Form 01077 (R7-92)

FORM 101.1	REVISION 14

CERTIFICATION OF ENGINEERING CALCULATION

STATION AND UNIT NUMBER: McGuire Nuclear Station, Units 1 and 2

TITLE OF CALCULATION: Analysis of Consequences Following a DBLOCA with VP in operation in Mode 3 Following the SGR Outage

CALCULATION NUMBER: MCC-1227.00-00-0064

ORIGINALLY CONSISTING OF:

PAGES 1 THROUGH 11

TOTAL ATTACHMENTS 9 TOTAL MICROFICHE ATTACHMENTS /A

TOTAL VOLUMES 1 TYPE I CALCULATION/ANALYSIS? YES [ ] NO [X]

TYPE I REVIEW FREQUENCY N/A

These engineering calculations cover QA Condition 1 items. In accordance with established procedures, the quality has been assured and I certify that the above calculation has been originated, checked or approved as noted below:

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ISSUED TO TECHNICAL SERVICES DIVISION D. miller DATE 11/15/97

RECEIVED BY TECHNICAL SERVICES DIVISION D. Miller DATE 11/15/96

MICROFICHE ATTACHMENT LIST: [ ] Yes [X] No See Form 101.4

REV. NO.	CALCULATION PAGES (VOL)			ATTACHMENTS (VOL)			VOLUMES		ORIG DATE	CHKD DATE	APPR DATE	ISSUE DATE REC'D DATE
	REVISED	DELETED	ADDED	REVISED	DELETED	ADDED	DELETED	ADDED				

1.0 STATEMENT OF PURPOSE

The purpose of t is calculation is to determine the isotopic inventory for the core after steam generator replacement and calculate offsite doses for the following sequence of events postulated to occur at either Unit of McGuire Nuclear Station:

1) Unit 1 or 2 is in Mode 3 as part of the process of being restarted following the Steam Generator Replacement (SGR) outage.

2) The Reactor Coolant (NC) System is at "hot zero power" (HZP) conditions with an average temperature of 557 °F.

3) The Containment Purge Ventilation (VP) System is in operation and fully functional.

4) A Design Basis Loss of Coolant Accident (DBLOCA) with minimum safeguards is assumed to occur.

This work is performed to support a proposed amendment to the Technical Specifications to operate the VP System one time in Modes 4 and 3 following the steam generator replacement (SGR) outage.

1.1 BACKGROUND

As part of the SGR project (SGRP), the supports for the NC System components will be modified. It has been decided that personnel will inspect these modified supports with the Unit at hot zero power conditions to evaluate the supports while they are under thermal loads approaching those associated with full power operations. New insulation will be added to the NC System piping as part of the SGR. As the insulation is heated, it will release to the containment atmosphere poisonous and noxious gasses with the following approximate composition: Ammonia - 76 ppm, Carbon Monoxide - 73 ppm, Acetic Acid - 45 ppm, Methyl Methacrylate - 17 ppm, Carbon Dioxide - 500 ppm. During similar replacements, Calvert Cliffs, Millstone, and Catawba had notable gas problems. The containment can be made habitable with the use of the VP System. Currently, use of the VP System, in lower containment, while in Modes 1 through 4 is prohibited by the Technical Specifications (Ref 8.01, TS 3/4.6.1.9). Therefore, use of the VP System within the restrictions of the TS 3/4.6.1.9 would require that the unit be cooled to Mode 5 conditions. This would add significantly to the SGR outage. Therefore, it is desirable to apply for an amendment to TS 3/4.6.1.9 to allow the VP System to be operated in Modes 4 and 3 for one span of time following the SGR outage.

The VP System is designed primarily to remove radioactivity from the containment and incore instrumentation room. This function is accomplished by exhausting air from the containment and the incore

instrumentation room through filters and replacing it with outside air. The only safety related function is containment isolation. (Ref 8.02, 8.04).

In the Standard Review Plan (Ref 8.05), Branch Technical Position (BTP) CSB 6-4, the NRC has established conditions and limitations regarding the use of containment ventilation systems during power operations (Modes 1 - 4). If these conditions and restrictions cannot be met, then it is required that the containment isolation valves be kept sealed closed. Duke attempted to show that the design of the VP System and associated containment isolation valves met the guidelines of BTP CSB 6-4 (Ref. 8.02. 8.03, 8.05). In Ref. 8.03, Supplement 2, the NRC stated that

> In the Safety Evaluation Report we stated that the applicant had not demonstrated the operability of the upper compartment purge system containment isolation valves; i.e., the capability of the valves to close while experiencing the pressure and temperature build-up within the containment upper compartment during the postulated loss-of-coolant accident. The applicant has confirmed that the design of the upper containment purge system's containment isolation valves meets the requirements of the valve operability program for active valves. Based on our review we conclude that the applicant has demonstrated the operability of the upper compartment purge system's containment isolation valves.
>
> Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," states that the need for purging should be minimized. The applicant has proposed to limit the use only to the upper compartment purge system during the plant operating modes of start-up, power, hot-standby and hot-shutdown, and to limit the use of the upper compartment purge system to less than 90 hours per year (approximately one percent of the time). These limitations will be included in the technical specifications governing the operation of the plant, as well as the requirement that the containment isolation valves in the containment upper compartment purge system be local leak rate tested (Type C) following each use of the system.

McGuire is currently allowed to operate the VP equipment connected to the upper compartment (one supply line and one exhaust line) for up to 250 hours per year.

As part of the attempt to show that the lower VP containment isolation valves would close following a DBLOCA, upper limits to the pressure drops across the inboard and outboard VP containment isolation valves were calculated (Ref 8.06). In the calculation, a value of 8 psig was assumed as the upper bound for the pressure in containment during the time at which the VP containment isolation valves would be required to close. From a review of the FSAR (Ref 8.02), Section 6.2.1.1.3, which specifies 4.0 psig as the compression peak, it is evident that this value is an upper bound to the compression peak. This pressure is calculated based on the

assumption that the air in the containment is compressed into the upper compartment and part of the ice condenser compartment. This gives a peak pressure of 7.67 psig. This method of calculation is a global model and is conservative for the upper compartment. Since the compression effect is generated from the *expansion* of steam flashed from the LOCA and the air it heats, the compression peak model may not be conservative for calculating pressures in subcompartments of the lower compartment in the blowdown phase of the LOCA. In fact, it is expected that peak pressures related to blowdown will occur in the lower containment *subcompartments* within the first few seconds. The analysis of conditions in the containment subcompartments (taken to have very strong spatial dependency) is performed with the computer code TMD (Transient Mass Distribution, cf Ref 8.02, Section 6.2.1.1.3).

2.0 QA CONDITION

This calculation will report the analysis of offsite doses following a LOCA at in either Unit during Mode 3 operations with the VP System in use. If applicable guideline values are met, the results will be used to support an amendment to TS 3/4.6.1.9 to allow the VP System to be used for one span of time while the unit is in Modes 4 and 3 as part of startup operations following the SGR outage. For these reasons, this calculation is classified QA Condition 1.

3.0 DESIGN METHOD

Use of the VP System in Mode 1-4 is prohibited, or restricted, as noted above. This restriction is based on the NRC review of the VP containment isolation valves. Technical justification for a license amendment to operate the VP System once in Modes 4 and 3 following the SGR outage will be based on this calculation.

The ORIGEN2 computer code will be used to calculate core isotopic inventory.

It is expected that the containment conditions for the first few seconds following a DBLOCA would include rapid flow and pressure transients. The pressures would be spatially variant. In particular, pressures in different containment subcompartments (i.e., steam generator enclosures, cold leg accumulator rooms, ice condenser compartments, etc., could vary significantly. Analysis of conditions in the containment in the first few seconds after a LOCA is performed by Westinghouse with the use of the computer code TMD. Some of the results of this analysis are reported in Ref 8.02 and 8.07. In particular, pressures for selected subcompartments are reported for the analysis of a LOCA in the reactor cavity and for the evaluation of ice condenser structural response to a DBLOCA. It is determined that the pressures for the subcompartments interfaced with the VP penetrations calculated with TMD for the latter analysis should be compared to the assumption made concerning containment back pressure in Ref 8.06.

Results from the Catawba analysis of consequences following a DBLOCA with VP in operation in Mode 3 following the SGR outage will be used for VP valve and containment issues. This calculation concluded that:

1) The inboard VP containment isolation valves will close following a DBLOCA. The outboard valves will close following a DBLOCA if their travel is limited to 80° or less.
2) The structural integrity of the Reactor Building will not be challenged following a DBLOCA with the VP containment valves initially open.

Offsite doses and doses to the Control Room operators will be computed.

Consideration was made for temporary installation of debris strainers for the inlet to the VP valves. Debris strainers are intended to protect the isolation valves from the loss of ability to fully close due to LOCA-generated debris entering the purge/vent lines during blowdown. Instead, a conservatively very high containment leakage rate function to provide a conservative upper bound on purge isolation valve leakage. The dose analysis assumes that the containment leaks at 100%/day for the first 24 hours and 50%/day for the remainder of the accident. This assumed leakage rate is considerably in excess of the 0.3%/day leakage rate that would apply if the purge/vent penetrations were sealed.

All leakage except bypass leakage from the containment will be assumed to be directed to the VE filters or the VP filters (cf Assumption 7.04).

No credit is taken for the scrubbing of iodine from the containment atmosphere by the ice condenser (cf Assumption 7.11). Credit is taken for one train of the Containment Spray (NS) System and the Containment Air Return Fans. Doses to the Control Room operators will be computed with the use of the computer code DOSECON (Ref 8.25). Offsite doses will be computed with the use of the computer code DOSESITE (Ref 8.26).

4.0 APPLICABLE CODES AND STANDARDS

4.01 10CFR100, having to do with reactor site criteria.

4.02 10CFR50, Appendix A, General Design Criterion (GDC) 19, having to do with Control Room.

5.0 DESIGN CRITERIA

Doses to individuals at the exclusion area boundary (EAB) at two hours and doses to the boundary of the low population zone for the entire period of passage of the radioactive cloud from the postulated release of fission products shall not exceed 25 rem to the whole body or 300 rem to the thyroid gland (Standard 4.01). Doses to the Control

Room operators following a design basis accident shall remain within the guideline values of 5 rem to the whole body, 75 rem (beta) to the skin, and 30 rem to the thyroid gland.

6.0 APPLICABLE FSAR CRITERIA

The analyses of the radiological consequences of a LOCA are presented in Ref 8.02, Section 15.6.5.3. Analyses of both offsite doses and doses to the Control Room operators are presented.

7.0 ASSUMPTIONS

7.01 Radiological consequences will not be analyzed for the Rod Ejection Accident. One of its characteristics is the release of reactor coolant through the hole left by the ejected rod to the containment. However, it has been determined that there would be no significant source term associated with a rod ejection accident for the Unit at Mode 3 after a shutdown of 70 days (Ref 8.07).

7.02 Radiological consequences will not be analyzed for the Locked Rotor Accident. Analysis of this accident as reported in Section 15.3.3 of the FSAR has shown that the unit response includes an increase in NC System pressure to values at which the pressurizer power operated relief valves (PORVs) and code safety valves (PSSVs) may open. A single failure of one of the valves to close on pressurizer pressure at valve closure setpoint and decreasing would result in a transient induced LOCA. However, the fact that the accident would be postulated to occur with the Unit at Mode 3 at the end of an 70 day outage would reduce the probability of a challenge to either the pressurizer PORVs or the PSSVS. It also has been determined that no significant source term would appear for this accident under the conditions postulated. (Cf Ref 8.07).

7.03 Partial credit is taken for the performance of the VE, VP, and VA Systems for this postulated accident. Containment leakage will be directed to either the VP system or the VE system, following the postulated DBLOCA. Failure of a VP valve to fully close will result in containment leakage through the VP filters. Should the VP system leak, the VE or VA system will be in operation to capture the leakage.

7.04 A single failure resulting in minimum safeguards is assumed. For example, a failure of one train of the Solid State Protection System could be assumed. In addition to leaving the affected unit with one train of Emergency Core Cooling System (ECCS) Pumps, such a failure would cause the VP containment isolation valves associated with the faulted train to fail to close on demand (the Phase A isolation - St - signal).

7.05 The results of the TMD analysis presented in Ref. 8.02, Section 6.2.1.1.3 and Ref.8.10 are used to identify the peak pressures in the

containment subcompartments in which the VP containment penetrations are located. The LOCA is postulated to occur while the Unit is in Mode 3 with NC System pressure of 2250 psia and an average temperature of 557 °F (HZP). The TMD analysis was based on the assumption that a LOCA occurs with the associated unit at full power and the NC System average temperature is 590.8 °F. The results of the TMD analysis would be an upper bound to peak pressures in the same subcompartments following a LOCA with the reactor initially at HZP at the end of an 70 day outage.

7.06 The 12 inch valves in the containment penetration adjoining the Incore Instrumentation Room are assumed to be closed. Current plans to use the VP System to purge noxious and poisonous gases from the containment do not include the use of these penetrations. This Technical Specification submittal will be based on only using three VP containment penetrations.

7.07 Ref 8.02, Section 15.6.5.3 and Table 6-120, specify a value of 2 GPH for filtered ECCS leakage (i.e., leakage from the ECCS Pump Rooms). This analysis uses a bounding value of 1.5 gpm.

7.08 The VP containment isolation valves are assumed to ramp closed with time. That is, the angular position of the VP valves is assumed to be a linear function of time.

7.09 Dry air at 14.7 psia is assumed to be in the annulus at the onset of the DBLOCA.

7.10 In the analysis of radiological consequences of the DBLOCA postulated to occur, the following is assumed to have occurred before the onset of source terms. Ice melt will have been completed. Cold leg recirculation will have been initiated. It has been shown clearly that source terms will not appear until at least 60 minutes (more likely at least 142 minutes) following a DBLOCA (Ref 8.07). For the DBLOCA analyzed in Ref 8.02, ice melt and the transfer to cold leg recirculation will have occurred by then. Although these events may be delayed for a DBLOCA postulated to occur with the unit in Mode 3 immediately following the SGR outage, it is conservative to postulate that they occur before the onset of source terms.

7.11 When they close, the VP containment isolation valves are assumed not to be sealed closed. Based on past experience with these valves, an upper bound is assumed for containment leakage past the valves when they are closed: 100% for the first day of the DBA and 50% per day for the remainder of the accident.

Additional assumptions will be noted in Section 9 as needed.

8.0 REFERENCES

8.01 McGuire Nuclear Station Technical Specifications, with Amendments Through 168/150.

8.02 McGuire Nuclear Station Final Safety Analysis Report, 1995 Update.

8.03 McGuire Nuclear Station Safety Evaluation Report, with Supplements 1 Through 7.

8.04 Design Basis Document MCS-1576.VP-00-0001 (Rev 0), Design Basis Specification for the Containment Purge Ventilation System (VP).

8.05 USNRC, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Rev 3, December, 1988.

8.06 R. Menichelli, Pressure Drop Calculation Across 24" Containment Isolation Valves CNC-1205.02-00-0002, November 22, 1984.

8.07 Analysis of Consequences Following a DBLOCA with VP in operation in Mode 3 Following the SGR Outage, CNC-1227.00-00-0066.

8.08 H.P. Smith, Nuclear Analysis Calculation to Determine Core Conditions After S/G Replacement, CNC-1227.00-00-0067

8.09 J.I. Glenn, Dose Consequence Impact of Mark-BW Fuel Reload for Accidents Analyzed in Chapter 15 of the McGuire FSAR, MCC-1227.00-00-0048 (Rev 4).

8.10 C.D. Ingram, "DOSECON" Computer Code Verification, COM-0204.C6-111-0042 (Rev 1).

8.11 C.D. Ingram, "DOSESITE" Computer Code Verification, COM-0204.C6-111-0042 (Rev 1).

8.12 MCEI-0400-38, Final Fuel Cycle Design, McGuire 1, Cycle 12

8.13 OSC-6183, ORIGEN2 Certification

8.14 MCC-1223.14.00-0008, "ANVENT2 Annulus System Analysis for Setpoint Changes", Rev. 2, 3/01/90.

9.0 CALCULATION METHODOLOGY AND INPUTS

9.1 Isotopic Inventory (ORIGEN2 Computer Code)

The isotopic inventory will be calculated with the ORIGEN2 computer code (Reference 8.13). A batch averaged approach will be used as input to ORIGEN2.

The Final Fuel Cycle Design for McGuire 1 Cycle 12 (Reference 8.12), figure 7, (attachment 1) provides the following inputs:

Region	Enrichment w/o U-235	Cycles Burned	Number of Assemblies	BOC Burnup MWD/MTU
11A	3.45	3	1	32,807
12A	3.60	2	52	27,972
13A	3.40	1	48	17,075
13B	3.55	1	24	15,730
14A	3.67	0	68	0

Heavy metal loading is calculated for each Region, by multiplying 88 metric tonnes by the ratio of assemblies in that region, to the total number of assemblies in the core.

Region	Number of Assemblies	MTU
11A	1	88*(1/193)= 0.456
12A	52	88*(52/193)= 23.709
13A	48	88*(48/193)= 21.886
13B	24	88*(24/193)= 10.943

Total fuel exposure time is calculated with the relationship below :

$$EFPDs = \frac{(MWD/MTU)(MTU)}{MWth} * \frac{(\#\text{ of assemblies in region})}{(\#\text{ of assemblies in core})}$$

Region	MWD/MTU	MTU	MWth	EFPD
11A	32,807	0.456	3411	17.67
12A	27,972	23.709	3411	919.02
13A	17,075	21.886	3411	848.33
13B	15,730	10.942	3411	424.17

The resultant isotopic inventory will be decayed for 70 days (i.e., SGR outage projected duration) as per assumption 7. (ORIGEN2 will decay precursors to daughters through the decay chains).

The ORIGEN2 results (attachments 2 through 5) for the nuclides listed below are:

Nuclide	Core Inventory (Ci)
I-131	9.307E+04
Xe-133	6.346E+04
Kr-85	3.673E+05

9.2 CALCULATION OF RADIOLOGICAL CONSEQUENCES

This section includes a report on the acquisition of the input used for the analyses of offsite doses with the program DOSESITE and doses to the Control Room operators with the computer code DOSECON. Unless otherwise specified, the input for these analyses is taken from Ref 8.09. Also included in this section is a report of radiation doses for the cases analyzed.

The value of the radioactivity released to containment is taken from Section 9.1. These values are 93070 Ci for I-131, 63460 Ci for Xe-133, and 367300 Ci for Kr-85.

By Assumption 7.11, a value of 0 is taken for the efficiency of iodine removal by the ice condensers. This was done for all forms of iodine (diatomic, particulate, and organic forms).

The leakage from the primary containment can be grouped into two categories: (1) leakage into the annulus volume and/or through the VP filters, and (2) leakage directly to the atmosphere (i.e., bypass leakage). Leakage into the annulus is filtered by the Annulus Ventilation System prior to being released to the environment. In this case, the VP valves are not assumed to fully shut, and leakage past these valves will be directed to the VP filters, assuming the VP ductwork in the annulus remains intact, or to the VE system, should a rupture of the VP ductwork occur. This will be modeled in DOSESITE and DOSECON by assuming that non-bypass leakage goes through the VE filters, with VP filter efficiency.

From section 6.24 of reference 8.03,

> We have reviewed the reactor building purge ventilation system for conformance with the guidelines of Regulatory Guide 1.52. In our evaluation, we have assigned the system decontamination efficiencies of 90 percent for elemental iodine, 70 percent for organic iodine and 99 percent for particulates, considering the operating mode for the system. We have determined that the system is capable of controlling the release of radioactive materials in gaseous effluents in accordance with applicable regulations following a postulated design basis accident and is acceptable.

To allow for conservatism, filter efficiencies of 50 percent for each form of iodine have been assigned.

Containment bypass leakage is seven percent. This assumes that VP will serve as containment for any leakage through the VP isolation valves.

The upper bound to the range of containment leak rates selected for these calculations is 100% of the containment volume in the first day (24hr), and 50% of the containment volume per day for the remainder of the accident.

By Assumption 7.07, the ECCS leakage rate is set to

(1.5 GPM)(60min/hr)(77/576)(ft3/gal)(30.48cm/ft)^3 = 3.4E5 cc/hr.

By Assumption 7.10, this is taken to apply for $t \geq 822$ seconds.

Using a containment volume of 1217900 ft^3 (Ref 8.02, Chapter 6), and a VP containment penetration leak rate of 1 containment volume per day, the annulus inleakage rate is:

(1217900ft^3/day)/[(24hrs/day)(60min/hr)] = 845.76 ft^3/min

This value is well within the ability of the VE system to remove, and maintain a negative pressure of -.25" w.g. in the annulus.

From Reference 8.15, the VE exhaust (without the increased inleakage of this scenario) averages 4000 cfm for the first 42 minutes of activation, and then averages 810 cfm for the remainder of the accident. To account for the increased inleakage from containment, the VE exhaust will be modeled as 4846 cfm for the first 42 minutes, and then 1656 cfm for the remainder of the accident.

Offsite doses were calculated for two cases. Attachments 6 and 7 document these results. Minimum safeguards were assumed for the "base case". Whole body doses (noble gases) are 6.92x10-2 Rem at the exclusion area boundary (EAB) and 2.19x10-2 Rem at the boundary of the low population zone (LPZ). Thyroid doses are 88.3 Rem at the EAB and 18.5 Rem at the PZ. From Standard 4.01, guideline values for offsite doses following a design basis accident (DBA) are 25 Rem to the whole body and 300 Rem to the thyroid gland. These limits apply to both the EAB and the LPZ.

Offsite doses also were calculated for a case in which no credit was taken for the VA filters. This was done by setting VA filter efficiency to 0 for all iodine species. Whole body doses were slightly higher than the minimum safeguards case: 6.93x10-2Rem at the EAB and 2.19x10-2Rem at the LPZ. Thyroid doses for this cases were computed to be 88.6 Rem at the EAB and 18.7 Rem at the LPZ.

Doses to the Control Room operators were computed with use of the computer code DOSECON. Attachments 8 and 9 document these results. For the base case, minimum safeguards was assumed. We have assigned the system decontamination efficiencies of 99 percent for elemental iodine, 99 percent for organic iodine and 99 percent for particulates, for the Control Room Ventilation (VC) System. This value, taken for all species of iodine, is judged to be conservative

given values of IPF associated with past flow balances of the VC System. Doses to the Control Room operators were computed to be 1.22x10-2 Rem to the whole body, 5.39 Rem to the skin, and 5.61 Rem to the thyroid gland.

Radiation dose to the Control Room operators was calculated for an additional cases. No credit was taken for the VA System filters. This was done as shown above. Whole body doses were 1.23x 10-2 Rem, while thyroid doses were 5.73 Rem. From Standard 4.02, the guideline values for doses to the Control Room operators following a DBA are 5 Rem to the whole body, 75 Rem to the skin, and 30 Rem to the thyroid gland.

In all cases, it is seen that doses to the EAB, the LPZ, and the Control Room operators are well within guideline values. This is attributed to the fact that the source term for this event is very small compared to the source term calculated for a DBLOCA with the affected unit initially at full power and at the end of cycle.

10.0 CONCLUSIONS

The radiation doses to the exclusion area boundary, the low population zone, and the Control Room Operators following a DBLOCA with the associated unit in Mode 3 immediately following an 70 day outage have been computed. In all cases analyzed, radiation doses were calculated to be within guideline values.