT/HR} \times 24\ \frac{\text{HR}}{\text{DAY}}}{123{,}000\ \text{CFT}}$$

$= \underline{5.85-2}$ Frac/day

(1) <u>First DRAGON Run</u> (Run # R2534C01, Job # 3260, 1/7/87)

Drywell leakage to Control Room

30-Day Beta Dose = 121 Rem

30-Day Gamma Dose = 5.2 Rem

(2) <u>Second DRAGON Run</u> (Run # R2534C01, Job # 4219, 1/7/87)

Drywell leakage to Turbine Building to Control Room

30-Day Beta Dose = 97.8 Rem

30-Day Gamma Dose = 3.5 Rem

(3) <u>Third DRAGON Run</u> (Run # R2534C01, Job # 5385, 1/8/87)

Drywell leakage to condenser to control room

30-Day Beta Dose = 12.3 Rem

30-Day Gamma Dose = 2.75-1 Rem

Beta Dose Reduction via Turbine Bldg leakage path $= \frac{121}{97.8}$

$= 1.24$

Gamma Dose Reduction via Turbine Bldg. leakage path $= \frac{5.2}{3.5}$

$= 1.49$

Beta Dose Reduction via Condenser leakage path $= \frac{121}{12.3}$

$= 9.84$

Gamma Dose Reduction via Condenser leakage path $= \frac{5.2}{2.75-1}$

$= 18.9$

As can be seen the beta dose reduction by a factor of ≈10 is possible if the condenser path is used as compared to a reduction by a factor of 1.24 for the turbine bldg path. Similarly, gamma dose reduction by a factor of 18.9 is possible with the condenser path compared to 1.49 with the turbine bldg path. It is concluded that the DRAGON model with the condenser will be used in further analysis since this gives a maximum dose reduction. Note that turbine bldg. is not included for ease

of modelling in DRAGON code. Therefore, results will be somewhat conservative.

I CONTROL ROOM GAMMA AND BETA DOSE EVALUATION

Gamma and beta doses in the control room are calculated using DRAGON model, drywell to condenser to control room, determined on p. 17 to give the maximum gamma and beta dose reduction. Leakage flow rates of 11.5, 25, 50, 100, 200, & 250 SCFH per MSIV are analyzed.

Ref 5 is reviewed to determine the dose contributions from the 4 main steam lines and the allowable increases to reach the regulatory limits. The results of the review are as follows:

Total Gamma Dose = 1.81 Rem (p.66, Ref 5)

Contribution from 4 main steam lines to total gamma dose = 7.28-2 Rem (p.F2, Ref 5)

Gamma Dose due to other sources

= (1.81) - (7.28-2)

= 1.74 Rem

Gamma Dose Limit = 5 Rem (Ref 3 & 4)

Total Beta Dose = 25.11 Rem (p. 66, Ref 5)

Beta Dose Contribution from 4 main steam lines = 3.65 Rem (p. F2, Ref 5)

Beta Dose Contribution from other sources = 25.11 - 3.65

= 21.46 Rem

Beta Dose Limit = 30 Rem (Ref 3 & 4)

Beta Dose Limit w/ personnel beta shielding

= 75 Rem (Ref 3 & 4)

Note that drywell actual leakage rate into the condenser at 11.5, 25, 50, 100, 200, & 300 SCFH/valve at T=0 is calculated using D/A #9 and input to DRAGON code. The condenser outleakage is calculated equal to MSIV leakage/valve in SCFH X 4 + condenser outflow of 100 CFH due to barometric changes.

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Total Gamma Dose in Control Room

@ 25 SCFH/MSIV (Total 100 SCFH) $= (1.74) + (9.84\text{-}2)$

p.19

RUN#R2554C01, JOB#5339, 1/8/87

$= \underline{1.84}$ Rem

@ 50 SCFH/MSIV (Total 200 SCFH) $= (1.74) + (2.75\text{-}1)$

RUN#R2554C01, JOB#5385, 1/8/87

$= \underline{2.02}$ Rem

@ 100 SCFH/MSIV (Total 400 SCFH) $= (1.74) + (8.15\text{-}1)$

RUN#R2554C01, JOB#5430, 1/8/87

$= \underline{2.56}$ Rem

@ 200 SCFH/MSIV (Total 800 SCFH) $= (1.74) + (2.44)$

RUN#R2554C01, JOB#5444, 1/8/87

$= \underline{4.18}$ Rem

@ 300 SCFH/MSIV (Total 1200 SCFH) $= (1.74) + (4.58)$

RUN#R2554C01, JOB#6021, 1/8/87

$= \underline{6.32}$ Rem

The above results are plotted in Figure 1.

In accordance with Figure 1, the maximum allowable MSIV leakage rate/valve based on gamma dose limit of 5 Rem

= 240 SCFH/MSIV

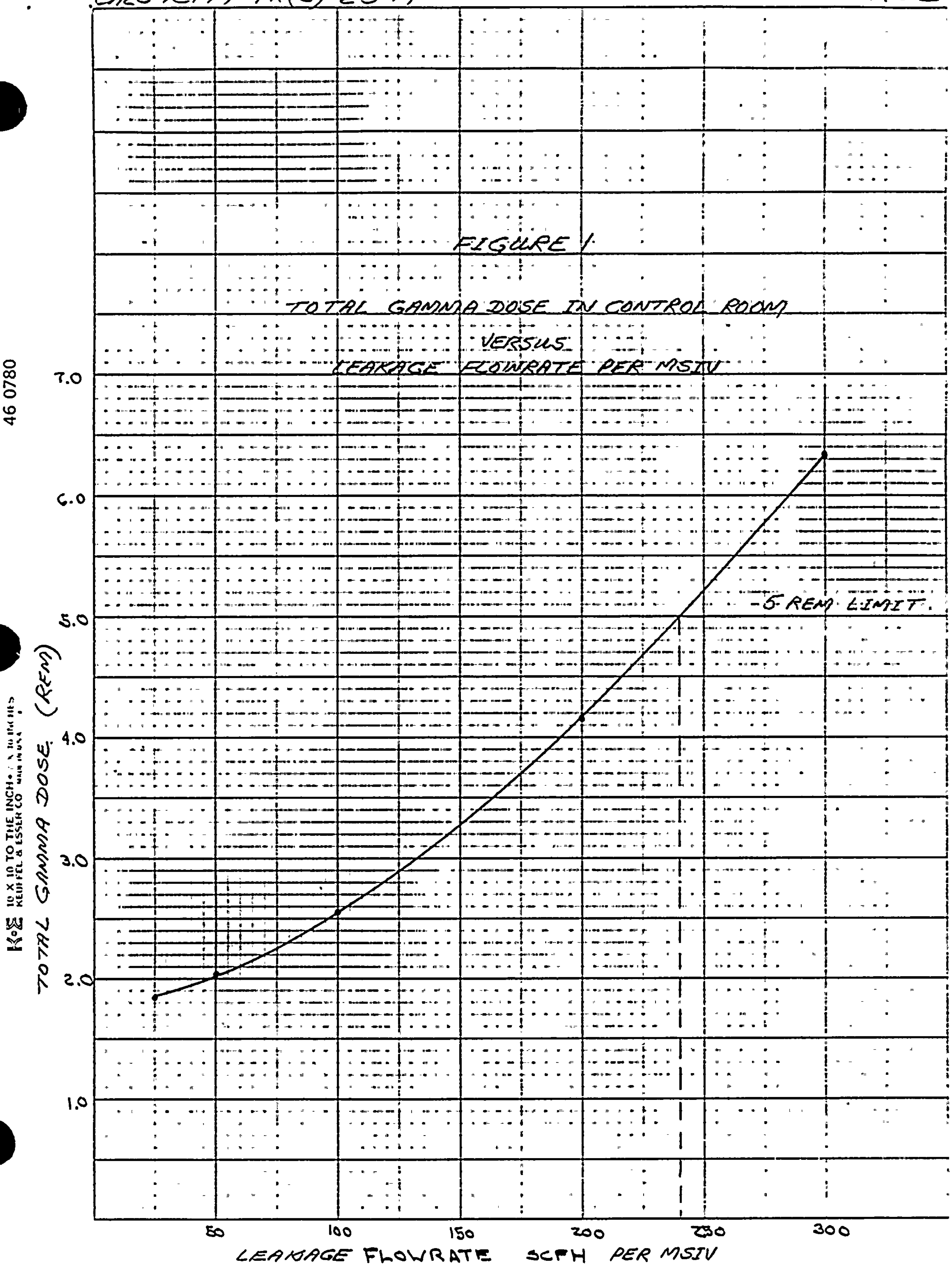
FIGURE 1
TOTAL GAMMA DOSE IN CONTROL ROOM
VERSUS
LEAKAGE FLOWRATE PER MSIV
5 REM LIMIT.
7.0
6.0
5.0
4.0
3.0
2.0
1.0
TOTAL GAMMA DOSE (REM)
50
100
150
200
250
300
LEAKAGE FLOWRATE SCFH PER MSIV
K-E 10 X 10 TO THE INCH 46 0780

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Total Beta Dose in Control Room

@ 25 SCFH/MSIV (Total 100 SCFH) = 21.46 + 4.48

21.46 → p. 20; 4.48 → RUN # R2554C01, JOB # 5339, 1/8/87

= 25.94 Rem

@ 50 SCFH/MSIV (Total 200 SCFH) = 21.46 + 12.3

12.3 → RUN # R2554C01, JOB # 5385, 1/8/87

= 33.76 Rem

@ 100 SCFH/MSIV (Total 400 SCFH) = 21.46 + 35.5

35.5 → RUN # R2554C01, JOB # 5430, 1/8/87

= 56.96 Rem

@ 200 SCFH/MSIV = 21.46 + 101.0

101.0 → RUN # R2554C01, JOB # 5444, 1/8/87

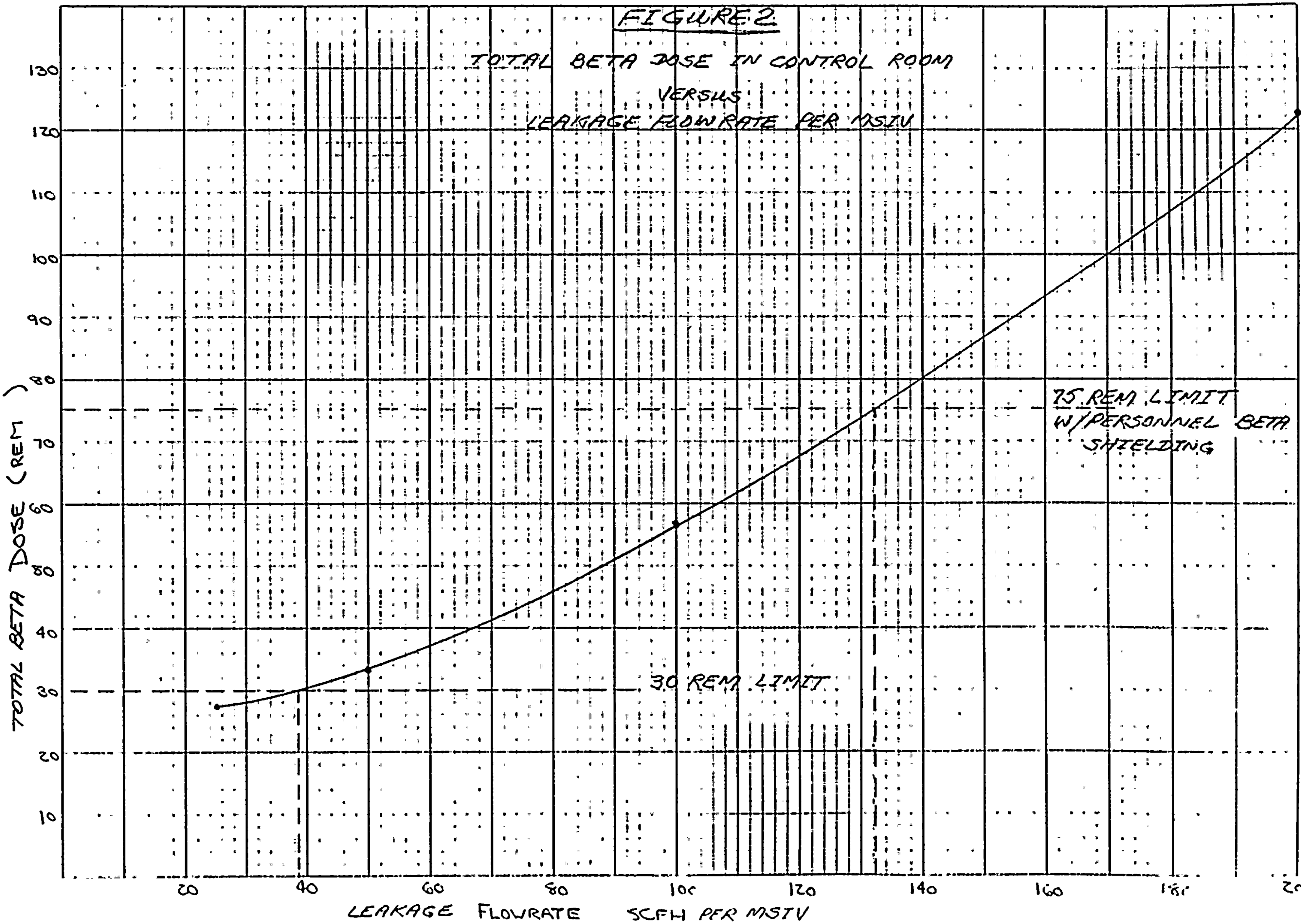
= 122.46 Rem

The above results are plotted in Figure 2. In accordance with Figure 2, Maximum Allowable MSIV Leakage Rate/Valve based on beta dose limit of 30 Rem

= 38 SCFH/MSIV

and, Maximum Allowable MSIV Leakage Rate/valve based on beta dose limit of 75 Rem (with personnel beta shielding)

= 132 SCFH/MSIV



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II. CONTROL ROOM THYROID DOSE EVALUATION

The gamma and beta doses calculated in the preceding section are based on MSIV leakage model that accounts for holdup and decay in the condenser. This model is based on the isolated condenser case of NUREG-1169 (Ref 6). In addition to holdup and decay in the condenser, NUREG-1169 takes credit for iodine plateout and deposition. NMP2 thyroid dose at the control room is calculated using the ratio of the NMP2 thyroid dose at LPZ without iodine plateout credit and the NUREG-1169 thyroid dose at LPZ with iodine plateout credit (isolated condenser case).

NMP2 Thyroid dose at LPZ due to MSIV leakage of 11.5 SCFH/valve without iodine plateout = 59.5 Rem (from Run R2554C01, Job #7192, 1/8/87)

NUREG-1169 Thyroid Dose at LPZ due to MSIV leakage of 11.5 SCFH/valve with iodine plateout model (isolated condenser case) $= 1.2 \times 10^{-3}$ Rem (Table 4.10, Ref 6)

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Ratio of NMP2 Thyroid Dose at LPZ due to 11.5 SCFH/MSIV w/o plateout and NUREG-1169 Thyroid Dose at LPZ due to 11.5 SCFH/MSIV w/plateout

$$= \frac{59.5}{1.2-3} = 4.96+4$$

Note that this ratio is valid based on the assumption (D/A #9)* that the NMP2 plant and the NUREG-1169 reference plant are similar.

NMP2 Thyroid Dose at the control room due to 11.5 SCFH/MSIV w/o plateout

= 27.8 Rem (from Run #R2554C01, Job #7973, 1/9/87)

NMP2 Thyroid Dose at the control room due to 11.5 SCFH/MSIV with plateout model similar to NUREG-1169

$$= \frac{27.8}{4.96+3^{**}} = 5.6-3 \text{ Rem}$$

From Table 4.13 of Ref 6

NUREG-1169 LPZ Thyroid Dose Ratio at 11.5 & 1000 SCFH/MSIV (isolated condenser case)

$$= \frac{4.9\times10^{-1}}{1.2\times10^{-3}} = 4.08+2$$

* CONFIRMATION REQUIRED - See p. 9

** Conservatively reduced by a factor of 10 to account for differences in NMP2 & NUREG-1169 X/Q's (See D/A #4 & 10)

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NMP2 Thyroid Dose at Control Room due to 1000 SCFH/MSIV with iodine plateout model similar to NUREG-1169

$= 5.6 \times 10^{-3} \times 4.08 \times 10^{2}$

$= \underline{2.29}$ Rem

Note that the thyroid dose is evaluated at 1000 SCFH/MSIV leakage which is selected based on the results of Section III.

The above dose is within the allowable increase as calculated below:

Total Thyroid Dose at Control Room

= 23.86 Rem (p. 66, Ref 5)

Thyroid Dose Contribution due to 4 MS lines

= 1.15+1 Rem (p. F2, Ref 5)

Thyroid Dose Limit = 30 Rem (Ref 3 & 4)

Maximum Allowable dose contribution due to MS lines = 30 − (23.86 − 11.5)

$= \underline{17.64}$ Rem

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III EAB & LPZ GAMMA DOSE EVALUATION :

MSIV LKG (SCFH/VLV)	EAB DOSES (REM)			LPZ DOSES (REM)		
	RUN/JOB/DATE	MSIV	TOTAL (MSIV+4.99) (P.29)	RUN/JOB/DATE	MSIV	TOTAL (MSIV+2.41) (P.30)
11 1/2	—	—	—	C01/7192/1-8	2.92-2	2.44 +0
50	C01/4303/1-7	1.32 -2	5.00 +0	C01/6646/1-8	2.30-1	2.64 +0
500	C01/6539/1-8	1.66 +0	6.65+0	C01/6577/1-8	7.74+0	1.02 +1
1000	C01/6192/1-8	6.38 +0	1.14 +1	C01/6283/1-8	2.06 +1	2.30 +1
2000	C01/5908/1-8	2.47 +1	2.97 +1	C01/5985/1-8	5.21 +1	5.45 +1

FIGURE 3 SHOWS THE RELATIONSHIPS BETWEEN MSIV LEAKAGE RATE AND THE RESULTING GAMMA DOSE AT THE EAB AND LPZ USING THE ABOVE DATA. THE GRAPHS SHOW THAT THE LEAKAGE RATES CORRESPONDING TO THE GAMMA LIMIT OF 25 REM ARE AS FOLLOWS:

EAB : 1750 SCFH / VALVE

LPZ : 1080 SCFH / VALVE

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EAB DOSE

TOTAL GAMMA DOSE = 4.99 REM (REF. 5, P. 66)

CONTRIBUTION FROM 4 MAIN STEAM LINES

= 0 REM (REF. 5, P. F4)

ALLOWABLE LIMIT = 25 REM (REF. 5, P. 66)

DOSE DUE TO OTHER CONTRIBUTIONS

= 4.99 - 0 = 4.99 REM

TOTAL THYROID DOSE = 90.14 REM (REF. 5, P. 66)

CONTRIBUTION FROM 4 MAIN STEAM LINES

= 0 REM (REF. 5, P. F4)

ALLOWABLE LIMIT = 300 REM

DOSE DUE TO OTHER CONTRIBUTIONS

= 90.14 - 0 = 90 REM

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LPZ DOSE

TOTAL GAMMA DOSE = 2.49 REM (REF.5, P.66)

CONTRIBUTION FROM 4 MAIN STEAM LINES

= 8.01 - 2 REM (REF.5, P.F3)

ALLOWABLE LIMIT = 25 REM (REF.5, P.66)

DOSE DUE TO OTHER CONTRIBUTIONS

= 2.49 - (8.01 - 2) = 2.41 REM

TOTAL THYROID DOSE = 61.20 REM (REF.5, P.66)

CONTRIBUTION FROM 4 MAIN STEAM LINES

= 2.80 + 1 REM (REF.5, P.F3)

ALLOWABLE LIMIT = 300 REM (REF.5, P.66)

DOSE DUE TO OTHER CONTRIBUTIONS

= 61.20 - (2.80 + 1) = 33 REM

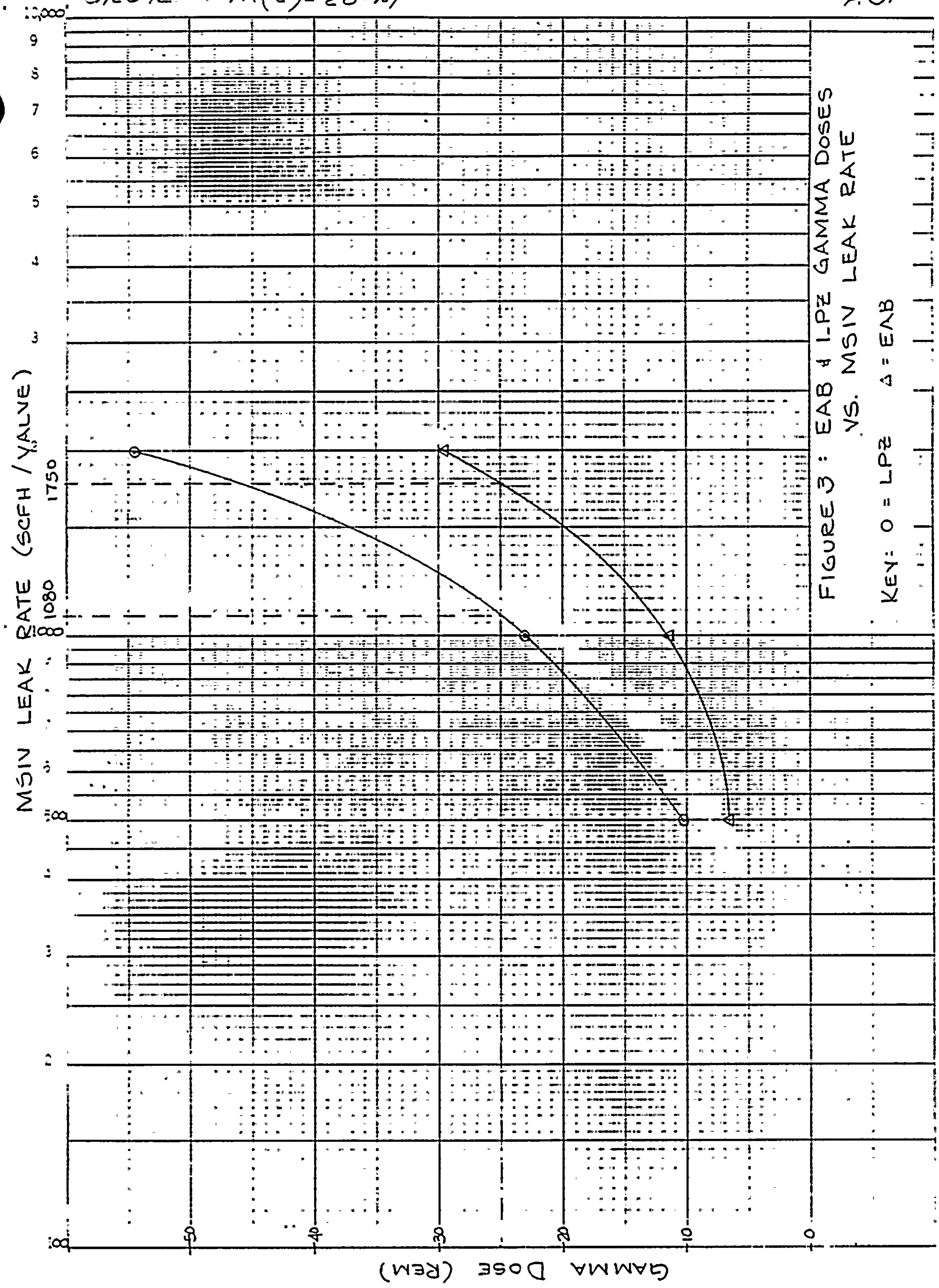
FIGURE 3: EAB & LPZ GAMMA DOSES
VS. MSIV LEAK RATE
KEY: O = LPZ Δ = EAB
MSIV LEAK RATE (SCFH / VALVE)
1750
1080
GAMMA DOSE (REM)
0
10
20
30
40
50
46 4970

IV EAB & LPZ THYROID DOSE EVALUATION:

THE GAMMA (WHOLE BODY) DOSES PRESENTED ON PAGE 28 ARE BASED ON A MSIV LEAKAGE MODEL THAT ACCOUNTS FOR HOLDUP AND DELAY IN AN ISOLATED CONDENSER. IN ADDITION TO USING AN ISOLATED CONDENSER MODEL, THE NUREG 1169 (REF. 6) ANALYSIS ALSO TAKES CREDIT FOR PLATEOUT AND DEPOSITION OF IODINES IN THE CONDENSER AND TURBINE BUILDING AS WELL AS HOLDUP IN THE TURBINE BUILDING. THYROID DOSES THAT ACCOUNT FOR THESE CREDITS CAN BE ESTIMATED BY SCALING THE NMPZ ISOLATED CONDENSER GAMA DOSES BY THE RATIO OF NUREG 1169 THYROID-TO-GAMMA RATIO AS FOLLOWS*:

$$\gamma\Big)_{\substack{\text{CONDENSER}\\ \text{CREDIT}^{\dagger}}} \times \frac{I}{\gamma}\Big)_{\substack{\text{CONDENSER}\\ \text{PLATEOUT CREDITS}}} = T\Big)_{\substack{\text{CONDENSER}\\ \text{PLATEOUT}\\ \text{CREDITS}}}$$

(NMPZ COMPUTER RUN)

EAB: MSIV THYROID DOSE =

$$6.38{+}0 \times \left(\frac{2.1{-}1}{7.4{-}3}\right) = 181 \text{ Rem}$$

(P. 28) (REF. 6, P. 4-24, TABLE 4.13)

* ALL DOSES ARE BASED ON 1000 SCFH/VALVE.

† DELAY IN CONDENSER DUE TO FEED AND BLEED

EAB (CONT'D):

TOTAL THYROID DOSE = (181 + 90 (P.29)) REM

= 271 REM

LPZ: MSIV THYROID DOSE =

$$2.06 + 1 \times \left(\frac{4.9 - 1}{6.5 - 2}\right) = .155 \text{ REM}$$

(P. 28) (REF. 6, P. 4-24, TABLE 4.13)

TOTAL THYROID DOSE = (155 + 33 (P30)) REM

= 188 REM

SINCE THE TOTAL THYROID DOSES AT THE EAB AND LPZ BASED ON A MSIV LEAK RATE OF 1000 SCFH/VALVE ARE WELL BELOW THE LIMIT OF 300 REM, THIS LEAK RATE WILL BE CONSIDERED TO BE A CONSERVATIVE LIMITING LEAK RATE FOR BOTH EAB AND LPZ.

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V TECHNICAL SUPPORT CENTER DOSE EVALUATION

Total Thyroid Dose @ TSC

= 26.0 Rem (p.57, Ref 20)

Thyroid Dose Contribution due to Main Steam Line @ 6 SCFH/MSIV = (1.82-4) + (9.66-5) (p.30, Ref 20)

= 2.79-4 Rem

Thyroid Dose Limit = 30 Rem (Ref 3 & 4)

Maximum Allowable Thyroid Dose due to 4 Main Steam Lines = 30 − {(26.0) − (2.79-4)}

= <u>4.0</u> Rem

If the MSIV leakage is increased from 6 to 1000 SCFH/MSIV, the thyroid dose of 2.79-4 Rem must be increased by a factor of $\frac{1000}{6}$ = 167, assuming a linear relationship of dose increase with respect to MSIV leakage.

Therefore, Thyroid Dose due to 1000 SCFH/MSIV

= (2.79-4) 167

= <u>4.65-2</u> Rem

It can be seen that there is sufficient margin for dose due to main steam lines (= 4.0 Rem) that

even with a non-linear relationship, the dose with a 1000 SCFH/MSIV + the contribution from other sources (= (26.0) - (2.79-4) ≈ 26.0 Rem) will not exceed the limit of 30 Rem. Note that this evaluation did not include credit for holdup and plateout in the condenser allowed for the isolated condenser case of NUREG-1169, rather it took credit for the plateout and holdup model developed for NMP2 in Ref 20.

Total Gamma Dose = 2.93 Rem (p.57, Ref 20)

Gamma Dose Contribution due to 4 Main Steam Lines @ 6 SCFH/MSIV

= (1.25-6) + (1.28-6) (p.30, Ref 20)

= 2.53-6 Rem

Gamma Dose Limit = 5 Rem

Max. Allowable Gamma Dose due to 4 Main Steam Lines = 5 - {(2.93) - (2.53-6)}

= 2.07 Rem

Using the same approach as for the thyroid dose, Gamma dose due to 1000 SCFH/MSIV based on linear

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relationship assumption

= (2.53-6) 167

= <u>4.23-4</u> Rem

Again, it can be seen that there is sufficient margin (allowable dose of 2.07 Rem) that even with non-linear relationship, the dose with a 1000 SCFH/MSIV + the contribution from other sources (= (2.93) - (2.53-6) ≈ 2.93 Rem) will not exceed the limit of 5 Rem. Note that this evaluation did not include the holdup credit in the condenser allowed for the isolated condenser case of NUREG-1169, rather it took credit for the plateout and holdup model developed for NMP2 in Ref 20.

Total Beta Dose = 9.37 Rem (p.32, Ref 20)

Beta Dose contribution due to 4 main steam line @ 6 SCFH/MSIV

= (9.47-5) + (1.33-4) (p.30, Ref 20)

= 2.28-4 Rem

Beta Dose Limit = 30 Rem (Ref 3, 4, & 21)

Max. Allowable Beta Dose due to Main Steam Lines = 30 - {(9.37) - (2.28-4)}

$= 20.63$ Rem

Using the same approach as for the thyroid dose, beta dose due to 1000 SCFH/MSIV based on linear relationship assumption $= (2.28-4)\,167$

$= \underline{3.81-2}$ Rem

It can be seen that there is sufficient margin (allowable dose of 20.63 Rem) that even with a non-linear relationship, the beta dose with a 1000 SCFH/MSIV + the contribution from other sources $(= (9.37) - (2.28-4)$ ≈ 9.37 Rem) will not exceed the limit of 30 Rem. Note that this dose evaluation did not take any credit for the holdup in the condenser allowed for the isolated condenser case of NUREG-1169, rather it took credit for the plateout and holdup model developed for NMP2 in Ref 20.

Therefore, it is concluded that in the case of TSC, maximum allowable MSIV leakage rate is ≥ 1000 SCFH/MSIV based on gamma, beta, and thyroid dose limits.

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RESULTS/CONCLUSIONS

Following are the maximum allowable MSIV leakage flowrates based on applicable regulatory limits at the EAB, LPZ, control room, and the TSC:

CONTROL ROOM

240 SCFH/MSIV (p.21) – Based on Gamma Dose limit of 5 Rem

Total MSIV Leakage = 240 X 4 = 960 SCFH

38 SCFH/MSIV (p. 23) – Based on Beta Dose limit of 30 Rem

Total MSIV Leakage = 38 X 4 = 152 ≈ 150 SCFH

132 SCFH/MSIV (p. 23) – Based on Beta Dose limit of 75 Rem with Personnel Beta Shield

Total MSIV Leakage = 132 x 4 = 528 ≈ 500 SCFH

1000 SCFH/MSIV (p. 27) – Thyroid dose (= 2.29 Rem, p.27) is well within the limit of 30 Rem

Total MSIV Leakage = 1000 X 4 = 4000 SCFH

EAB

1750 SCFH/MSIV (p. 28) – Based on Gamma Dose Limit of 25 Rem

Total MSIV Leakage = 1750 X 4 = 7000 SCFH

EAB

1000 SCFH/MSIV (p. 33) – Total Thyroid Dose (p. 32 & 33) is well within the limit of 300 Rem

Total MSIV Leakage = 1000 X 4 = 4000 SCFH

LPZ

1080 SCFH/MSIV (p. 28) – Based on Gamma Dose limit of 25 Rem

Total MSIV Leakage = 1080 X 4 = 4320 SCFH

1000 SCFH/MSIV (p. 33) – Total Thyroid Dose (p. 32 & 33) is well within the limit of 300 Rem

TSC

1000 SCFH/MSIV (p. 37) – Total Thyroid Dose (p. 34 & 35), Total Gamma Dose (p. 35 & 36), and Total Beta Dose (p. 36 & 37) are well within the applicable code limits

Note that above results are based on some credits taken for non-safety grade structures and equipment in accordance with the isolated condenser case of NUREG-1169.

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CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO. 12177	DIVISION & GROUP PR(C)	CALCULATION NO. 28-N	OPTIONAL TASK CODE NA	PAGE 40

COMPUTER PROGRAM/ANALYSIS IDENTIFICATION

COMPUTER PROGRAMS USED:

PROGRAM NAME: DRAGON VERSION: 05 LEVEL: 00
LIBRARY REFERENCE NO.: NU-115

COMPUTER RUNS:

PROGRAM	RUN #	JOB #	DATE	DESCRIPTION *
DRAGON	R2554C01	3260	1-7-87	CONTROL ROOM, 200 SCFH (DW → CONTROL ROOM)
		4219	1-7-87	CONTROL ROOM, 200 SCFH (DW → TURB. BLDG. → CR)
		5385	1-8-87	CONTROL ROOM, 200 SCFH
		6021	1-8-87	1200 SCFH
		5539	1-8-87	100 SCFH
		5444	1-8-87	800 SCFH
		5430	1-8-87	400 SCFH
		7973	1-9-87	200 SCFH
		4303	1-7-87	EAB, 200 SCFH
		6539	1-8-87	2000 SCFH
		6192	1-8-87	4000 SCFH
		5908	1-8-87	8000 SCFH
		7192	1-8-87	LPZ, 46 SCFH
		6646	1-8-87	200 SCFH
		6577	1-8-87	2000 SCFH
		6283	1-8-87	4000 SCFH
		5985	1-8-87	8000 SCFH

********* CARD IMAGE OF INPUT SUBMITTED TO DRAGON *********

```
CARD COLUMNS           1         2         3         4         5         6         7         8      CARD COLUMNS
              12345678901234567890123456789012345678901234567890123456789012345678901234567890
CARD NO.
    1         **POST-LOCA CONTROL ROOM DOSES DUE TO 200 SCFH RELEASE
    2           7  1010 1 1010    0.0     1.0    0.0     0.0    0.25     0.0    1.0
    3         DRYWELL                  2.85+5                               6.55-3
    4         CONTROL ROOM             3.81+5   750          0.99          1500  0.99          1.0
    5             3.04+0 9.18+7 1.34+8 1.92+8 2.11+8 1.81+8 8.76+7 1.09+7
    6             1.93+7 2.32+7 3.91+7 1.10+7 2.35+7 1.05+6 4.50+7 6.38+7
    7             7.95+7 5.51+5 8.06+6 1.93+8 3.63+7 2.49+7 1.69+8 1.61+8
    8          1        1.0                 1.0         1.0   1.01.29-3          1.0       0.0001
    9          2        1.0                 1.0         1.0   1.01.29-3          1.0          1.0
   10          3        1.0                 1.0         1.0   1.01.29-3          1.0          6.0
   11          4        1.0                 1.0         1.0   1.01.29-3          1.0          8.0
   12          5       0.97                 1.0         1.0   1.09.90-4          1.0         24.0
   13          6       0.91                 1.0         1.0   1.03.37-4          1.0         96.0
   14          7       0.72                 1.0         1.0   1.09.92-5          1.0        720.0

              12345678901234567890123456789012345678901234567890123456789012345678901234567890
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CALC 12177-PR(C)-28-M-0

Run #R254C01, Job #3260, 1/7/87

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* * * * * * * * * * CARD IMAGE OF INPUT SUBMITTED TO DRAGON * * * * * * * * * *

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CARD COLUMNS           1         2         3         4         5         6         7         8     CARD COLUMNS
              12345678901234567890123456789012345678901234567890123456789012345678901234567890
CARD NO.
    1         **POST-LOCA CONTROL ROOM DOSES DUE TO 200 SCFH RELEASE THRU TURB BLDG
    2            7   1110 1 1110    0.0     1.0     0.0     0.0    0.25     0.0    1.0
    3         DRYWELL                    2.85+5                              6.55-3
    4         TURBINE BLDG               2.47+6                               5.068
    5         CONTROL ROOM               3.01+5   750         0.99          1500   0.99            1.0
    6               3.04+0 9.18+7 1.34+8 1.92+8 2.11+8 1.81+8 8.76+7 1.09+7
    7               1.93+7 2.32+7 3.91+7 1.10+7 2.35+7 1.05+6 4.50+7 6.38+7
    8               7.95+7 5.51+5 8.06+6 1.93+8 3.63+7 2.49+7 1.69+8 1.61+8
    9            1         1.0          1.0    1.0          1.0   1.01.29-3          1.0      0.0001
   10            2         1.0          1.0    1.0          1.0   1.01.29-3          1.0         1.0
   11            3         1.0          1.0    1.0          1.0   1.01.29-3          1.0         6.0
   12            4         1.0          1.0    1.0          1.0   1.01.29-3          1.0         8.0
   13            5         0.97         1.0    1.0          1.0   1.09.90-4          1.0        24.0
   14            6         0.91         1.0    1.0          1.0   1.03.37-4          1.0        96.0
   15            7         0.72         1.0    1.0          1.0   1.09.92-5          1.0       720.0

              12345678901234567890123456789012345678901234567890123456789012345678901234567890
```

CALC 12177-PR(C)-28-M-0

Run #R2534C01, Job #4219, 1/7/87

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* * * * * * * * * CARD IMAGE OF INPUT SUBMITTED TO DRAGON   * * * * * * * * *

CARD COLUMNS            1         2         3         4         5         6         7         8       CARD COLUMNS
               12345678901234567890123456789012345678901234567890123456789012345678901234567890
CARD NO.
     1         **POST-LOCA CONTROL ROOM DOSES DUE TO 200 SCFH RELEASE THRU CONDENSER
     2           7  1110 1 1110    0.0     1.0     0.0     0.0    0.25     0.0     1.0
     3         DRYWELL                  2.85+5                          6.55-3
     4         CONDENSER                1.23+5                          5.85-2
     5         CONTROL ROOM             3.81+5   750         0.99      1500   0.99             1.0
     6             3.04+0 9.18+7 1.34+8 1.92+8 2.11+8 1.81+8 8.76+7 1.09+7
     7             1.93+7 2.32+7 3.91+7 1.10+7 2.35+7 1.05+6 4.50+7 6.38+7
     8             7.95+7 5.51+5 8.06+6 1.93+8 3.63+7 2.49+7 1.69+8 1.61+8
     9           1        1.0         1.0    1.0          1.0   1.01.29-3        1.0      0.0001
    10           2        1.0         1.0    1.0          1.0   1.01.29-3        1.0         1.0
    11           3        1.0         1.0    1.0          1.0   1.01.29-3        1.0         6.0
    12           4        1.0         1.0    1.0          1.0   1.01.29-3        1.0         8.0
    13           5       0.97         1.0    1.0          1.0   1.09.90-4        1.0        24.0
    14           6       0.91         1.0    1.0          1.0   1.03.37-4        1.0        96.0
    15           7       0.72         1.0    1.0          1.0   1.09.92-5        1.0       720.0

               12345678901234567890123456789012345678901234567890123456789012345678901234567890
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CALC 12177-PR(C)-20-M-0

Run #R2554C01, Job #5385, 1/7/87

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***** PROGRAM --    DRAGON    -- NU115.VER05.LEV00--  4/30/86 -- *****                                    PAGE  1

* * * * * * * * * CARD IMAGE OF INPUT SUBMITTED TO DRAGON   * * * * * * * * *

CARD COLUMNS            1         2         3         4         5         6         7         8    CARD COLUMNS
               12345678901234567890123456789012345678901234567890123456789012345678901234567890
CARD NO.
    1          **POST-LOCA EAB DOSES DUE TO 2000 SCFH RELEASE THRU CONDENSER
    2            1   1101 1 0001     0.0     1.0     0.0     0.0   0.25    0.0     1.0
    3          DRYWELL                        2.85+5                                  1.0
    4          CONDENSER                      1.23+5                               4.10-1
    5               3.04+0 9.18+7 1.39+8 1.92+8 2.11+8 1.81+8 8.76+7 1.09+7
    6               1.93+7 2.32+7 3.91+7 1.10+7 2.35+7 1.05+6 4.50+7 6.38+7
    7               7.95+7 5.51+5 8.06+6 1.93+8 3.63+7 2.49+7 1.69+8 1.61+8
    8           1       7.07-2         1.0                                 1.90-4         1.0   2.0

               12345678901234567890123456789012345678901234567890123456789012345678901234567890
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CALCULATION 12177-PE(C)-28-M
EWR No. R2554-C01 JOB No. 6539 1/8/87

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* * * * * * * * * CARD IMAGE OF INPUT SUBMITTED TO DRAGON   * * * * * * * * * *

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CARD COLUMNS            1         2         3         4         5         6         7         8   CARD COLUMNS
               12345678901234567890123456789012345678901234567890123456789012345678901234567890
CARD NO.
    1          **POST-LOCA LPZ ROOM DOSES DUE TO 2000 SCFH RELEASE THRU CONDENSER
    2            4   1101 1 0001     0.0      1.0      0.0      0.0     0.25     0.0     1.0
    3          DRYWELL                   2.85+5                                  1.0
    4          CONDENSER                 1.23+5                                4.10-1
    5              3.04+0 9.18+7 1.34+8 1.92+8 2.11+8 1.81+8 8.76+7 1.09+7
    6              1.93+7 2.32+7 3.91+7 1.10+7 2.35+7 1.05+6 4.50+7 6.38+7
    7              7.95+7 5.51+5 8.06+6 1.93+8 3.63+7 2.49+7 1.69+8 1.61+8
    8           1      6.57-2         1.0                                1.78-5           1.0   8.0
    9           2      6.37-2         1.0                                1.19-5           1.0  24.0
   10           3      6.00-2         1.0                                4.93-6           1.0  96.0
   11           4      4.70-2         1.0                                1.40-6           1.0 720.0

               12345678901234567890123456789012345678901234567890123456789012345678901234567890
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CALCULATION 12177-PR(C)-28-M
Run No. E2554C01   Job No. 6577   1/8/87

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CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE 46
12177	PR(C)	28-M	NA	

REFERENCES

(1) US NRC Reg. Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors", Rev. 2, June 1974

(2) US Code of Federal Regulations, Title 10 CFR 100, Revised as of Jan. 1, 1982

(3) US NRC Standard Review Plan 6.4, "Control Room Habitability Systems", Rev 2, July 1981

(4) General Design Criterion 19 of Appendix A, 10 CFR 50

(5) CALC 12177-PR(C)-21-V-5, "Design Basis Loss of Coolant Accident Doses in the Control Room and at the EAB and the LPZ from all Sources", 12/24/86

(6) NUREG-1169, "Technical Findings Related to Generic Issue C-8, Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods", J. N. Ridgely and M. L. Wohl, June 1986

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CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE 47
12177	PR(C)	28-M	NA	

REFERENCES (CONT'D)

(7) CALC 12177-ES-177-5, "Containment Bypass Leakage", 10/22/86

(8) CALC 12177-HVT-40-0, "Turbine Building Volume", 9/10/83

(9) Dwg 12177-EA-2A-6, "Elevations, Sh 1, Turbine Building, Elec. Bay"

(10) Dwg 12177-EA-2B-8, "Elevations, Sh 2, Turbine Building, Heater Bay, Electrical Bay, RR Passage"

(11) Dwg 12177-EA-3A-3, "Wall Sec., Sh 1, Turbine Building"

(12) Dwg 12177-EA-3B-4, "Wall Sec, Sh 2, Turbine Building and Electrical Bay"

(13) Dwg 12177-EA-3C-5, "Wall Sec., RR Passage & Heater Bays"

(14) Dwg 12177-EA-3D-2, "Wall Sec, Sh 4, Turbine Building and Electrical Bay"

(15) Dwg 12177-EA-3E-3, "Wall Sec, Sh 5, Turbine Building"

(16) Dwg 12177-EA-3F-2, "Sections & Details, Turbine Building"

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CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE 48
12177	PR(C)	28-M	NA	

REFERENCES (CONT'D)

(17) Dwg. 12177-EA-43A-4, "Isometrics Sh 1, NW & SW Views"

(18) Dwg 12177-EA-43B-4, "Isometrics Sh 2, NE & SE Views"

(19) NMP2 Preoperational Test Report N2-POT-55, Rev 1 Page 1 of 1 of Attachment 10, dated 11/5/86

(20) CALC 12177-PR(C)-23-C-1, "30-Day Integrated Post-LOCA Doses to NMPNS 1&2 Technical Support Center from Unit 2 Internal and External Airborne Sources Including a Summary of Skyshine & Direct Contributions", 1/13/87

(21) NUREG-0737, "Clarification of TMI Action Plan Requirements", Nov. 1980, Section III.A.1.2

(22) S&W Spec No. (12177) NMP2-P111A, "Steam Surface Condenser Equipment", Add #3 dated 1/6/77, Page 3-3, Section 3

(23) Washington Nuclear Power Station (WNP2) FSAR, Chapter 15

[illegible]